Final Storage of Spent Nuclear Fuel – KBS-3

SKBF/KBS

Final Storage of Spent Nuclear Fuel – KBS-3

IV Safety

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17 SAFETY PRINCIPLES

This chapter deals with the principles that have served as a basis for evaluating the safety of the handling and final storage systems.

17.1 GENERAL PREMISES

The following basic principles for the final disposal of spent nuclear fuel can be derived from current legislation in Sweden.

- A very high level of long-term safety is required.

- It shall be possible to carry out the necessary measures with the highest possible degree of national independence.

- Burdens on future generations shall be avoided.

These basic principles lead to a number of criteria

- The level of safety in connection with the handling and final storage of spent nuclear fuel shall lie within the limits laid down in national and international standards and recommendations

- The long-term safety of the overall system shall be based on a number of independent barriers so that total safety is not jeopardized by deficiencies in one of the barriers

- The safety function of each barrier should be assessed cautiously

- The design and layout of the final repository shall be based on knowledge of possible sites in Sweden and on a technology that is available within the country

-    The long-term safety of the final repository may not be based
     on supervision and opportunity to adopt measures in the sealed
     repository

-    The final repository shall not alter the natural radiation
     conditions in the region in the short or long term

The possibility of establishing quantitative acceptance criteria
for individual barriers or groups of barriers is currently being
discussed in certain countries. See /17-1/. Such criteria can be
formulated as minimum life of the canister, minimum travel time for
the groundwater, maximum leakage from the repository to the geo-
sphere etc. No similar effort has been made in the KBS work, since
such an approach would preclude the possibility of evaluating and
optimizing the overall function of a studied barrier system in its
entirety. At the present time, safety is best demonstrated by
showing that some combination of barriers at some site offers
possibilities, with margin, for a final disposal with acceptably
low impact on the environment.

Occupational safety is being ensured by prior review of the handling
operations and the systems but above all by evaluating the actual
operation of the facility after it has been commissioned. Protection
of the public and the environment is affected by the releases/leak-
age that can take place from the facility both during the operating
period and after sealing. During the operating period, it is pos-
sible to control the release situation continuously by means of
various active measures.

After the facility has been sealed, however, there is no longer any
way to influence leakage. Naturally, this affects the basis for the
safety evaluations. Discussions of these matters are being held
within a number of international agencies. As yet, however, no
special guidelines or regulations have been established for the
long-term storage.

17.2   QUALITY REQUIREMENTS AND QUALITY ASSURANCE

The importance of different material properties and the durability of the materials has been dealt with in preceding chapters, mainly 9, 10 and 11, where quality control was also examined. The purpose of this quality control is to provide assurance that there are no deviations from the desired quality that can significantly impair the safety function. A systematic programme of quality control, of the type applied within the nuclear power sector, will also be applied to the design, manufacture and construction of the various parts of the repository system.

For the facilities involved here, the occupational safety will be of great importance, along with protection against ionizing radiation. For the construction and operation of the facilities, extensive experience exists from similar work in mines, power stations and the mechanical engineering industry, where regulations and established forms for occupational safety exist. These matters are therefore not dealt with here.

17.3   RADIATION PROTECTION PRINCIPLES

Occupational safety in radiological work is dealt with thoroughly in international recommendations, primarily those issued by the International Commission on Radiological Protection (ICRP), and a national system of rules has been established by the Swedish National Institute of Radiation Protection (SSI).

International agreement exists on the principles of protecting human beings from the effects of radiation. Both national authorities and international organizations apply the following principles:

-    No practice that involves a radiation shall be adopted unless it can be shown to entail greater advantages than disadvantages from the viewpoint of society.

-    No individual shall receive radiation doses that exceed the limits recommended by the ICRP.

- All exposures shall be kept as low as reasonably achievable, economic and social factors taken into account (the ALARA principle).

As yet, there are no guidelines for how these principles should be applied with regard to long-lived radioactive waste with a potential for radiological effects far into the future.

17.4 RADIATION PROTECTION STANDARDS

17.4.1 Internationally recommended dose limits

The ICRP's recommendations aim at limiting somatic effects in the individual, hereditary effects in the next few generations and hereditary and somatic effects within the population as a whole.

The ICRP issues the following recommendations concerning maximum permitted radiation dose (aside from background radiation and planned irradiation for medical purposes).

Personnel in radiological work   50 mSv/y

Most exposed group among the public

- Occational exposure   5 mSv/y

- Prolonged exposure   1 mSv/y

17.4.2 Swedish standards and guidelines regarding radiation doses

The National Institute of Radiation Protection in Sweden (SSI) applies the international recommendations issued by the ICRP. Beyond this, SSI has laid down the following goal for the design of nuclear power facilities /17-2/. "The facilities shall be engineered with the objective that the sum of all committed dose equivalents should be less than 0.1 mSv/y to the most exposed group among the public."

SSI has further stipulated that 0.5 mSv should not be exceeded in the actual operation of the facility.

In evaluating other parts of the nuclear fuel cycle, SSI requires an optimization of the radiation protection efforts in the sense expressed by the ICRP in its Publication No. 26, i.e. it shall be shown that the radiation doses cannot be further reduced by reasonable efforts.

In order for such an optimization to be meaningful, it is necessary to make a realistic estimation of the difference in the total health consequences of different final disposal alternatives. These health consequences (known as the "detriment" of the practice, according to the terminology used by the ICRP and by the SSI) can be considered to be proportional to the total collective dose commitment from the final repository.

In using the concept of collective dose in connection with the final storage of spent nuclear fuel, however, special problems are encountered. This is due to the fact that the waste contains radioactive substances with long half-lives. The collective dose commitment can then be large without any individual being subjected to appreciable radiation doses or any future generation being exposed to significant risk. Moreover, the doses are to a large degree caused by radioactive substances in nuclear fuel that were originally present in the natural uranium from which the fuel was fabricated. No guidelines exist today for the evaluation of such matters.

Nor do any norms exist on how the dose consequences of improbable events are to be judged. The possibility of weighing in the probability that the situation will arise in the evaluation has been discussed internationally.

17.4.3   Directives for radiological work etc.

SSI issues regulations and directives for radiological work in Sweden. Since the handling of the spent fuel for a final storage is similar to that at nuclear power stations and the CLAB, it has been

assumed that the directives will largely coincide with directives for such facilities.

Special regulations based on recommendations issued by the IAEA apply to the transport of radioactive materials. /17-3/

Special regulations also apply to protection against radon in underground facilities. /17-4/

17.5 SAFETY-RELATED GROUNDS FOR EVALUATION OF FUNCTION OF REPOSITORY SYSTEM

The repository system has been designed primarily to meet the requirements of radiation protection.

In evaluating the final storage scheme, it should also be borne in mind that radioactive substances occur in nature and that ionizing radiation from these substances are a part of man's natural environment. The natural background radiation in Sweden varies between 0.7 and 1.4 mSv/y. At certain places abroad, natural radiation levels exceed these values by a factor of more than ten. Of particular interest in this context is the occurrence of those natural radioactive substances in water that can also occur in the releases from the final repository.

The following guidelines have been used here for the assessment of whether the repository system can be regarded as being acceptably safe.

- The expected contribution to the radiation dose to the most highly exposed group shall be less than SSI's design goal for nuclear power facilities of 0.1 mSv/y.

- The repository's contribution to the radiation dose to the most highly exposed group among nearby residents shall, even under very unfavourable circumstances, be less than the limit recommended by the ICRP of 1 mSv/y.

- The final repository shall not essentially alter the natural radiation environment in the repository's surroundings.

Collective doses are reported for the purpose of illustration, but have not been credited with any value for consequence comparisons, since the repository system has not been optimized.

18   SITE-SPECIFIC DATA

| SPENT FUEL | → | FACILITIES | → | NEAR FIELD | → | GEOSPHERE | → | BIOSPHERE | → | SAFETY |

SPECIFIC DATA RELATED TO LOCATION
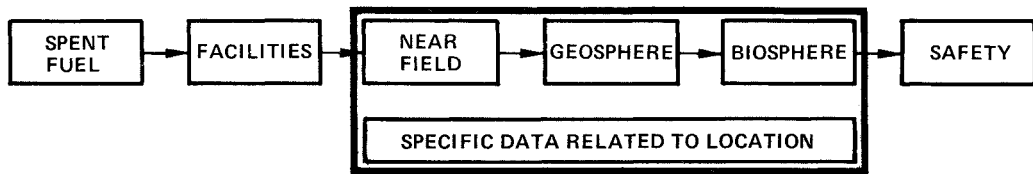
Geological, hydrological and chemical data from the study sites at Fjällveden, Svartboberget, Gideå and Kamlunge are presented in this chapter. The data obtained serve as the basis for calculations of groundwater conditions within each site. The general characteristics of the sites as regards climate, hydrology, ground conditions, vegetation and recipient conditions are described. In addition, selected results from investigations at other sites are presented. The site locations are shown in Fig. 18-1.
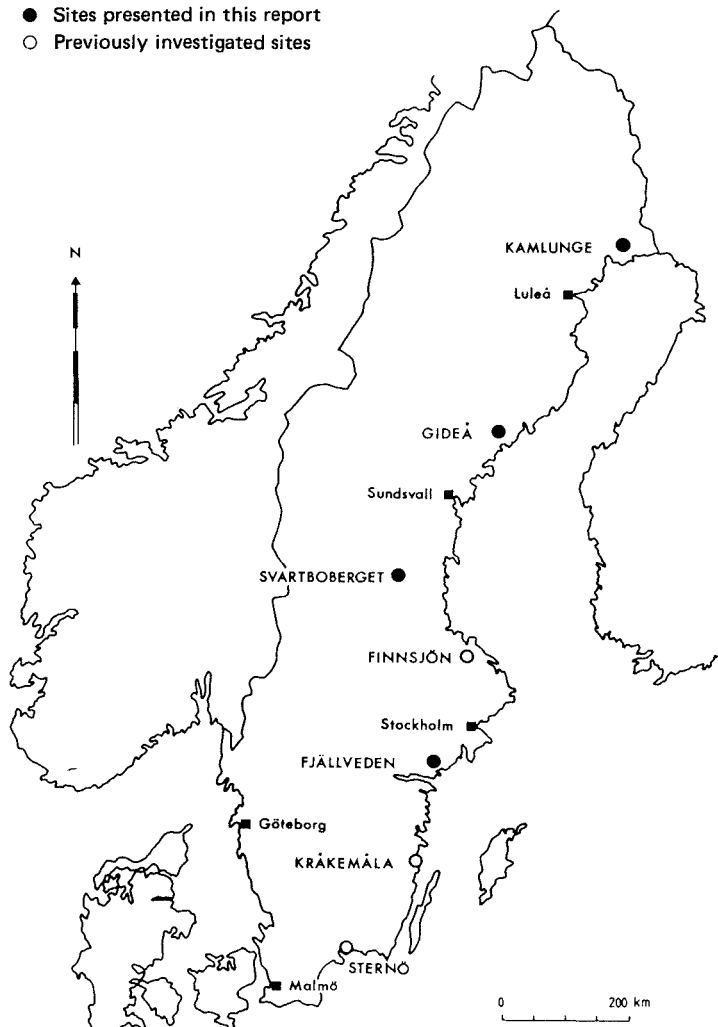
Figure 18-1. Study sites.
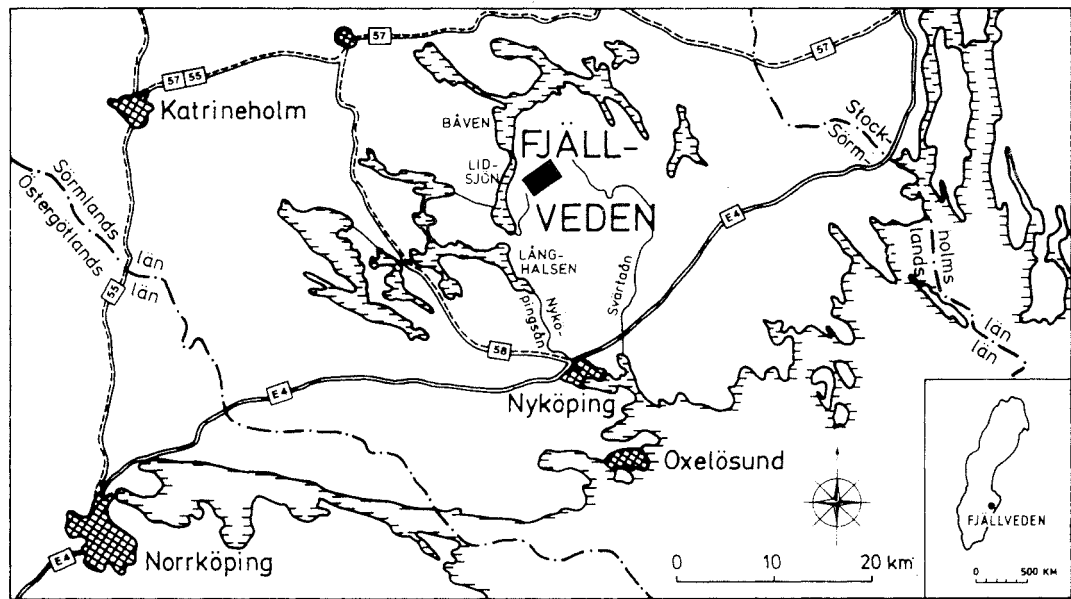
Figure 18-2. Locality map for the study site at Fjällveden.

18.1 GENERAL

The general programme for geological, hydrological and geochemical investigations carried out within the respective sites is described in chapter 5. Each site has specific conditions with regard to rock types, fracture zones, hydrology and groundwater chemistry. Hence, the scope of the investigations has varied among the sites.

The site-specific data presented for the sites at Fjällveden, Gideå, Kamlunge and Svartboberget include the information required for model computations of groundwater conditions and for schematic positioning of an intended repository. These data are:

- topography
- rock types and ore mineralizations
- regional and local fracture zones
- fracture frequencies
- hydraulic conductivity in different rock types and fracture zones
- groundwater chemistry

The repository will be located in the rock mass, between fracture zones. The distance to the nearest fracture zone or section of
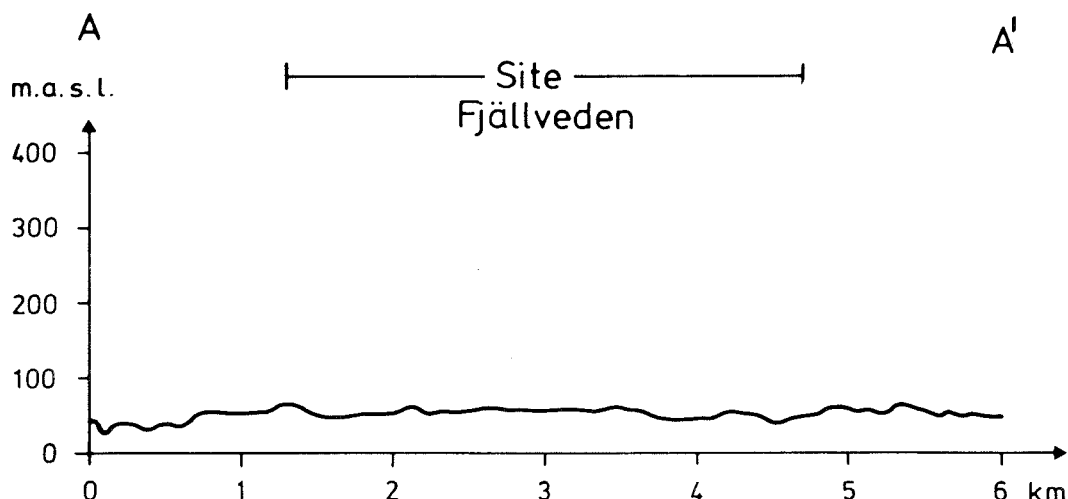
Figure 18-3.   Topographical profile through the site at Fjällveden. Location shown in figure 18-4.

rocks with high hydraulic conductivity - the "respect distance" - has been chosen as about 100 m. Depending upon the distance between fracture zones, the repository may comprise one or several storeys.

18.2   FJÄLLVEDEN

18.2.1   Location and topography

The study site at Fjällveden is located about 80 km SW of Stockholm and about 20 km NNW of Nyköping (Fig. 18-2). This part of Söderman-land is characterized by a flat topography with minor, dominantly NW-oriented valleys. Fjällveden lies between two such valleys 3 km apart. The relief within the site is small. The highest point in Fjällveden is 76 metres above sea level, while the valleys on either side are at 40 metres above sea level. A topographical profile through the site is shown in Fig. 18-3.

The site is forested with small marshes. The Quatarnary deposits are mainly moraine, in low depressions normally covered by clays. About 10% of the site comprises rock outcrops.
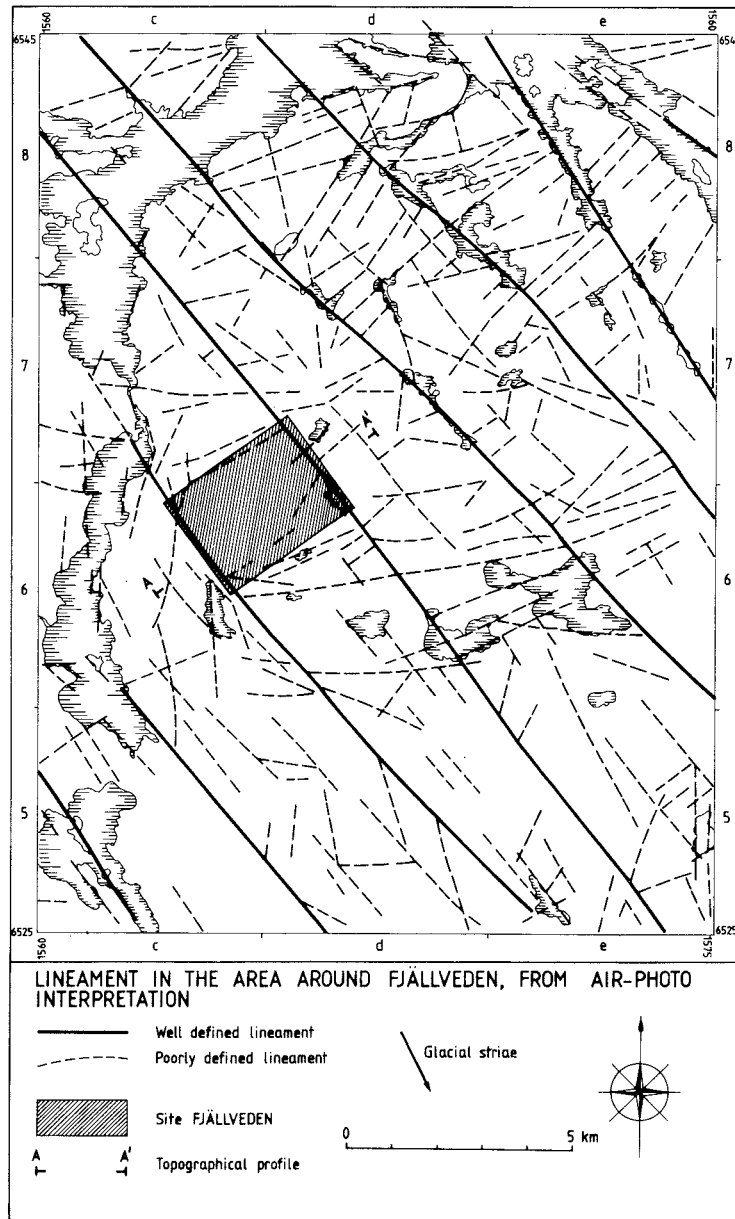
Figure 18-4. Interpreted lineaments in area around Fjällveden.

A 2.7 by 3.7 km area within Fjällveden, including the two fracture vallyes, was investigated in detail (Fig. 18-4). The results of these studies are presented in /18-1/.

18.2.2 Bedrock geology

The main rock type within the site at Fjällveden is veined gneiss. This rock formed through metamorphism of sand and mud strata under high pressure and temperature to produce the appearance and composition the rock has today. This metamorphism is about 1 800 million

years old /18-2/ and involved partial melting of the bedrock, with recrystallization and neomineralization (migmatization). The rock is characterized by veins and other irregular bodies of varying mineral composition. These veins generally strike NE and are usually vertical.

The essential minerals in the veined gneiss are quartz 50%, potash felspar 25%, biotite 15% and plagioclase 10%. Sulphide minerals, usually pyrrhotite and iron pyrites, occur as small mineral concentrations or fracture fillings. Ore minerals are present in such low quantities that the area hs no potential for mining. For example, the highest analysis value for copper is 0.01%.

There are also other small bodies of a dark basic rock, amphibolite, within the site. These appear as elongated bands in the veined gneiss, parallel to the gneissic structure.

Granite gneiss occurs in a larger, coherent body in the site's NW part and in a smaller area in its SW part. It has been affected by the same metamorphism and deformation which has also shaped the veined gneisses. Granite gneiss has been found at 27 levels in cores from boreholes Fj 1 - Fj 13, mainly as thin vertical layers parallel to the gneissic structure. The width ranges from a few dm up to around 15 m. The layers have a greater persistence laterally than vertically. Of a total core length of 7 334 m, granite gneiss makes up 179 m (2.4%).

18.2.3   Fracture zones

A total of 15 core (diamond) boreholes and 49 percussion holes have been drilled in Fjällveden. For the core boreholes, dip and length are given in Table 18-1. The locations of the core boreholes are shown in Fig. 18-5.

Fjällveden is bounded on the SW and NE by regional fracture zones (Fig. 18-4). These fracture zones run with a spacing of 2.5-3 km. Within the zones are rocks which have been crushed but now form mylonites and breccias, as well as parts that are crushed and altered to clays. The regional fracture zone NE of Fjällveden is

Table 18-1.   Summary of length and dip of core (diamond) drill
boreholes on site at Fjällveden.

Borehole	Dip (degrees)	Borehole length (m)	Borehole depth (m)
Fj 1	75	711.4	695
FJ 2	60	700.7	575
Fj 3	60	426.1	370
FJ 4	60	700.5	585
Fj 5	60	700.4	570
Fj 6	60	702.6	590
Fj 7	60	760.4	660
Fj 8	60	731.8	615
Fj 9	50	700.4	520
Fj 10	45	199.0	140
Fj 11	60	250.6	211
Fj 12	60	150.4	130
Fj 13	80	151.3	146
Fj 14	60	350.1	298
Fj 15	50	355.4	275

80-90 m wide and dips about 75° SW, i.e. in towards the site. Fracture zones of similar character have been observed in connection with mapping of the geological mapsheet Nyköping SO /18-2/. The occurrence of diabases in these fracture zones shows that the zones are more than 1 200 million years old.

The regional fracture zones confine a 9 km^2 block. Local fracture zones of lesser width and persistence occur within this block. These zones display both limited continuity and little contrast against the surrounding rock mass with regard to hydraulic conductivity. The distance between these zones is 300-900 m (Fig. 18-5). Indications of local fracture zones, obtained from geophysical ground surveys and from topographical relationships, have been investigated by percussion boreholes. Only in a few cases have these indications of fracture zones turned out to be fracture zones
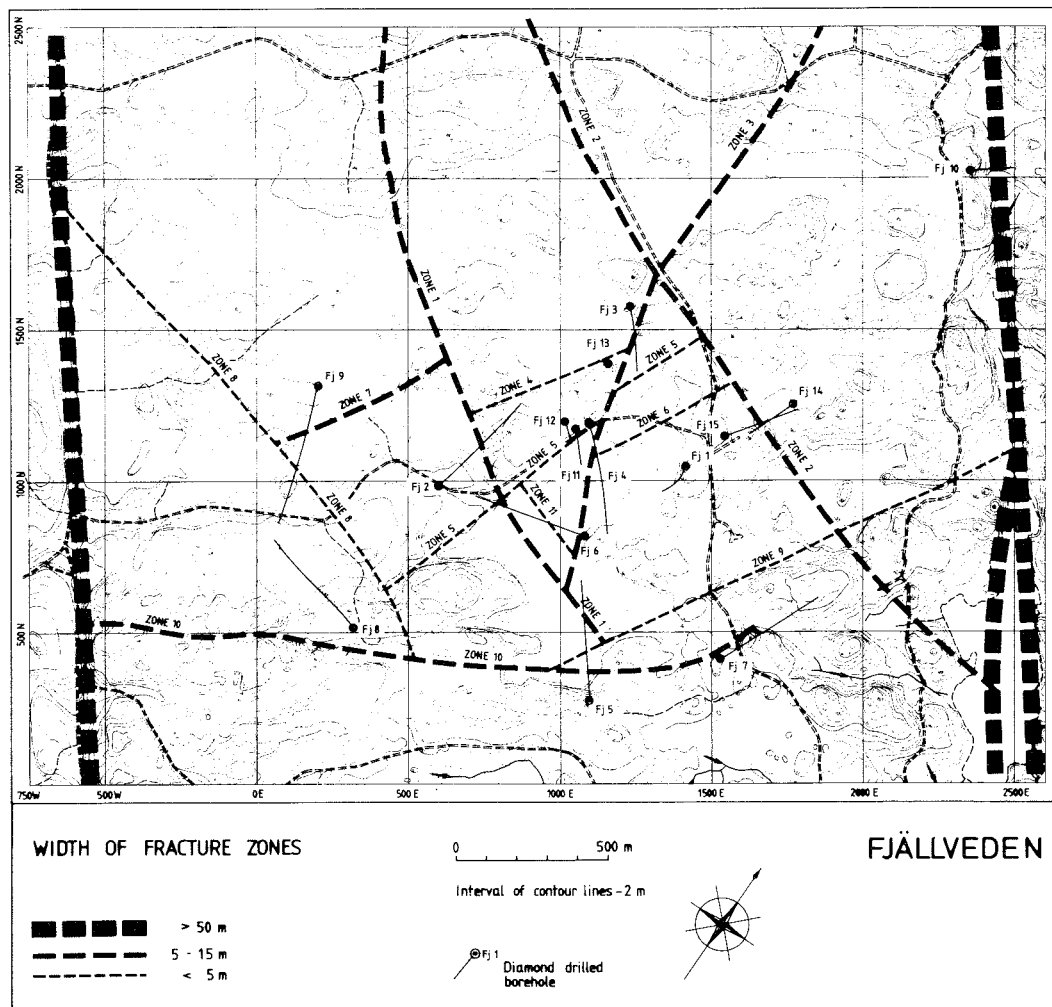
Figure 18-5. Fracture zones at the surface within the study site at Fjällveden.

in the boreholes. The indications have instead resulted from dis-
turbances caused by electrically conductive clays or by variable
weathering of different layers in the gneiss. No horizontal fracture
zones have been found on the site.

The fracture zones indicated within the central part of Fjällveden
have been penetrated by core boreholes at a total of 21 different
places. The width of the fracture zones varies from 0.2 to 14 m,
with an average width of 5 m. The width of the fracture zones was
determined in the borehole from the point at which fracture fre-
quency increases markedly to the point where it returns to its
normal value. To calculate the true width, a correction was then
made for the angle made by the borehole in relation to the fracture
zone.
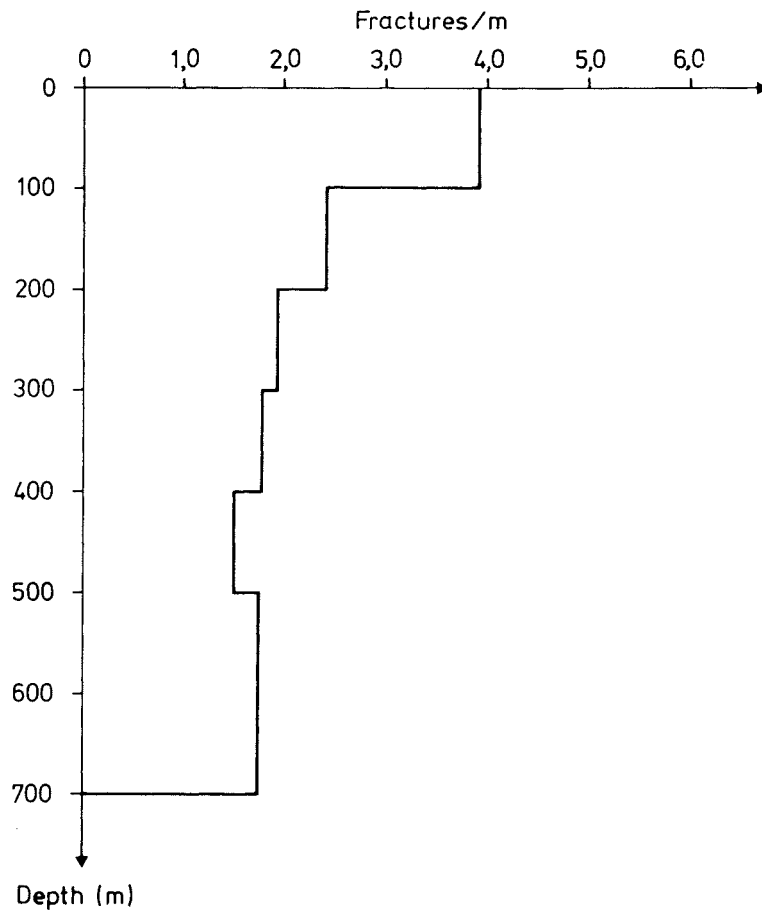
Figure 18-6. Fracture frequency in the rock mass within the site at Fjällveden.

Within the local fracture zones are sections with brecciated, weathered and clay-altered rock. Commonly occurring fracture minerals are calcite, kaolinite, chlorite and illite. Crushed sections in the fracture zones have to a large extent been sealed by the formation of clay minerals. A brief resumé of the fracture zones is presented in Table 18-2.

18.2.4   Fracturing of the rock mass

Fracturing in the rock mass was mapped partly from outcrops and partly from cores. Fractures observed on the surface have two main directions, NE and NW, i.e. parallel with and perpendicular to the gneissic structure. The frequency of fractures longer than 0.5 m in outcrops is 0.9 fracture/m.

Table 18-2.   Resumé of fracture zones in Fjällveden.

Fracture zone	Position in in borehole (m)	Dip (degrees)	True width (m)	K value m/s
1	Fj 2 (340-354)	90	7	3×10^{-9}
	Fj 6 (479-486)	90	3	8×10^{-10}
	Fj 5 (469-473)	90	1	2×10^{-9}
2	Fj 14 (115-134)	80 NO	12	not measured
	Fj 15 (304-321)	80 NO	9	not measured
3	Fj 3 (150-175)	90	5	3×10^{-7}
	Fj 4 (140-192)	90	10	1×10^{-7}
	Fj 6 (37- 59)	90	11	not measured
4	Fj 2 (596-600)	80 SO	1	7×10^{-9}
5	Fj 4 (61- 63)	80 NV	1	1×10^{-6}
	Fj 6 (610-611)	80 NV	0.5	1×10^{-11}
	Fj 11 (64- 66)	80 NV	1	not measured
	Fj 12 (99-101)	80 NV	1	not measured
6	Fj 1 (674-676)	75 SO	0.2	1×10^{-10}
7	Fj 9 (110-130)	60 NV	14	2×10^{-8}
8	Fj 9 (424-433)	90	4.5	5×10^{-8}
9	Fj 5 (173-185)	75 SO	5	5×10^{-7}
	Fj 7 (685-731)	75 SO	5	1×10^{-10}
10	Fj 5 (96-102)	70 SO	5	2×10^{-9}
	Fj 7 (53- 89)	70 SO	6	not measured
11	Fj 6 (245-256)	90	3	not measured
Regional eastern zone	Fj 10 (70-165)	75 SV	90	1×10^{-6}
Regional western zone	-	75 SV*	90*	-

* Calculated from geophysical observations.

The variation in fracture frequency with depth in the rock mass between the fracture zones is shown in Fig. 18-6. The data presented comprise average values from all the boreholes. Veined gneiss and granite gneiss have similar fracture frequency.

The fracture frequency is highest in the two uppermost 100 m inter-
vals. Deeper in the rock mass, the fracture frequency is 1.8 frac-
tures/m. The higher fracture frequency for cores in comparison with
outcrop measurements results from underestimation of the frequency
of horizontal fractures in outcrop mapping. In addition, the frac-
ture frequency for cores includes all fractures regardless of
length, in contrasts to the outcrop mapping where all fractures
shorter than 0.5 m have been omitted.

Kaolinite is a common fracture mineral in the rock mass as well.
The wide occurrence of this mineral indicates that Fjällveden's
groundwater has or has had chemical properties which promoted
alteration (weathering) of the mineral plagioclase to kaolinite.

18.2.5   Hydrology, meterology and recipient conditions

Fjällveden lies on the watershed between the catchment of the
rivers Nyköpingån to the west and the Svärtaån to the east. Both of
these rivers discharge 20 km away into the Baltic Sea. There are
several large lakes outside the site. In the two confining fracture
valleys are the lakes Ålskäggaren (0.13 km^2), Sågsjön (0.18 km^2)
and Morpasjön (0.4 km^2).

Morpasjön belongs to the Nyköpingån's runoff area and has a catch-
ment area of 4.5 km^2. The theoretical water turnover time, calcula-
ted as lake volume divided by catchment, is 3.5 years. Other large
lakes located downstream within the Nyköpingån's runoff area have a
theoretical turnover time of about 5 months. At the Nyköpingån's
outflow into Stadsfjärden, the turnover time is about 8 months.

The location of the site on the watershed means that it forms an
inflow area for groundwater. Low-lying parts within the site com-
prise local outflow areas for shallow groundwater /18-1/. Morpasjön
is the first major recipient for surface water and groundwater in
Fjällveden. The natural concentrations of uranium and radium in the
lake are given in Table 18-3.

Mean precipitation within Fjällveden is 650 mm/y. Of that total 20% falls as snow. Evaporation is estimated at 450 mm/y /18-3, 18-4/. Runoff is estimated at 200 mm/y. This amount constitutes the maximum groundwater recharge within the site. According to /18-5/, the groundwater recharge within a clay-outcrop-moraine area can be estimated at 10-20% of annual precipitation, i.e. 65-130 mm/y for Fjällveden.

Table 18-3.   Measured concentrations of uranium and radium at Morpasjön Lake.

	Uranium	Radium	Unit
Sediment	200	100	$Bq\ kg^{-1}$
Lake water	0.005	<0.004	$Bq\ 1^{-1}$
Soil	750	70	$Bl\ kg^{-1}$

18.2.6   Hydraulic conductivity of the bedrock

The bedrock in Fjällveden has been divided into different hydraulic units (chapter 6) as follows:

- regional fracture zones (2)
- local fracture zones (11)
- the rock mass

The location of the regional and local fracture zones is shown in Fig. 18-5. Hydraulic conducitivty, K, has been measured in 219 sections 25 m long and 61 sections either 5 or 10 m long in bore-holes Fj 1-Fj 9. In addition, the hydraulic conductivity in the site's eastern regional zone was determined by measurement in borehole Fj 10. All measured values representing rock mass and local fracture zones are shown in Figs. 18-7 and 18-8, respectively.

Granite gneisses in the rock mass have higher hydraulic conducti-vity than the veined gneiss. In Fig. 18-7 the relationship between depth and hydraulic conductivity has been plotted for the rock mass as a whole, and for granite gneiss and veined gneiss spearately.
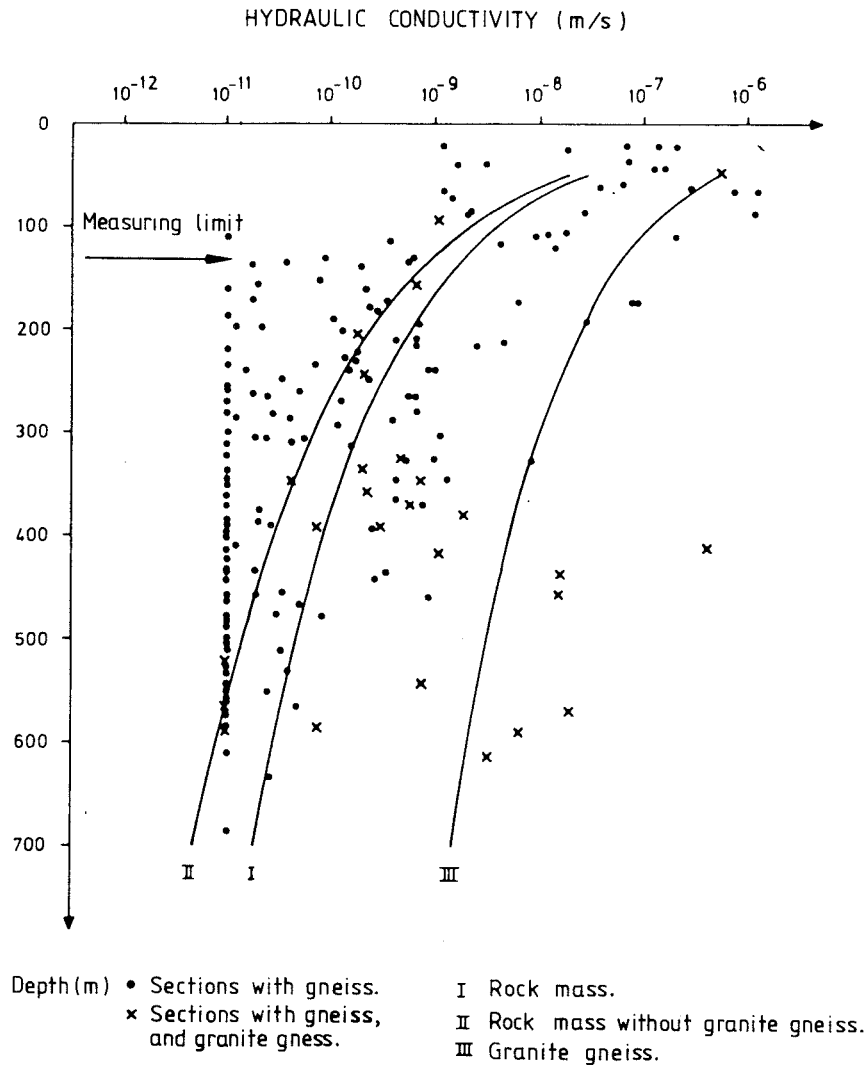
Figure 18-7. Correlation between hydraulic conductivity and depth within the rock mass at Fjällveden.

The correlation curve for granite gneiss was determined from K values calculated with regard to the proportion of granite gneiss in each measurement interval. Fig. 18-8 shows the equivalent relationship for local fracture zones. The correlation applies down to 700 m.

To determine the hydraulic fracture frequency, 126 measurements have been made in 2 m sections in borehole Fj 2. This fracture frequency is less than 0.6 fracture/m at depths below 300 m. The measuring limit for these measurements is 1.3×10^{-10} m/s.

HYDRAULIC CONDUCTIVITY (m/s)
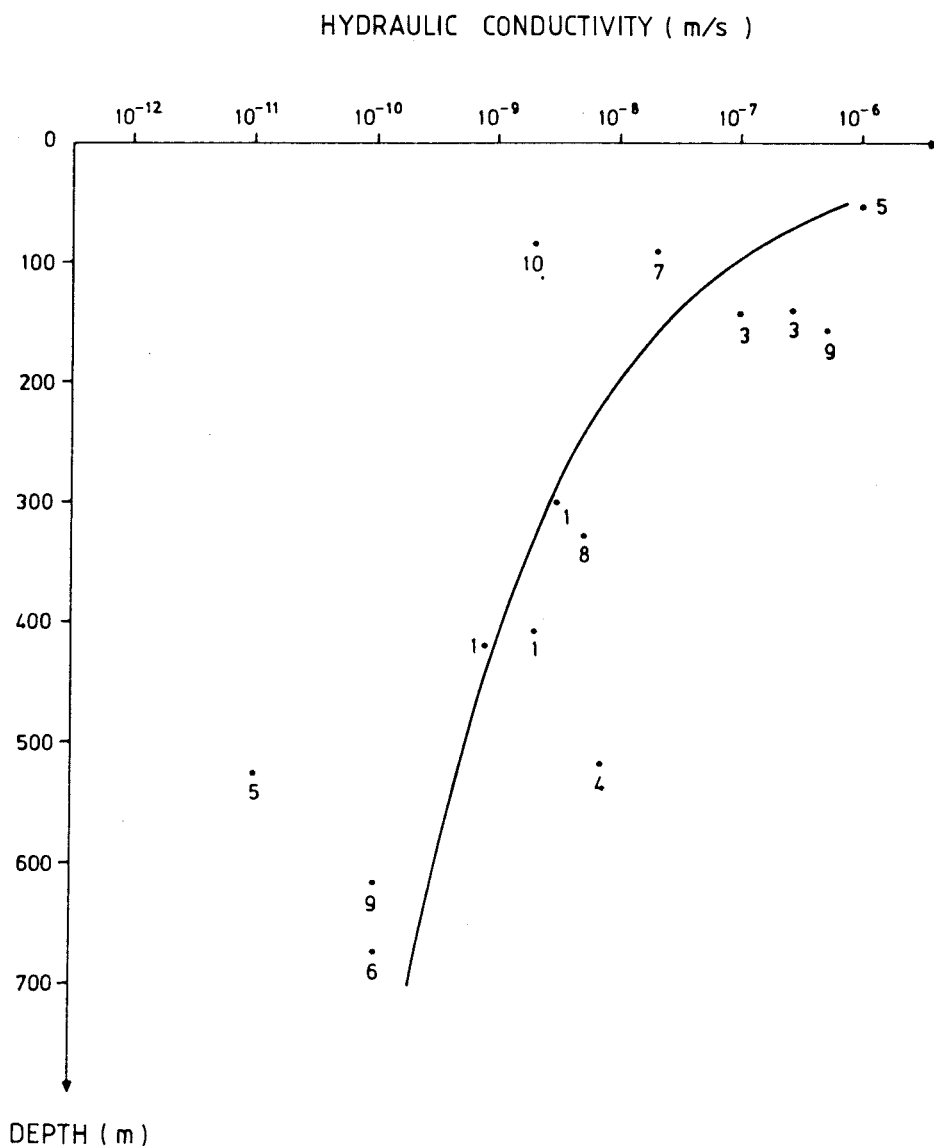
Figure 18-8. Correlation between hydraulic conductivity and depth for fracture zones in Fjällveden.

18.2.7 Groundwater chemistry

Chemical analyses have been carried out on water samples taken in boreholes Fj 2, Fj 4 and Fj 8 from altogether 10 different sections between 106 and 562 m, vertical depth.

The results of these analyses are reported in /18-6, 18-7, 18-8/. Table 18-4 summarizes the sampled sections and analysis results of significance for characterizing the site.

The analysis results show that the composition of the groundwater generally agrees with what was presented in chapter 7. Thus, for
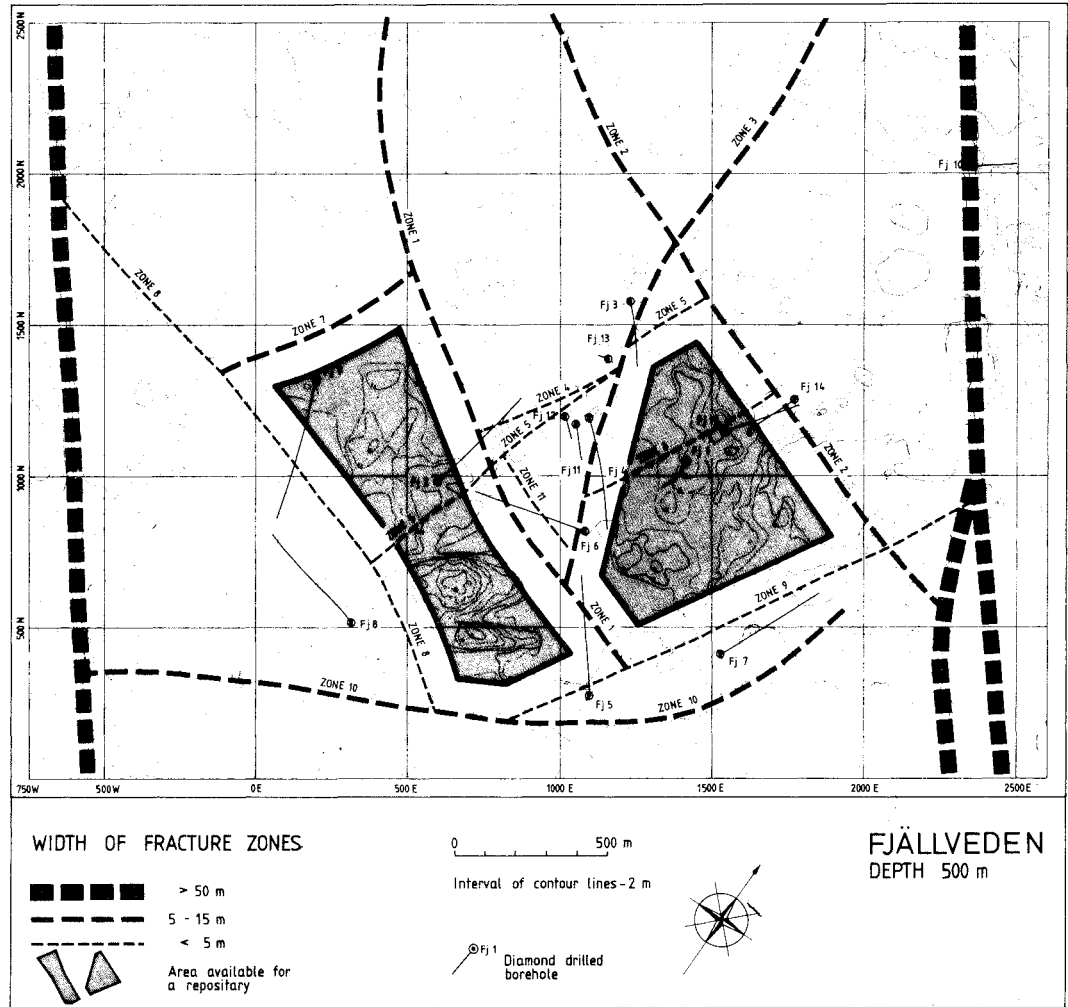
Figure 18-9. Location of the fracture zones at 500 m depth in Fjällveden. Available area for a respository at 450 and 550 m depth is indicated.

Table 18-4.   Results of chemical analyses on groundwater from Fjällveden.

Borehole	Depth m	pH	Eh V	HCO_3^- mg/l	Cl^- mg/l	HS^- mg/l	Na^+ mg/l	Ca^{2+} mg/l	Mg^{2+} mg/l	Fe^{2+} mg/l	TOC^a mg/l
Fj 2	106	8.1	-0.08	160	6	0.06	35	20	3	1.0	5
	293	7.3	-0.09	141	8	0.03	26	19	3	5.5	6
	409	7.4	-0.11	170	8	0.03	32	21	4	6.5	6
	506	8.8	-0.16	89	170	0.11	130	12	1	0.4	4
Fj 4	131	8.2	-0.09	220	8	0.05	50	20	3	1.1	7
	272	7.7	-0.11	192	9	0.03	38	28	4	6.5	7
	349	8.0	-0.10	194	5	0.20	60	17	2	1.8	7
	420	8.7	-0.17	200	10	0.13	62	14	2	1.3	7
Fj 8	402	8.5	-0.12	129	4	0.02	13	25	5	2.4	3
	562	9.0	-0.19	129	4	0.08	13	26	4	2.7	3

a Total organic carbon content of the water.
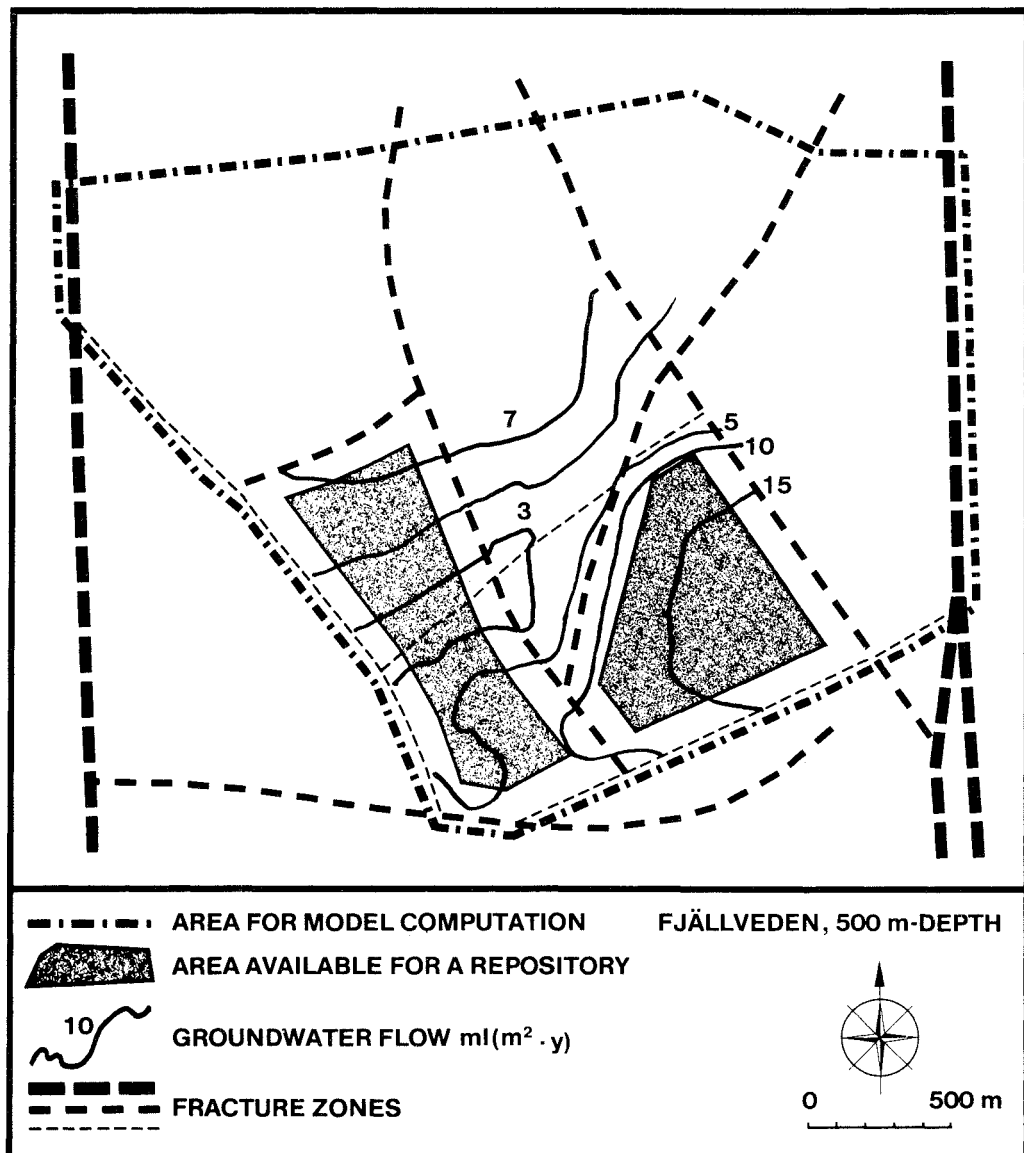
Figure 18-10. Groundwater flow at 500 m depth.

Fjällveden, the values that apply for solubility and sorption of radionuclides are those given in chapter 12 for reducing conditions.

18.2.8   Available space for a final repository

On the site at Fjällveden, the repository has been located on two storeys. Fig. 18-9 shows the area available for a two-storey repository at 450 and 550 m depth. The proposed areas have a total surface area of 1.85 km^2, which is about 90% more than the net requirements.

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                    ┌─────────────┐
                    │ UPPER SOIL  │
                    │ RESERVOIR   │
                    └─────────────┘
                          │
                    ┌─────────────┐
                    │ LOWER SOIL  │
                    │ RESERVOIR   │
                    └─────────────┘
                          │
                    ┌─────────────┐
                    │  GROUND-    │
                    │  WATER      │
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┌─────────────┐    ┌─────────────┐    ┌─────────────┐
│  MORPASJÖN  │───▶│ LIDSJÖN-    │───▶│ NYKÖPINGS-  │───▶
│             │    │ LÅNGHALSEN  │    │ FJÄRDEN     │
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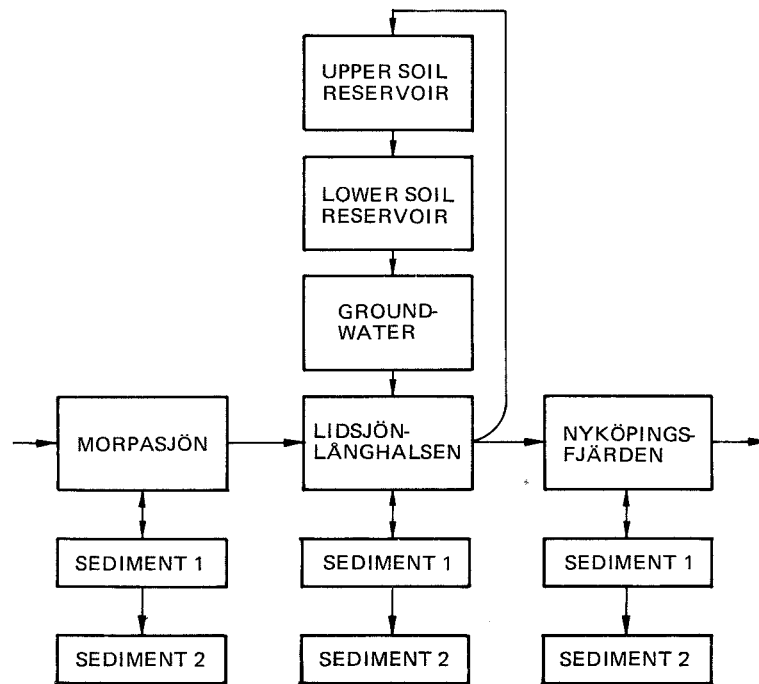
Figure 18-11. Flow chart for BIOPATH model's reservoir system for Fjällveden.

18.2.9   Model calculations

Model calculations of groundwater conditions within Fjällveden cover an area of 4.3 km^2 (Fig. 18-10). The hydraulic units within the computed area are noted in section 18.2.6. The calculation method and demarcation of the area follow the description given in section 6.3.2.

The occurrence of vertical layers of granite gneiss with higher hydraulic conductivity than the surrounding veined gneiss means that the rock mass has different water permeability in different directions, anisotropic properties. Two different model calculations have been carried out for Fjällveden to illustrate the effect of the anisotropy. In the first calculation, the rock mass has similar properties in all directions, isotopic conditions, with the relationship between depth and hydraulic conducitvity calculated from all the measured values without regard to rock type (Fig. 18-7).

In the second model computation, anisotropic properties in the rock mass have been assumed. At repository depth, a conductivity has been assumed that is 7 times greater parallel to the strike and dip of the rocks than perpendicular to that plane.

The results of the model calculations show that the groundwater flow at repository depth varies between 0.002 and 0.02 $1/(m^2 \cdot y)$ for the isotropic calculation, while and anisotropic calculation gives a flow varying between 0.01 and 0.05 $1/(m^2 \cdot y)$ - Fig. 18.10.

A certain check of the model calculations is obtained by comparing the calculated groundwater recharge to the bedrock with measured hydrometeorological data. Measured data indicate a total groundwater recharge of about 65-130 mm/y. Only a part of this comprises ground-water recharge to the bedrock. The remainder runs off through the soil cover to the nearest surface watercourse. Groundwater recharge to the bedrock according to model calculations is 45 mm/y. The size of the calculated groundwater recharge is considered to agree well with existing knowledge of groundwater recharge in crystalline bedrock.

Transport and uptake of radionuclides in the biosphere have been calculated on the basis of /18-91/. The model system for Fjällveden is shown in Fig. 18-11.

18.2.10   Concluding remarks

The study site at Fjällveden is characterized by a flat topography, local narrow fracture zones and a rock mass with low hydraulic conductivity. Alteration to clays is common in the bedrock and contributes to a sealing of the fracture zones. The wide occurrence of the clay mineral kaolinite indicates that Fjällveden has or has had conditions leading to deep weathering.

Within the investigated site the local fracture zones limit the chances for a single-storey repository. However, no horizontal fracture zones have been found which would restrict the possibility of a repository comprising several storeys.
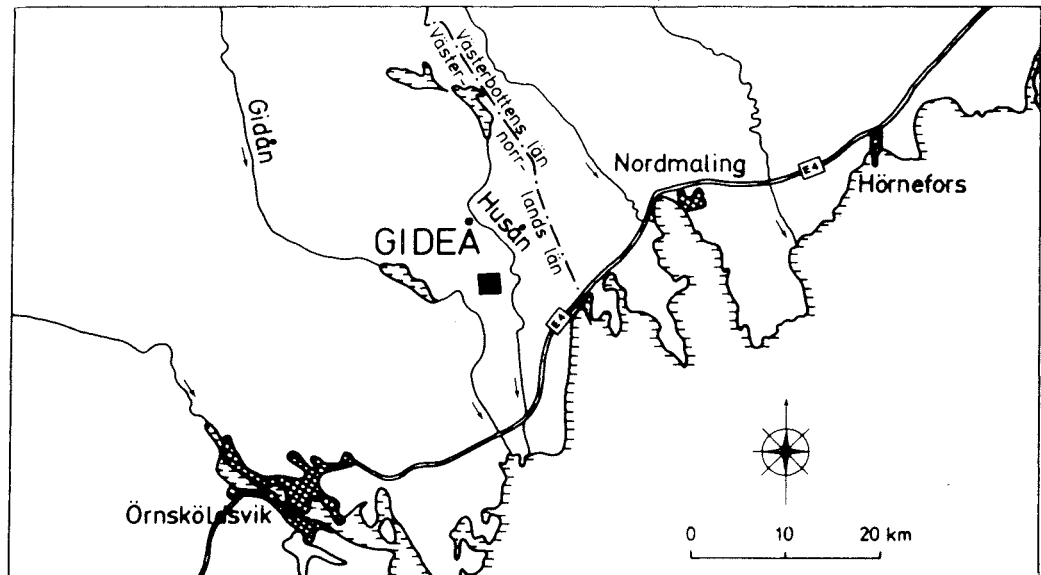
Figure 18-12. Locality map for the study site at Gideå.

The local fracture zones in the area have low hydraulic conductivity. At 500 m depth it is about 5×10^{-10} m/s (10 times higher than the rock mass). Layers of granite gneiss give the rock mass a directional hydraulic conductivity. These vertical layers have a hydraulic conductivity of 3×10^{-9} m/s at 500 m depth, and they comprise about 3% of the rock mass. The fracture frequency at repository depth is low, and model calculations show that groundwater flow within the site of an intended repository is low. Groundwater composition is such that it does not affect safety around a final repository.

The geological and hydrological prerequisites for construction of a final repository probable exist at Fjällveden. The randomly occurring, steeply dipping layers of granite gneiss with higher hydraulic conductivity than the surrounding, very impervious rock makes it difficult to judge how great a portion of the investigated site could be used for deposition of fuel. A final assessment of the suitability of the site can therefore not be made without supplementary investigations.
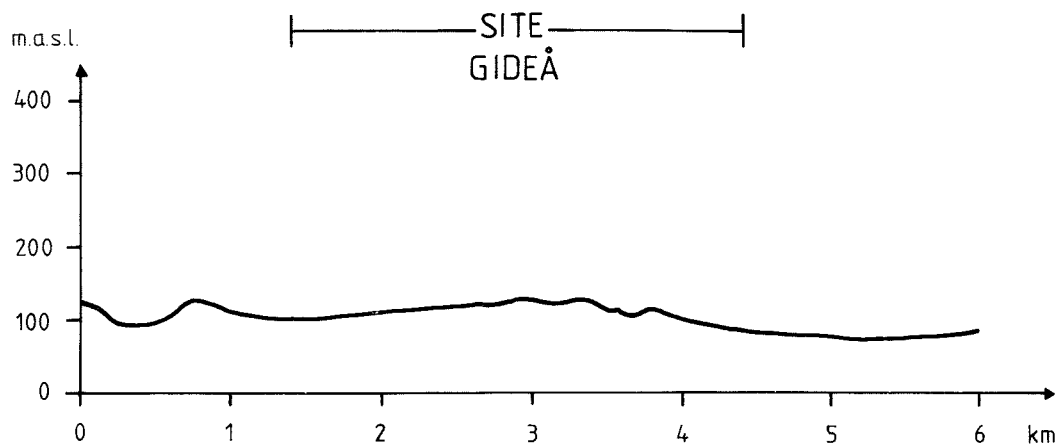
Figure 18-13. Topographical profile throughout the site at Gideå. Location shown in figure 18-14.

18.3   GIDEÅ

18.3.1   Location and topography

The study site at Gideå is situated in northern Ångermanland about 30 km NE of Örnsköldsvik (Fig. 18-12). The site lies within a more than 100 km^2 plateau about 100 metres above sea level and consists of a smaller elevated area with flat topography. The ground level within the site ranges between 80 and 130 metres above sea level. The topography of the site is shown in Fig. 18-13.

The site is forested, with small marshes. A large marsh district is located in the NE part, although this mostly lies outside the area investigated in detail. The Quaternary deposits are mainly moraine, in depressions usually covered with peat. About 15% of the site consists of rock outcrops.

A 2 by 3 km area (Fig. 18-14) was investigated in detail. The results of these investigations are presented in /18-10/.

LINEAMENTS IN THE AREA AROUND GIDEÅ, FROM AIR-PHOTO INTERPRETATION
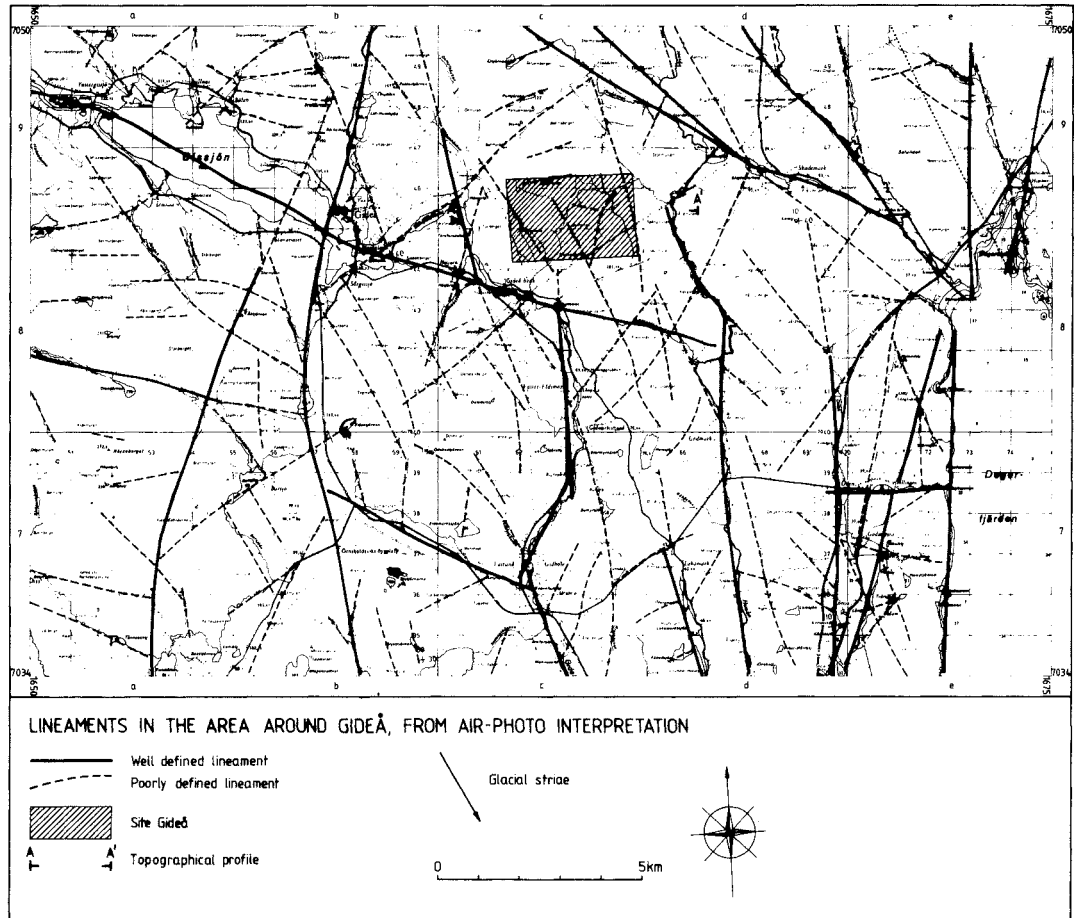
Figure 18-14. Interpreted lineaments in area around Gideå.

18.3.2 Bedrock geology

The main rock type within the site at Gideå is veined gneiss, i.e. the same type of rock as in Fjällveden (section 18.2.2). As in Fjällveden, this rock type is characterized by veins and other irregular bodies of varying mineral composition. These veins generally have a NE strike, and usually dip shallowly, 10-30°, to the north.

The essential minerals in the veined gneiss are quartz 56%, biotite 19%, plagioclase 13% and microcline 6%. Sulphide minerals usually pyrrhotite are present to a relatively small extent in the veined gneiss in the form of small mineral concentrations or fracture fillings. The amounts of ore minerals are so low that mining is not feasible.

As in Fjällveden there are some areas of granite gneiss. Here as well, this rock type was affected by the metamorphism and deformation that shaped the veined gneiss. The granite gneiss occurs as thin horizontal layers parallel to the genissic structures, and makes up 6% of the total core length.

Pegmatities also occur within the site in the form of small bodies and metre-wide dykes. Diabase occurs in the form of narrow (1-10 m), steep dykes with an E-W strike. The diabases comprise 1.9% of the total core length.

18.3.3   Fracture zones

In Gideå, 13 core (diamond) boreholes and 24 percussion boreholes have been drilled. The dip and length of the core boreholes are given in Table 18-5. The locations of these boreholes are shown in Figs. 18-15 and 18-19.

Large regional zones occur outside the site investigated in detail. The zones coincide largely with the valleys of the Gideån and Husån rivers, located about 5 km apart (Fig. 18-14).

The local fracture zones in the central part of Gideå have been penetrated with core boreholes in 16 different places. No horizontal fracture zones have been found, all the zones encountered have a marked dip. The width of the fracture zones range from 1 to 24 m, with an average of 11 m. A resumé of the fracture zones is presented in Table 18-6. Fig. 18-15 shows the location of the local fracture zones within the Gideå site.

The local fracture zones in Gideå bound a triangular rock block with a surface area of 1.8 km^2. This block is intersected by only two local fracture zones, about 4 m wide. The distance between these two zones and the somewhat wider fracture zones which bound the block varies between 400-800 m. Diabase dykes traverse the northern part of the block. In the local fracture zones, some rocks are weathered and altered to clays. Commonly occurring fracture minerals are calcite, chlorite, iron pyrites and laumontite.

Table 18-5.   Summary of length and dip of core (diamond) drill
boreholes on site at Gideå.

Borehole	Dip (degrees)	Borehole length (m)	Borehole depth (m)
Gi 1	90	704.3	701
Gi 2	60	705.5	617
Gi 3	60	703.0	626
Gi 4	70	6907	657
Gi 5	60	702.0	605
Gi 6	60	704.0	648
Gi 7	60	700.5	635
Gi 8	62	701.6	619
Gi 9	67	281.9	250
Gi 10	65	702.6	632
Gi 11	65	701.5	632
Gi 12	61	249.8	218
Gi 13	61	704.5	616

18.3.4   Fracturing of the rock mass

Fracturing in the rock mass was mapped partly from outcrops and
partly from drill cores. The fractures have two main strikes, N and
NE. Fracture frequency in outcrops was measured at 1.2 fractures/m.
The variation of fracture frequency with depth in the rock mass
between the fracture zones is shown in Fig. 18-16. The fracture
frequency is 4.5 fractures/m down to 400 m. Below 500 m depth, the
fracture frequency is 2.0 fractures/m. The reasons for the higher
fracture frequency in drill cores compared to outcrop mapping are
the same as those discussed in section 18.2.4. As in Fjällveden,
fracture frequency is similar in veined gneiss and genissic granite.
Fracture frequency is markely higher in the diabses, 20 fractures/m.
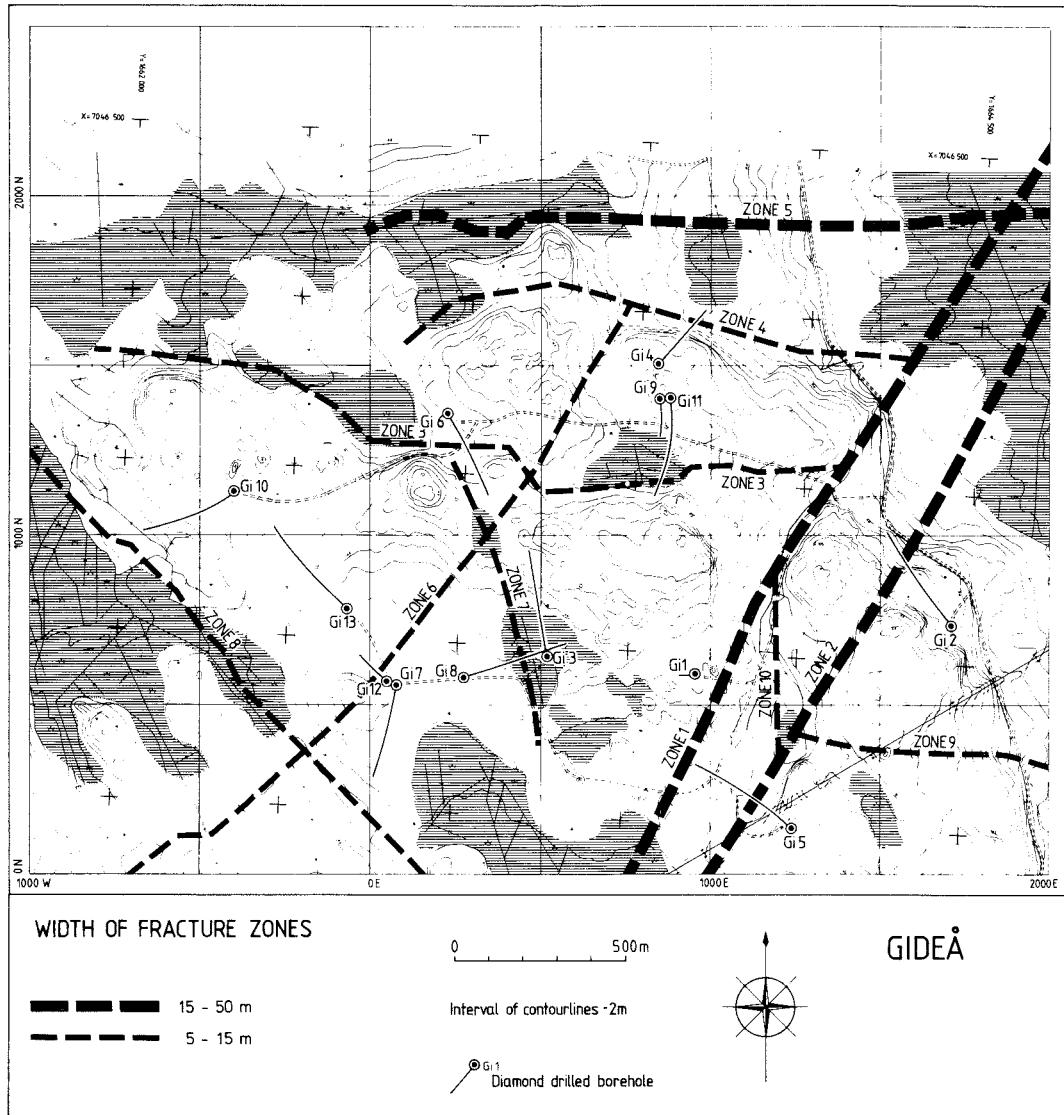
Figure 18-15. Fracture zones at the surface within the study site at Gideå.

18.3.5 Hydrology, meterology and recipient conditions

The northern part of the study site at Gideå is drained to the NE be small streams into the Husån River. The remainder of the area is drained to the west by small streams into the Gideälven River. Husån and Gideälven discharge after 19 and 16 km, respectively, into the Gulf of Bothnia. There are no lakes on the site. The location of the rivers is shown in Fig. 18-12. The natural concentrations of uranium and radium in the pond at Gideå are given in Table 18-7.

As in Fjällveden, the topography of the site is such that for the most part it constitutes an inflow area for groundwater. Low-lying

Table 18-6.   Resumé of fracture zones in Gideå.

Fracture zone	Position in borehole (m)	Dip (degrees)	True width (m)	K value m/s
1	Gi 2 (309-335)	40 SO	24	7×10^{-10}
	Gi 5 (210-232)	40 SO	22	3×10^{-6}
2	Gi 5 (520-567)	70 NV	11	$<5 \times 10^{-12}$
3A	Gi 4 (217-259)	30 N	20	2×10^{-7}
	Gi 6 (51- 80)	30 N	24	2×10^{-7}
	Gi 9 (129-146)	30 N	15	not measured
	Gi 11 (119-130)	30 N	10	1×10^{-9}
3B	Gi 6 (222-240)	80 N	9	7×10^{-12}
	Gi 11 (345-352)	80 N	4	7×10^{-11}
4	Gi 4 (606-655)	90	10	2×10^{-10}
5	-	80 N*	50*	-
6	Gi 3 (622-629)	70 SO	4	5×10^{-9}
	Gi 4 (670-690)	70 SO	1	$<5 \times 10^{-12}$
	Gi 7 (362-397)	70 SO	3	2×10^{-11}
	Gi 12 (52- 61)	70 SO	8	not measured
7	Gi 3 (329-342)	75 O	7	7×10^{-11}
	Gi 6 (443-452)	75 O	1	1×10^{-10}
8	-	70 SV*	10	-
9	-	70 N*	5*	-
10	-	90*	5*	-

* Calculated from geophysical observations.

areas in the site comprise local outflow areas for shallow ground-water (18-10). The pond at Gideå mill is the first major recipient for surface water and groundwater from the site. The theoretical water turnover time in the pond is about 1 day. At the rivers' outlets into the Gulf of Bothnia the turnover time is about 9 days.

Mean precipitation within the study site at Gideå is 765 mm/y. Of that total 33% falls as snow. Evaporation is estimated at 420 mm/y /18-3/, 18-4/. Runoff is estimated at 345 mm/y (11 l/s · km^2): Groundwater recharge on the site is at most equal to runoff. Only a
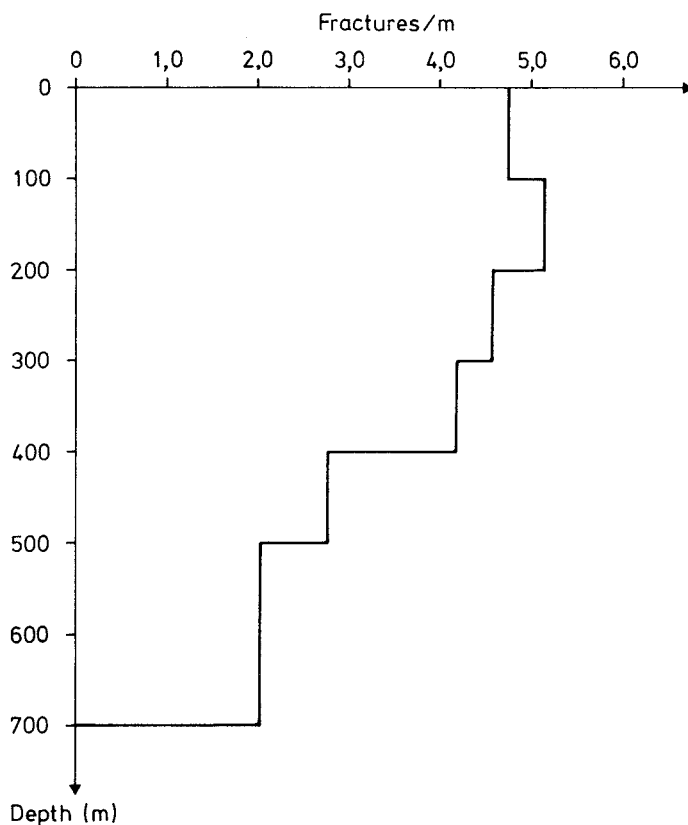
Figure 18-16. Fracture frequence in rock mass within the study site at Gideå.

part of the recharge consists of groundwater recharge to the bed-
rock. The remainder runs off through the soil cover to the nearest
surface watercourse.

Table 18-7.   Measured concentrations of uranium and radium at
                Gideå pond.

	Uranium	Radium	Unit
Sediment	80	60	$Bq\ kg^{-1}$
Lake water	<0.006	<0.002	$Bq\ l^{-1}$
Soil	75	50	$Bq\ kg^{-1}$

18.3.6   Hydraulic conductivity of the bedrock

The bedrock in Gideå has been divided into different hydraulic
units (chapter 6) as follows:

HYDRAULIC CONDUCTIVITY (m/s)
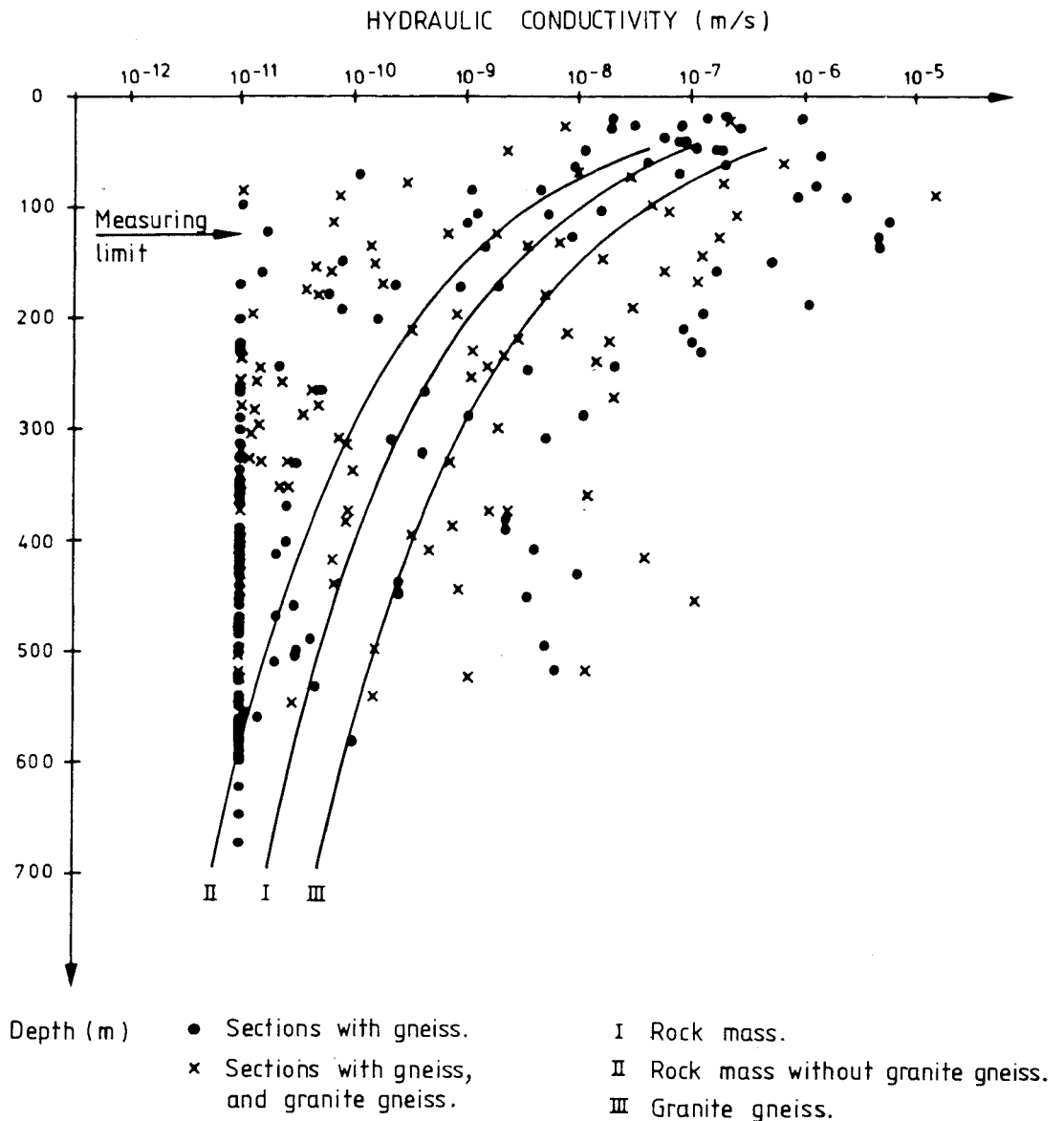
Figure 18-17. Correlation between hydraulic conductivity and depth within the rock mass.

- local fracture zones (10)
- the rock mass

Hydraulic conductivity has been measured in 288 sections 25 m long and 86 sections 5 or 10 m long in boreholes Gi 1-Gi 8, Gi 10, Gi 11 and Gi 13. All measured values are presented in Fig. 18-17 for the rock mass and Fig. 18-18 for the local fracture zones. The locations of the local fracture zones are shown in Fig. 18-15.
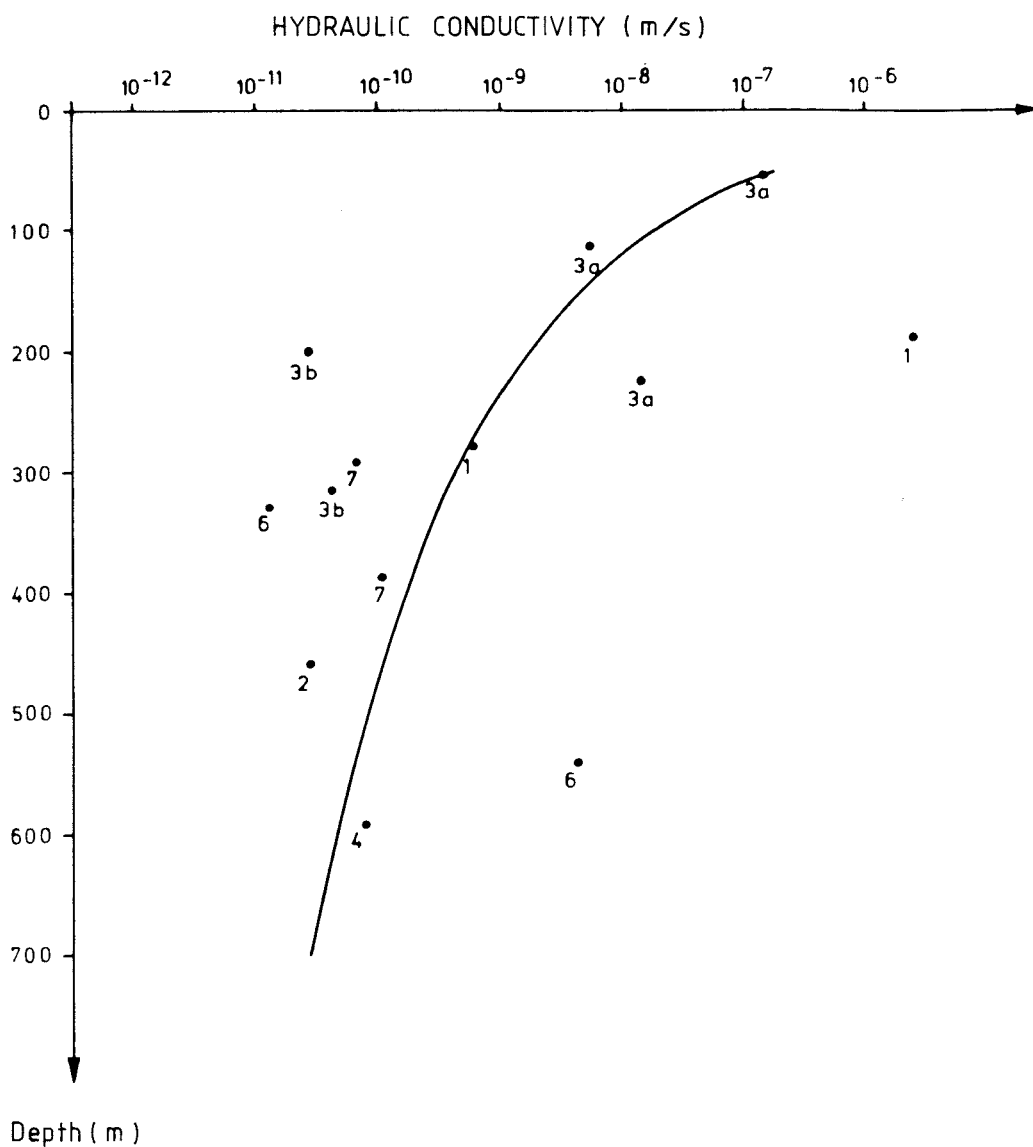
Figure 18-18. Correlation between hydraulic conductivity and depth for fracture zones in Gideå.

As in Fjällveden, the granite gneisses in the rock mass have higher hydraulic conductivity than the veined gneiss. The diabases and veined gneisses have similar hydraulic conductivity. Figs. 18-17 and 18-18 show the relationship between depth and hydraulic conductivity. The relationships apply down to 700 m.

To determine the hydraulic fracture frequency, 162 measurements have been made in 2 m sections in borehole Gi 7. This fracture frequency varies between 0.4 and 1.5 fractures/m at depths below 300m. The measuring limit for these measurement is 1×10^{-11} m/s.

18.3.7   <u>Groundwater chemistry</u>

Chemical analyses have been carried out on water samples taken in the two boreholes Gi 2 and Gi 4 from altogether 10 different sections between 91 and 596 m depth.

The results of these anlyses are presented in /18-7, 18-8, 18-11/. Table 18-8 summarizes the sampled sections and analyses results of significance for characterizing the site.

Table 18-8.   Results of chemical analyses on groundwater from Gideå.

Borehole	Depth m	pH	Eh v	HCO_3^- mg/l	Cl^- mg/l	HS^- mg/l	Na^+ mg/l	Ca^{2+} mg/l	Mg^{2+} mg/l	Fe^{2+} mg/l	TOC^a mg/l
Gi 2	157	8.8	-0.10	161	4	0.02	50	10	3	0.1	4
	288	8.8	-0.09	163	5	0.03	49	10	3	0.6	5
	353	8.7	-0.09	162	5	0.04	52	10	2	0.5	3
	478	8.9	-0.09	160	5	0.02	50	10	2	0.2	3
	528	8.7	-0.10	159	5	0.03	50	10	2	-	2
Gi 4	91	8.0	-0.03	140	2	0.03	11	32	4	0.8	3
	212	9.0	-0.12	133	8	0.10	48	10	1	0.3	5
	385	9.3	-0.16	20	180	0.20	103	21	1	0.1	2
	498	8.5	-0.12	120	2	0.01	5	30	4	7.4	3
	596	8.9	-0.32	70	300	2.70	150	60	1	-	-

a Total organic carbon content of the water

The deepest section in Gi 4 produced very little water (1.6 l/day against normally 100-300 l/day), which explains why certain tests could not be performed, e.g. Fe^{2+} and TOC. The sulphide content is unusually high here. A lower sulphide content was used to calculated the extent of canister corrosion (see chapter 10). However, the value from Gi 4 still lies within the bounds of reasonable variation and should not alter the estimates of canister life.

Otherwise, the analysis results show that the composition of the groundwater generally agrees with what was presented in chapter 7. Accordingly, for Gideå, the values applicable for solubility and sorption of radionuclides are those presented in chapter 12 for reducing conditions.
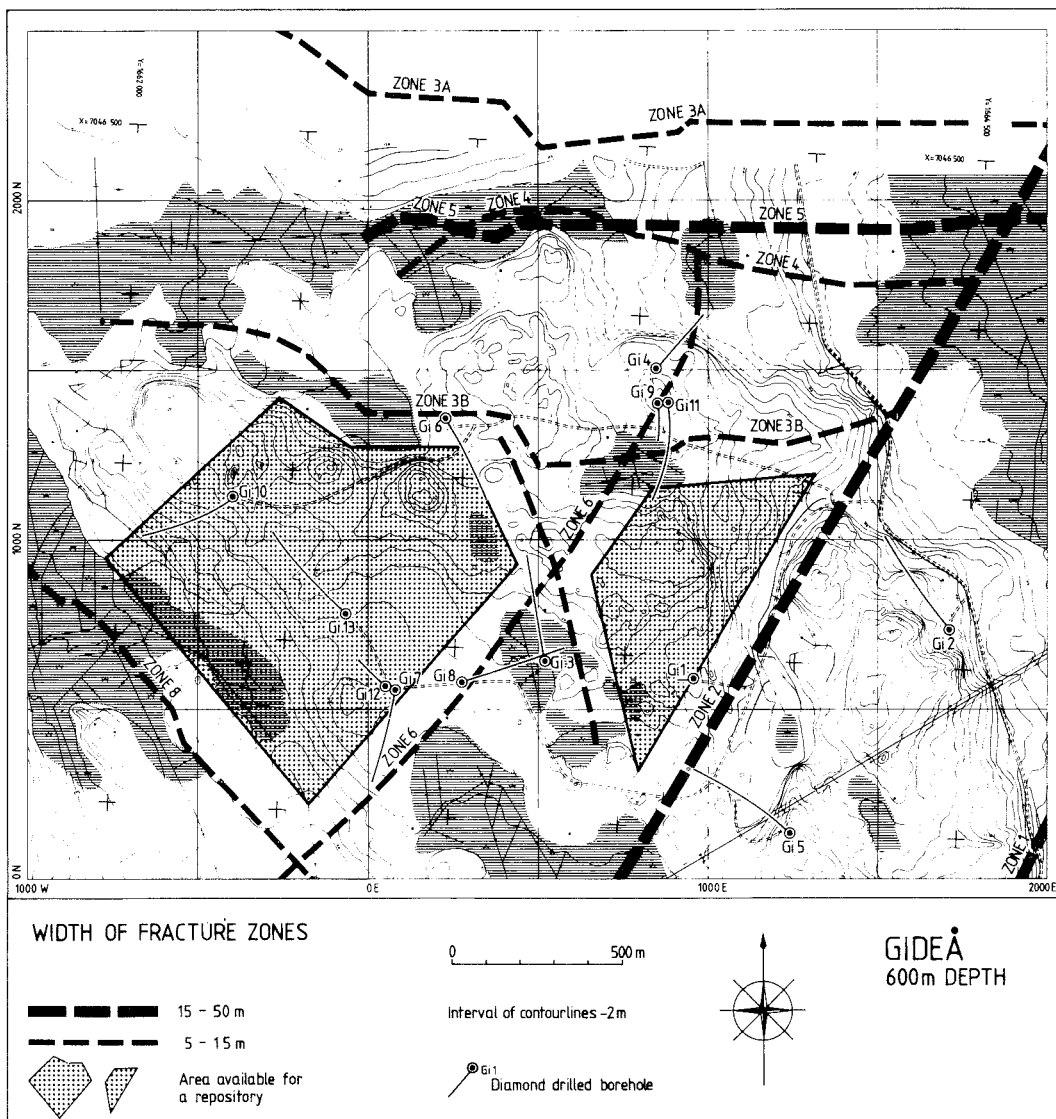
Figure 18-19. Location of the fracture zones at 600 m depth in Gideå. Available area for a repository at 500 and 600 m depth is indicated.

18.3.8   Available space for a final repository

The repository has been located on two storeys in the investigated parts of Gideå. Fig. 18-19 shows the area available for a repsitory between 500 and 600 m depth. The proposed area has a total surface area of 2.0 km^2, which is about 120% greater than the net requirement.
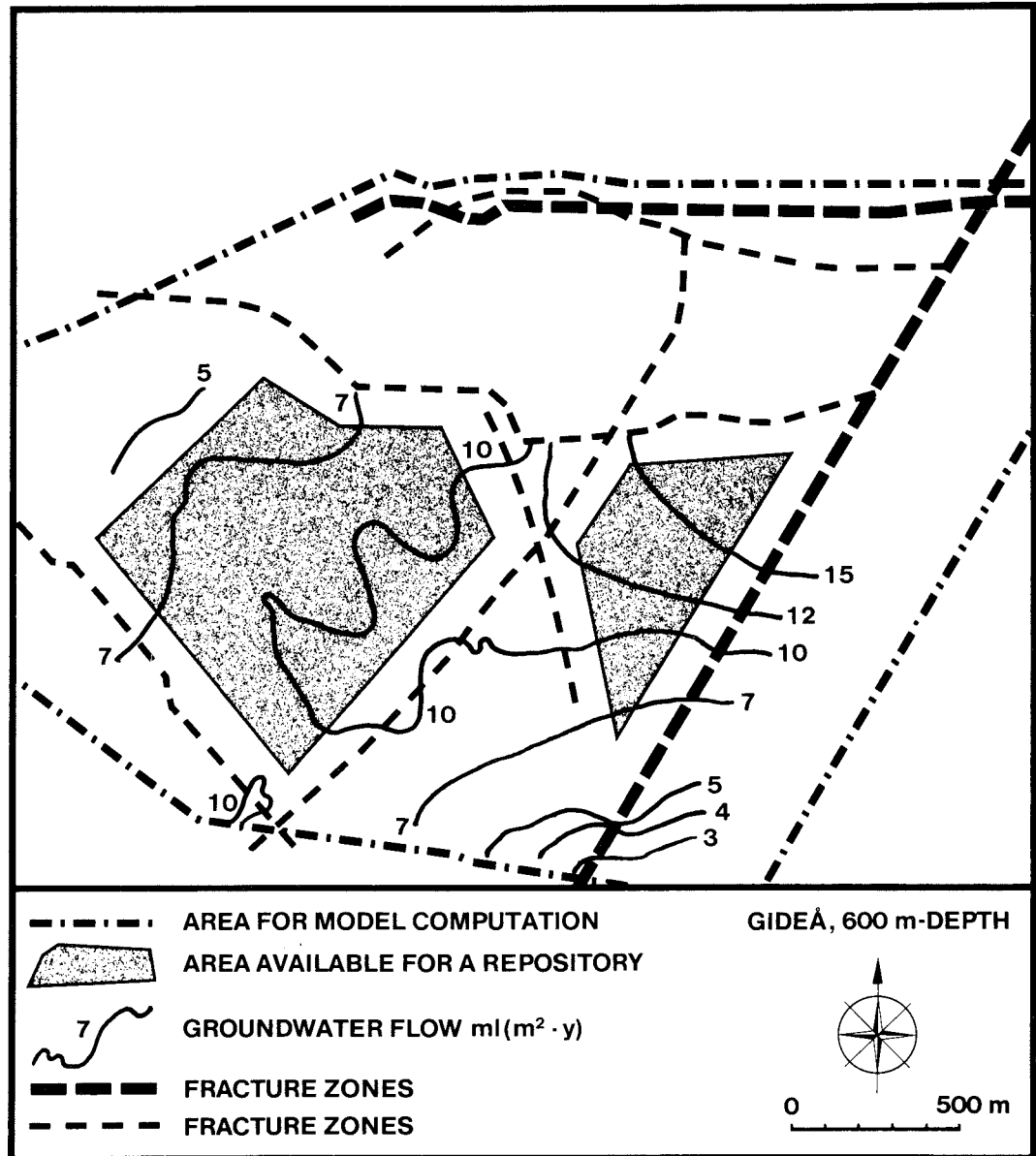
Figure 18-20. Groundwater flow at 600 m depth.

18.3.9   Model calculations

Model calculations of groundwater conditions within Gideå cover an area of 5.6 km^2 (Fig. 18-20). The hydraulic units within the computed site are given in section 18.3.6.

The presence of almost horizontal layers of granite gneiss having higher hydraulic conductivity than the veined gneisses means that the rock mass has anisotropic hydraulic properties. The model calculations have therefore followed the same pattern as for Fjäll-
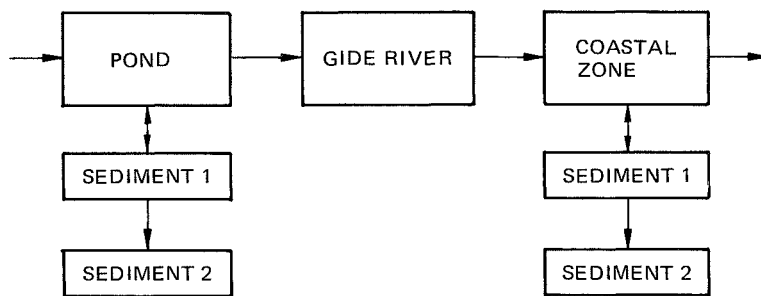
Figure 18-21. Flow chart for BIOPATH model's reservoir system for Gideå.

veden. The difference is that the granite gneisses in Gideå are in horizontal layers, compared with vertical in Fjällveden. In the anisotropic calculations, a hydraulic conductivity 1.5 times greater horizontally than vertically has been assumed at repository depth.

The results of the model computations show that the groundwater flow at repository depth varies between 0.004 and 0.02 $1/(m^2 \cdot y)$ for the isotropic calculation (Fig. 18-20). The anisotropic calculation gives a flow varying between 0.01 and 0.02 $1/(m^2 \cdot y)$. On the average, however, the flow is about half as much in the anisotropic estimate.

According to the model computations, groundwater recharge to the bedrock in Gideå is about 110 mm/y. According to the measured meteorological data, the total groundwater recharge in Gideå is 345 mm/y. The size of the calculated groundwater recharge is considered to agree well with existing knowledge of groundwater recharge in crystalline bedrock.

Transport and uptake paths of radionuclides in the biosphere have been calculated /18-9/. The model system for Gideå is shown in Fig. 18-21.

18.3.10   Concluding remarks

The study site at Gideå is characterized by a flat topography, local fracture zones and a rock mass with low hydraulic conducti-

vity. Alteration to clays is common in the bedrock and contributes to a sealing of the fracture zones.

The local fracture zones on the site have low hydraulic conductivity: At 500 m depth, it is about 9×10^{-11} m/s (2 times greater than for the rock mass). Layers of granite gneiss give the rock mass a directional hydraulic conductivity. These layers have a hydraulic conductivity of about 1×10^{-10} m/s (9 times higher than the veined gneiss) at 500 m depth, and they form about 6% of the rock mass. Thus, they have the same hydraulic properties as the local fracture zones.

The model calculations show that the groundwater flow is small at 500 m depth within the site of a conceivable final repository. The low hydraulic conductivity in the layers of granite gneiss means that water transport in these layers is not essentially higher than in the veined gneisses. The horizontal orientation of the layers of granite gneiss means that they do not form direct transport paths for groundwater between the repository and biosphere. Due to the orientation of the layers the transport paths are instead longer than if the rock mass were homogeneous.

At 600 m depth, the distance between the fracture zones is 500-900 m, and fracture frequency below 400 m depth is low. Due to their higher fracture frequency, the diabases should be avoided for positioning of storage holes. As in Fjällveden, the groundwater composition is such that it does not affect the safety of an intended final repository. The geological and hydrological prerequisites for a repository are consequently favourable in Gideå, and the available volume is very large.

18.4   KAMLUNGE

18.4.1   Location and topography

The study site at Kamlunge is in southern Norrbotten, 65 km NE of Luleå (Fig. 18-22). This part of Norrbotten is well-forested and hilly, with valleys oriented mainly NNW.
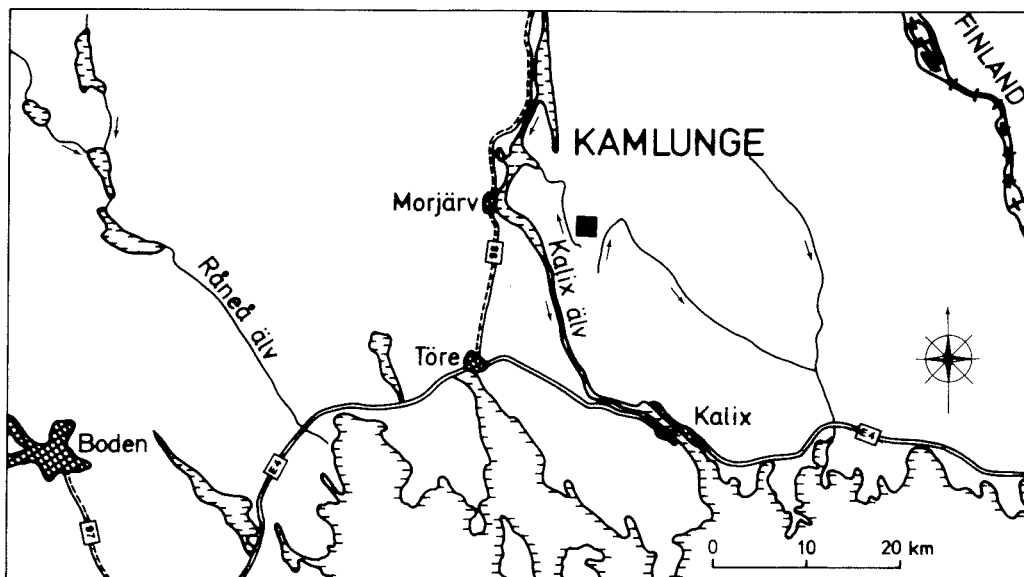
Figure 18-22. Locality map for the study site at Kamlunge.

The investigated site is located on a 16 km^2 rock plateau, Kamlunge-kölen, 5 km E of the Kalixälven River. The relief between the plateau and the adjacent valleys is about 100 m. The topography of the site is shown in Fig. 18-23. Up on Kamlungekölen's central and western parts, the soil cover is thin, and about 30% of this part consists of rock outcrops which form large continuous rock surfaces. The eastern part of Kamlungekölen comprises mainly marshy ground with a few scattered rock outcrops. The soil thickness here is 5-20 m.

The dominant Quaternary deposit is moraine with low boulder content. There are areas with "shingle pavement". The vegetation on Kamlungekölen consists of forests.

A 2.5 x 3 km area was investigated in detail (Fig. 18-24). The results of those studies are presented in /18-12/.

18.4.2   Bedrock geology

Gneisses and red granite dominate the bedrock in Kamlungekölen (Table 18-9). It also contains amphibolite and a rock with grano-dioritic to dioritic composition. The gneisses are the oldest
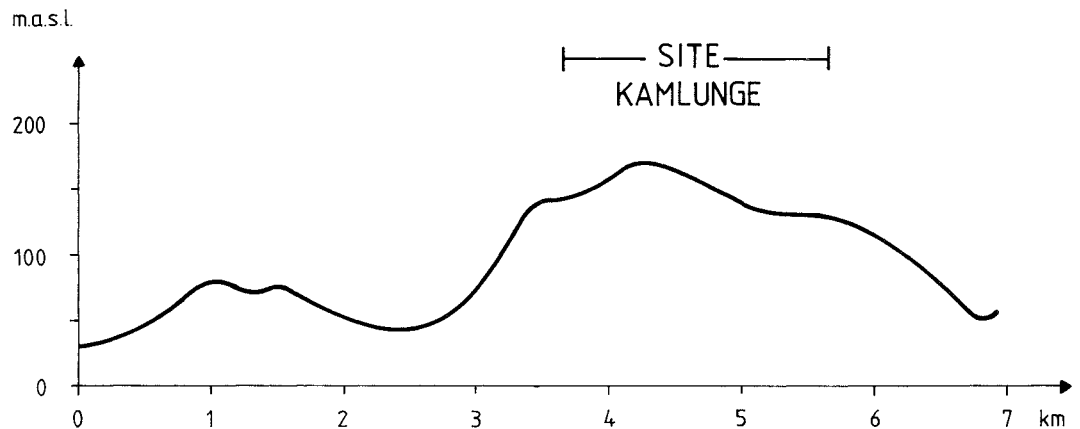
Figure 18-23. Topographical profile through the site at Kamlunge.

rocks, about 1 900-2 500 Ma. /18-13/, and are formed by metamorphism at high pressure and temperature of marine sands and muds. They occur as two rock types, biotite gneiss and quartzitic gneiss. These rock types form large isolated bodies, commonly surrounded by granite and granodiorite to diorite. The red granite, Lina granite, is 1 560 million years old /18-14/ and dominates in the southern parts of the site. There are pegmatites within and associated with the granite. The essential mineral components of the various rock types are given in Table 18-9.

The contacts between different rock types vary from gradational to sharp. There is usually no increase in fracturing or high hydraulic conductivity at the contacts. No workable ore or mineral deposits have been found.

18.4.3   Fracture zones

In Kamlunge, the bedrock has been investigated with 16 core boreholes and 21 percussion boreholes. The dip and length of the core boreholes are given in Fig. 18-25.
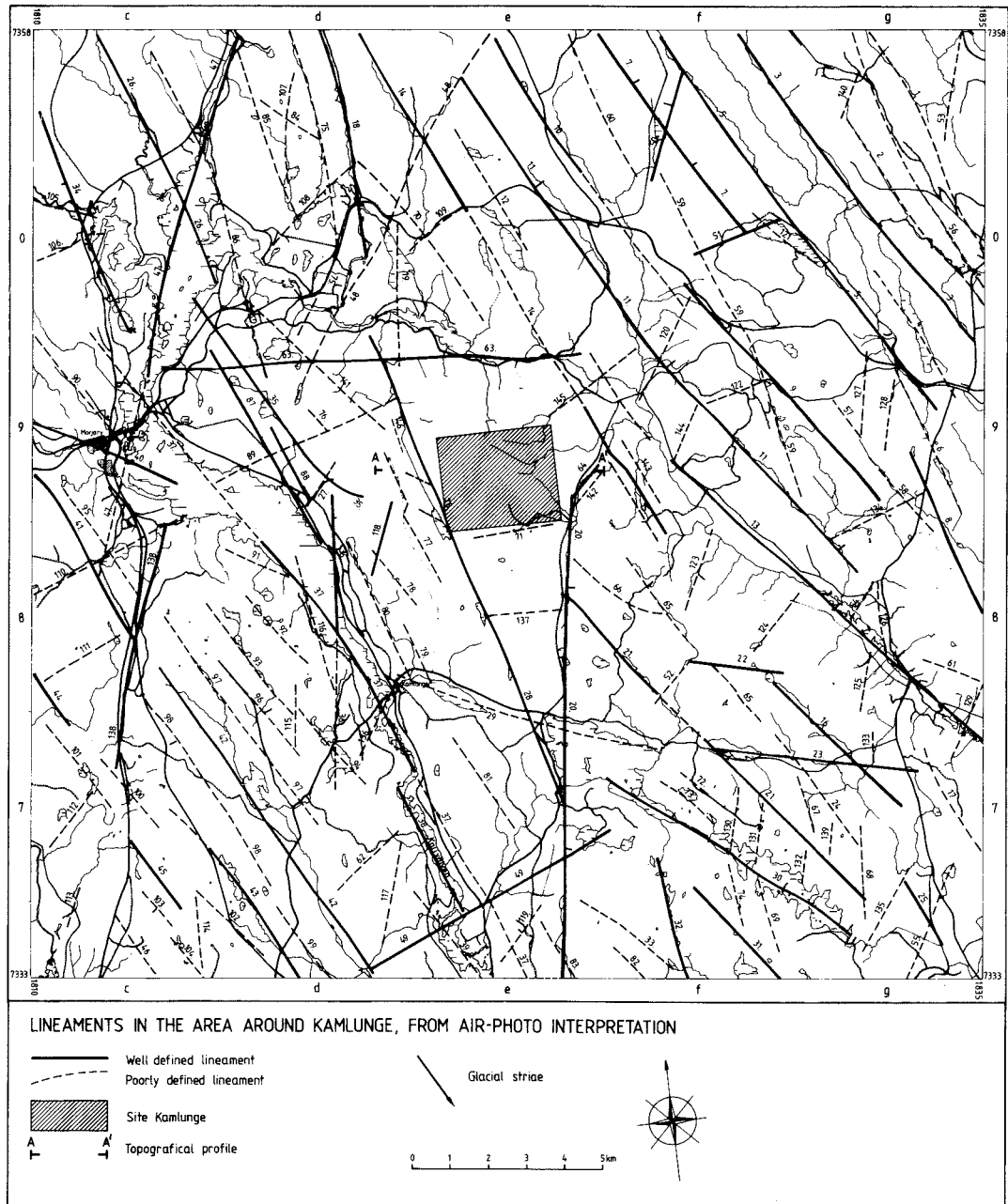
Figure 18-24. Interpreted lineaments in area around Kamlunge.

Kamlungekölen is bounded by regional fracture zones on the north, east and west (Fig. 18-24). Kamlungekölen can therefore be regarded as a 16 km^2 triangular block. In the western regional fracture zone, a soil depth of 72 m has been measured in one percussion borehole. The underlying bedrock here is fractured and forms a marked aquifer. The relief between the rock surface in the western valley and the highest point on Kamlungekölen is thus about 200 m.

Table 18-9.   Mineral composition and occurrence in drill cores
of different rock types from Kamlungekölen.

Rock type	Percentage of core length (%)	Essential minerals
Red granite and pegmatite	40	Quartz, potash feldspar, plagioclase, biotite
Granodiorite to diorite	13	Biotite, plagioclase, amphibole
Amphibolite	7	Amphibole, plagioclase
Biotite gneiss	26	Biotite, quartz, plagioclase
Quartzitic gneiss	14	Quartz, potash feldspar, plagioclase, biotite

Table 18-10.   Summary of length and dip of core (diamond) drill
boreholes on site at Kamlunge.

Borehole	Dip	Borehole length (m)	Borehole depth (m)
Km 1	85	675.0	670
Km 2	60	701.3	566
Km 3	60	700.0	583
Km 4	60	700.1	577
Km 5	60	251.4	210
Km 6	60	104.5	89
Km 7	60	249.0	208
Km 8	60	251.3	208
Km 9	60	449.3	366
Km 10	60	287.0	205
Km 11	60	700.4	544
Km 12	60	801.9	636
Km 13	60	703.1	582
Km 14	60	700.2	579
Km 15	60	251.2	210
Km 16	60	252.6	211
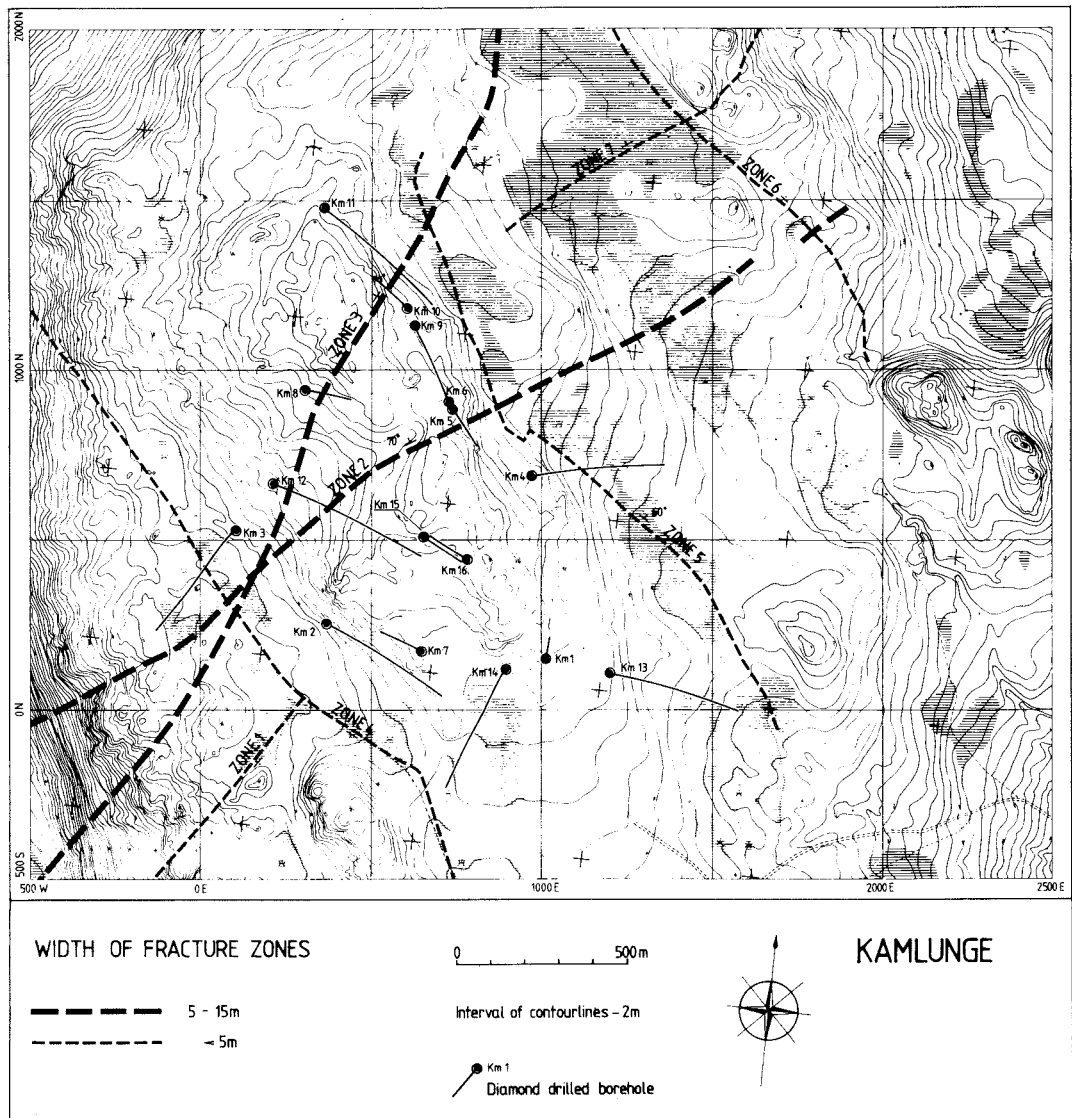
Figure 18-25. Fracture zones at the surface within the study site at Kamlunge.

The local fracture zones in Kamlungekölen (Fig. 18-25) are narrow, steeply dipping and spaced 500-1 500 m apart. The fracture zones have been investigated with core boreholes in 14 different places. A resumé of the fracture zones is presented in Table 18-11. The width of the fracture zones varies from 3 to 14 m, with an average width of 6 m. The fracture zones contain sections with weathered, brecciated and crushed rocks. Core losses are common during drilling through the zones. Commonly occurring fracture minerals are chlorite, calcite, laumontite, smectite and various iron oxides. A horizontal fracture zone was encountered at 555 m depth below Kamlungekölen (Fig. 18-26). The zone forms an aquifer, 4-14 m wide.
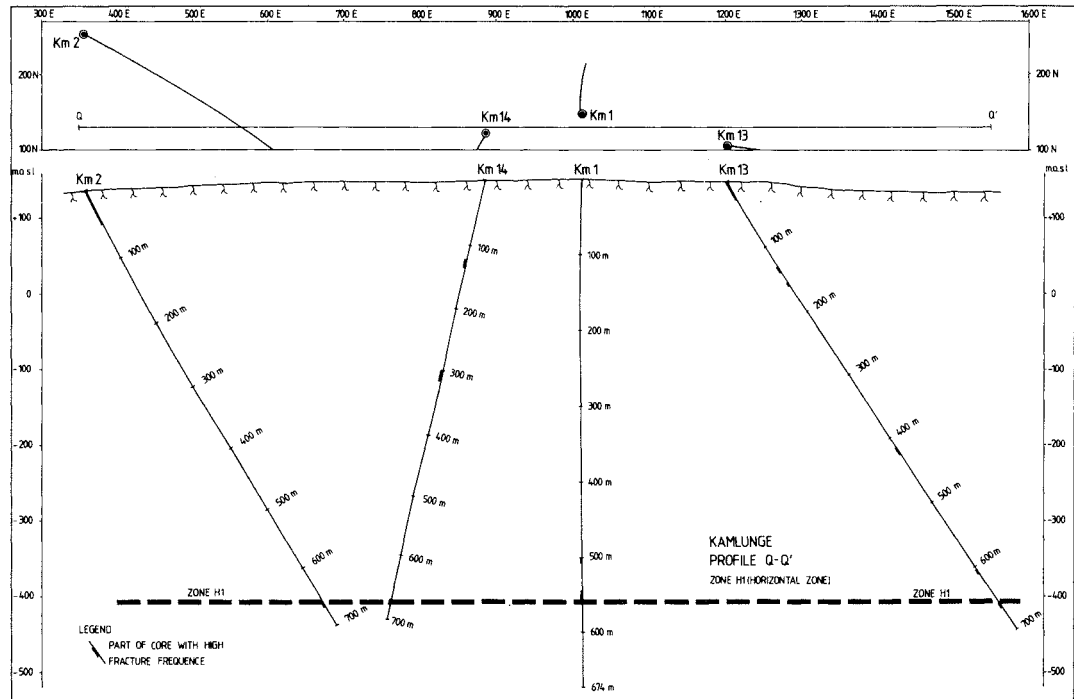
Figure 18-26. Vertical section through Kamlungekölen showing the location of the horizontal fracture zone.

This fracture zone is less crushed and weathered than the steeply dipping fracture zones. It has been penetrated by 4 core boreholes. In two of these, Km 1 and Km 13, the zone is fractured and reddened through alteration and precipitation of the iron mineral haematite. Chlorite, formed after haematite, is also present. The other two boreholes, Km 2 and Km 14, show only an increased fracture frequency at this level. The lateral extent of this zone has not been established, but it is not found in Km 12, for example.

18.4.4   Fracturing of the rock mass

The dominant fracture direction in the outcrops is WNW. In the gneisses, the fracture frequency in outcrops is 1.3 fractures/m, while the diorites, granites and granodiorites have 1.1 fractures/m. In the cores as well, the various rock types show little difference in fracturing. The total fracture frequency in the rock mass between the fracture zones decreases with depth (Fig. 18-27). The reasons for the higher fracture frequency in cores compared to outcrop measurements are the same as those given in section 18.2.4. The

fracture frequency below 400 m is about 2.5 fractures/m. The high frequency of fractures in the upper intervals is due to horizontal jointing.

18.4.5   Hydrology, meterology and recipient conditions

Kamlungekölen lies on the watershed between the catchment area of the Kalixälven River to the west and the Sangisälven River to the east. These watercourses drain into the Gulf of Bothnia after 30 and 35 km, respectively. The major part of the investigated area drains via the Korpikån River into the Sangisälven River. The location of the rivers is shown in Fig. 18.22.

Table 18-11.   Resumé of fracture zones in Kamlunge.

Fracture zone	Position in borehole (m)	Dip (degrees)	True width (m)	K value m/s
1	-	90*	3*	-
2	Km 3 (313-337)	70 NV	4	7×10^{-11}
	Km 12 (195-210)	70 NV	12	2×10^{-7}
	Km 9 (414-425)	70 NV	9	2×10^{-9}
	Km 5 (47- 53)	70 NV	5	not measured
	Km 6 (96- 94)	70 NV		not measured
3	Km 12 (52- 60)	70 NV	7	4×10^{-9}
	Km 8 (63- 69)	70 NV	4	not measured
	Km 11 (324-335)	70 NV	10	not measured
	Km 3 (441-450)	70 NV	1	3×10^{-8}
4	Km 3 (504-517)	80 SV	4	4×10^{-11}
5	-	60 NO*	4*	-
6	-	85 NO*	3*	-
7	-	75 NV*	3*	-
H1	Km 1 (544-560)	horizontal	14	not measured
	Km 2 (676-684)	horizontal	7	4×10^{-9}
	Km 13 (669-674)	horizontal	4	1×10^{-8}
	Km 14 (667-673)	horizontal	5	6×10^{-9}

* Calculated from geophysical observations.
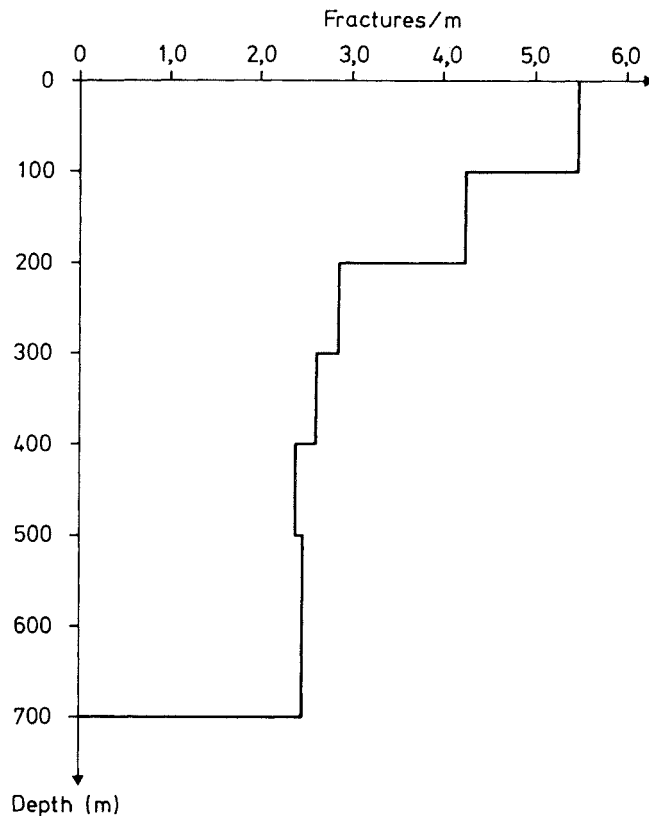
Figure 18-27. Fracture frequency in the rock mass within the site at Kamlunge.

There are only a few small lakes up on Kamlungekölen. Larger lakes are located below the site, such as Stora Lappträsket (4.1 km^2), Byträsket, Idträsket and Granträsket. The catchment area for Stora Lappträsket is about 15 km^2 and the lake's theoretical water turn-over time is 5-8 years. The turnover time in the lower part of the Kalixälven River is 1 day, and in the coastal zone at Repskärsjärden 10 days.

Kamlungekölen's topography is such that it forms an inflow area for groundwater. The bordering valleys constitute outflow areas, including groundwater from great depths /18-12/. Stora Lappträsket is the first major recipient for surface water and groundwater from Kamlungekölen. The natural concentrations of uranium and radium in the lake are shown in Table 18-12.

Mean precipitation in Kamlunge is 690 mm/y. Of that total 45% falls as snow. Evaporation is estimated at 350 mm/y /18-3, 18-4/. Runoff is estimated at 340 mm/y or about 10 $\text{l/(s} \cdot \text{km}^2)$.

Table 18-12.   Measured concentrations of uranium and radium in the
              Kamlunge area.

	Uranium	Radium	Unit
Sediment	100	80	$Bq\ kg^{-1}$
Lake water	0.004	<0.001	$Bq\ 1^{-1}$
Soil	90	50	$Bq\ kg^{-1}$

18.4.6   Hydraulic conductivity of the bedrock

The bedrock in Kamlunge has been divided into different hydraulic
units as follows:

-     local fracture zones (7)
-     horizontal fracture zone (1)
-     the rock mass

Hydraulic conductivity has been measured in 187 sections 25 m long
and 72 sections 5 m long in boreholes Km 1-Km 4, Km 9 and Km 12-Km
14. The hydraulic conductivity in the horizontal fracture zone,
penetrated in its lowermost parts by boreholes Km 2, Km 13 and Km
14, was determined by measurement in longer sections. The hydraulic
conductivity of this zone has been calculated by assigning the
entire water flow measured in the tests to the zone. All measured
values representing the rock mass and local fracture zones are
shown in Figs. 18-28 and 18-29, respectively.

Figs. 18-28 and 18-29 show the relationship between hydraulic
conductivity and depth, valid down to 700 m. The hydraulic conduc-
tivity of the horizontal fracture zone has been determined to
be 1×10^{-8} m/s (Table 18-11).

Measurements of hydraulic conductivity have been made in 175 sec-
tions 2 m long in borehole Km 2 in order to determine the hydraulic
fracture frequency. This fracture frequency is less than 0.1 frac-
ture/m at depths below 300 m. The measuring limit for these measure-
ments is 1.3×10^{-10} m/s.
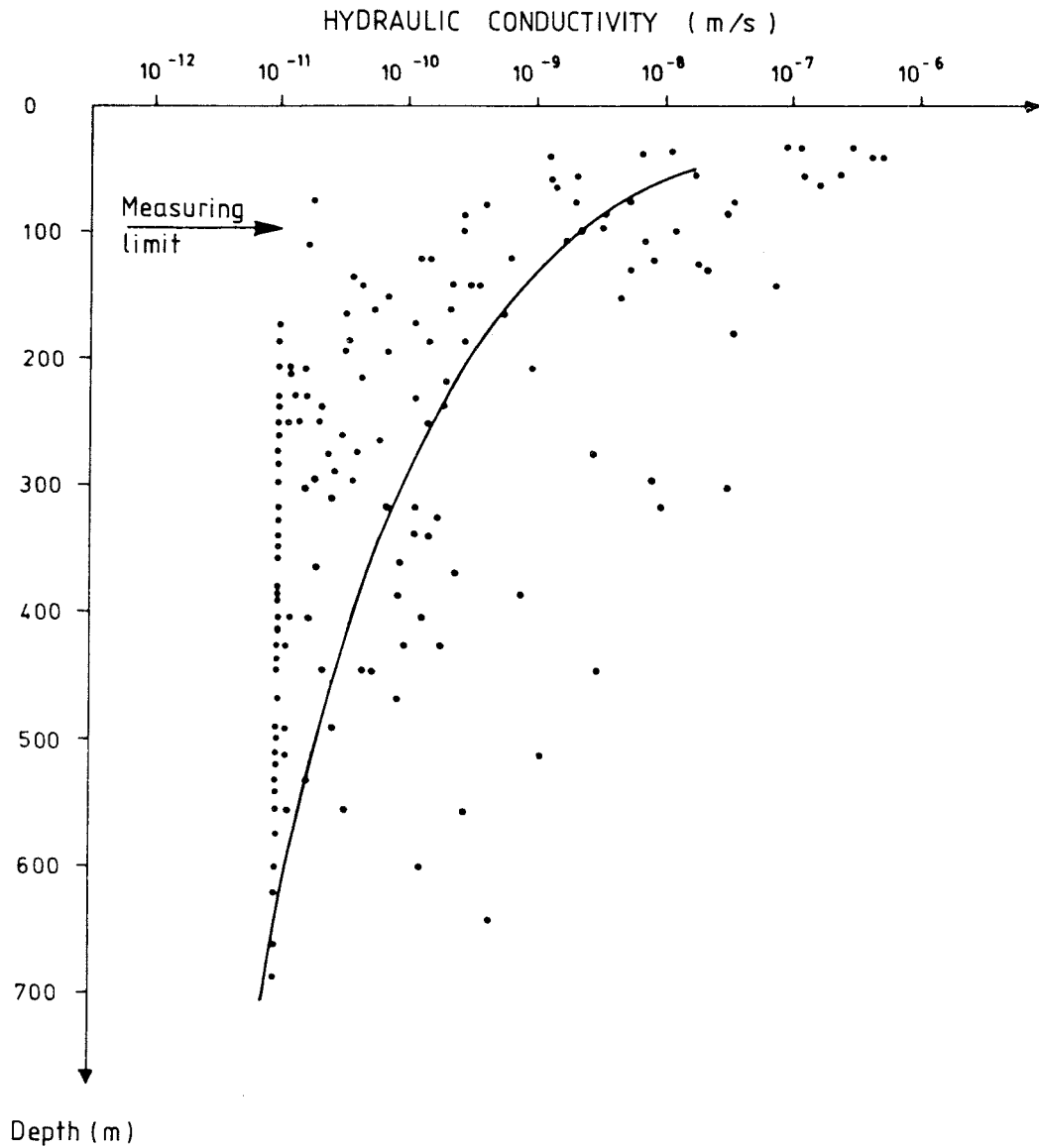
Figure 18-28. Correlation between hydraulic conductivity and depth for the rock mass at Kamlunge.

18.4.7   Groundwater chemistry

Chemical analysis have been carried out on water samples taken in boreholes Km 3 and Km 13 from altogether 3 different sections between 106 and 555 m depth.

The results of these analyses are presented in detail in /18-7, 18-8, 18-15/. Table 18-13 summarizes the sampled sections and analyses results of significance for characterizing the site.

HYDRAULIC CONDUCTIVITY (m/s)

10^{-12} 10^{-11} 10^{-10} 10^{-9} 10^{-8} 10^{-7} 10^{-6}

0

• 3

100

• 2

200

• 2

300

• 2

400

• 3

• 4

500

H1
• • H1
•
H1

600

700
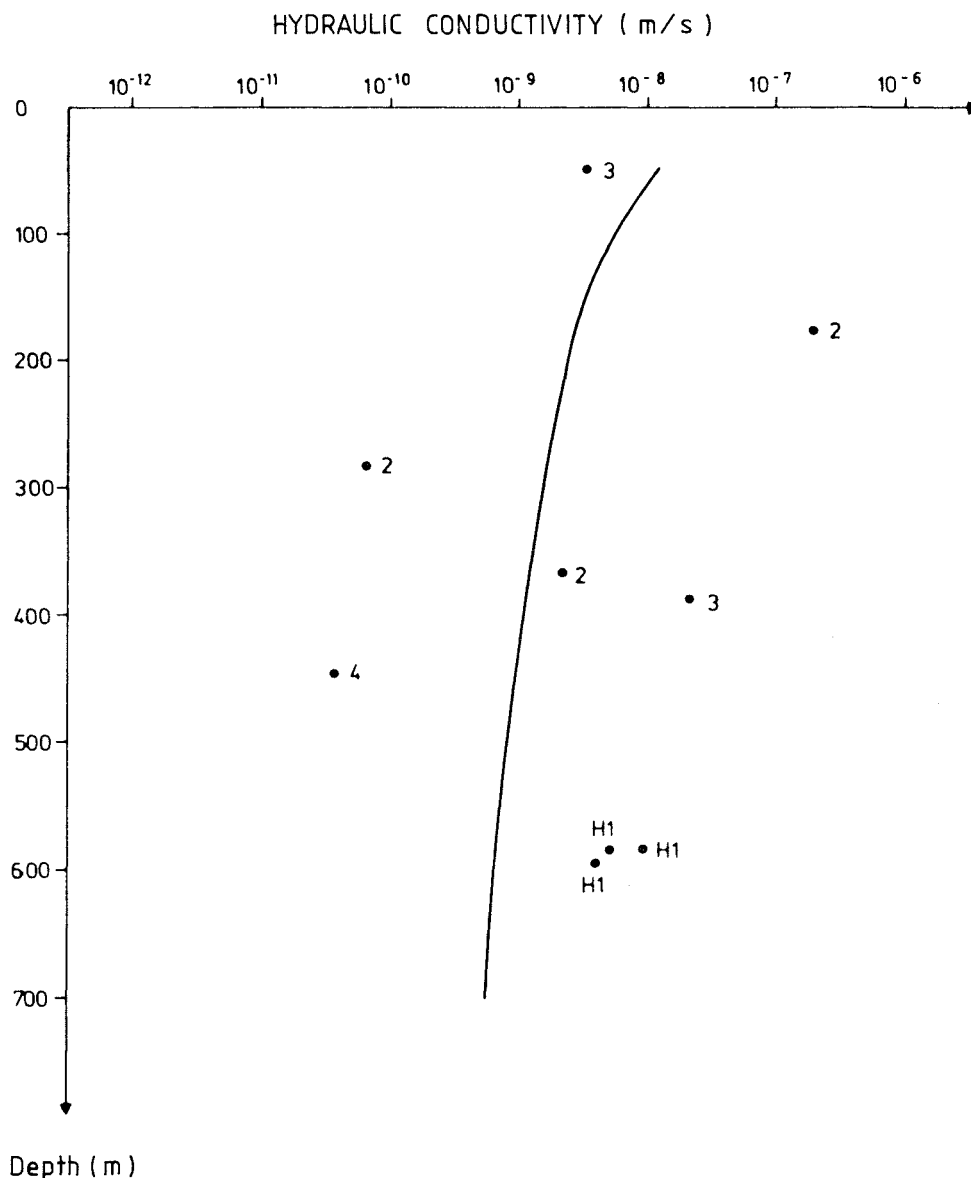
Depth (m)

Figure 18-29. Correlation between hydraulic conductivity and depth for fracture zones in Kamlunge. The horizontal fracture zone H1 is not indluded in the correlation curve.

Table 18-13.   Results of chemical analyses on groundwater from Kamlunge.

Borehole	Depth m	pH	Eh V	HCO_3^- mg/l	Cl^- mg/l	HS^- mg/l	Na^+ mg/l	Ca^{2+} mg/l	Mg^{2+} mg/l	Fe^{2+} mg/l	TOC[a] mg/l
Km 3	106	7.9	0.06	65	2	0.01	6	13	3	0.2	5
	376	8.7	-0.20	65	3	0.01	5	13	3	0.8	5
Km 13	555	7.6	-0.13	32	2	0.01	2	5	1	8.5	-

a Total organic carbon content of the water.

WIDTH OF FRACTURE ZONES

━ ━ ━ ━ ━ ━ 5 - 15 m

─ ─ ─ ─ ─ ─ ─ -5 m

Area available for
a repository

0 _____ 500 m

Interval of contourlines -2m

Km 1
Diamond drilled borehole

KAMLUNGE
450 m DEPTH
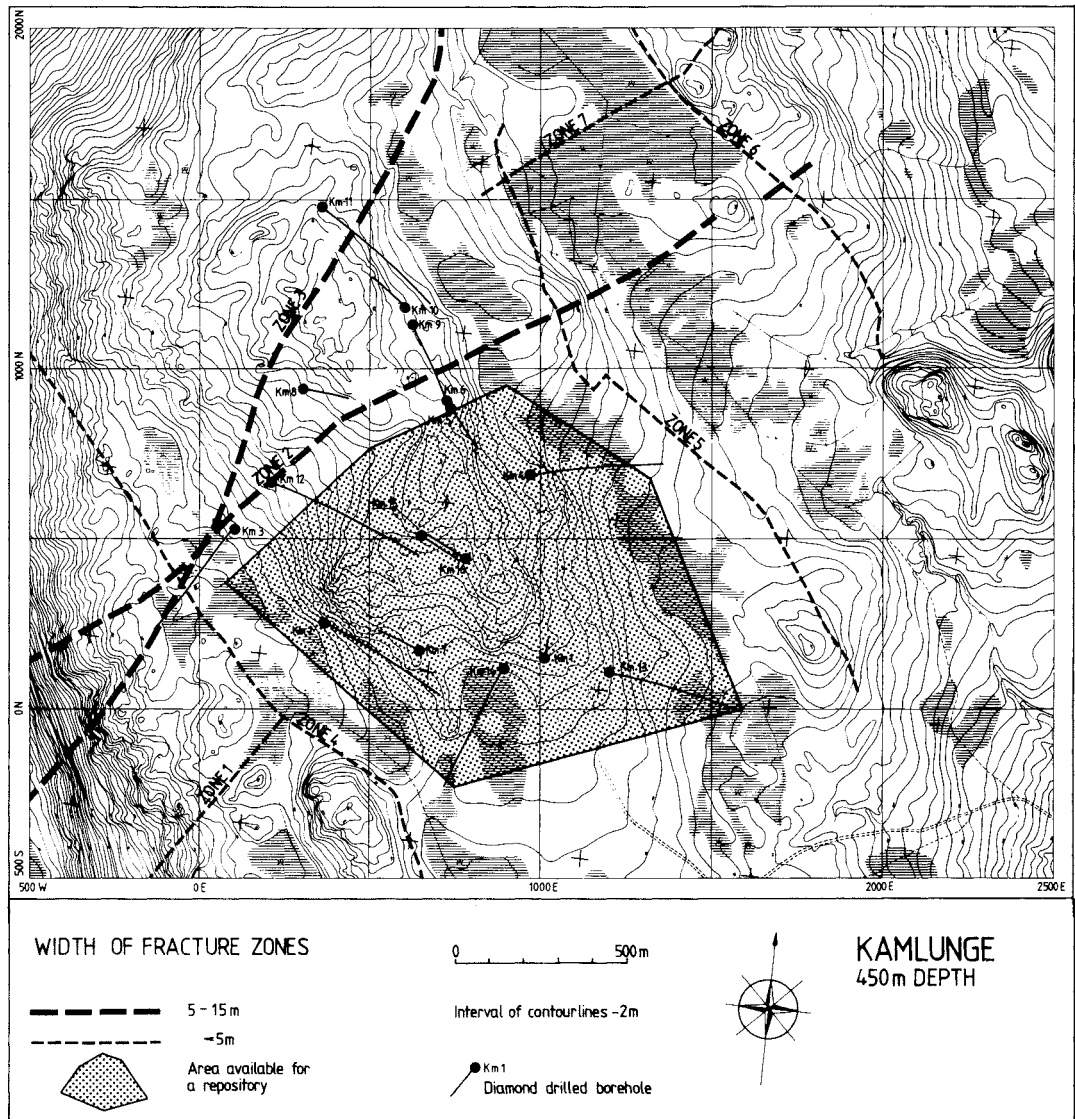
Figure 18-30. Location of the fracture zones at 450 m depth in Kamlunge. Available area for a repository at 450 m depth is indicated.

In the upper section at 106 m depth in Km 3, only positive redox potentials have been measured. This probably means that uranium here is oxidized and has a higher solubility than under the reducing conditions that prevail deeper down in the rock. However, for other radionuclides, the values given in chapter 12 for solubility and sorption under reducing conditions are applicable to all the sample levels.

The groundwater in Kamlunge has a relatively low content of dissolved salts, which favours the formation of colloidal particles. The problems connected with this are discussed in chapter 12.
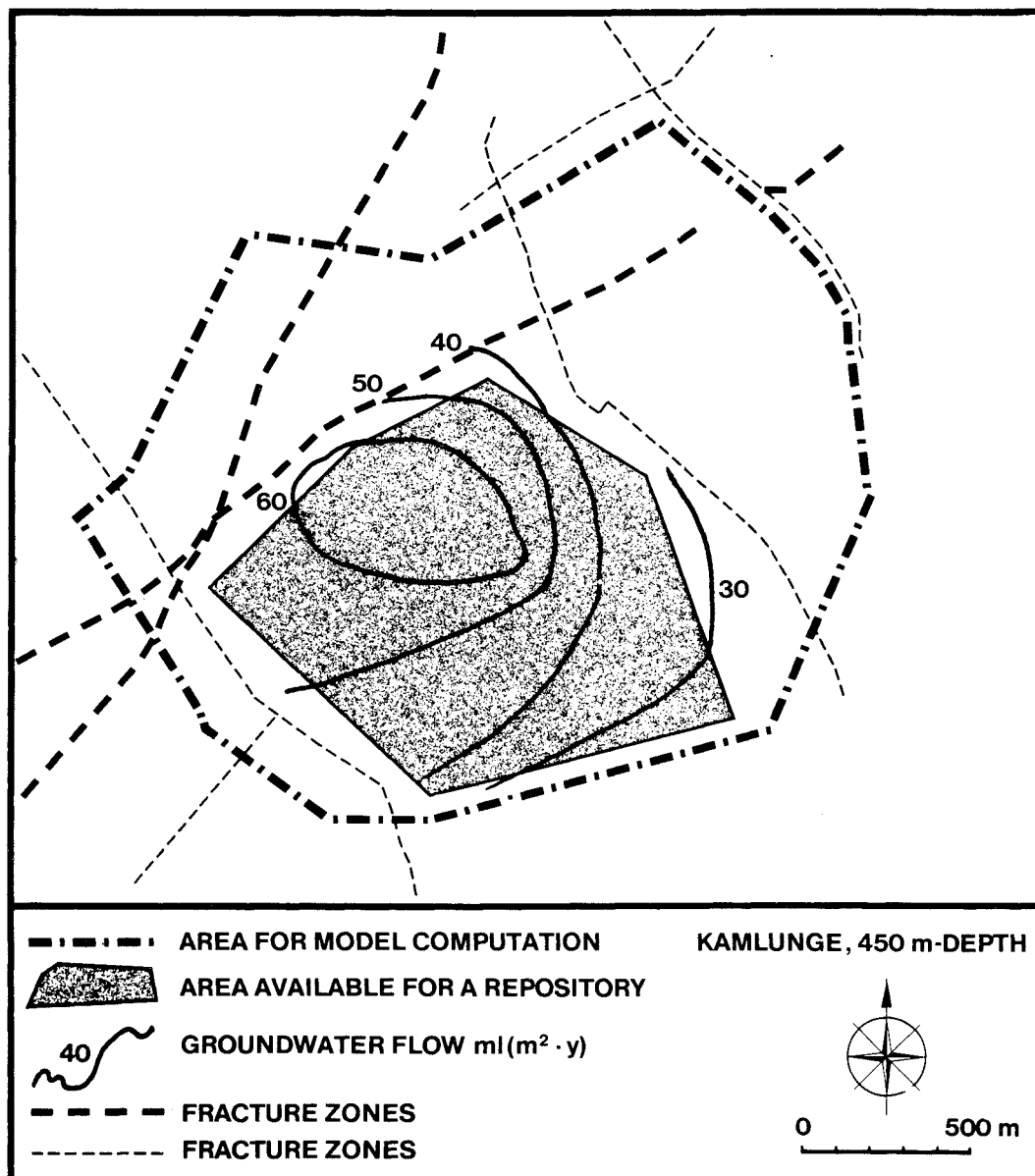
Figure 18-31. Groundwater flow at 450 m depth.

Migration of radionuclides in the form of colloids is only of importance at such high water flows that these in themselves would make the site unacceptable for a repository.

18.4.8   Available space for a final repository

The presence of a horizontal fracture zone at 555 m depth below Kamlungekölen means that a repository could be placed at about 450 m depth. This gives a "respect distance" of 100 m to the under-

lying horizontal zone. Fig. 18-30 shows the available space for a repository with a total area of 1.1 km^2. The repository has been located within a rock volume investigated by 6 deep and 3 short core boreholes. The repository is configured as a single-storey repository which gives about 50% more area than the net requirement. The model calculations carried out show that from a hydraulic viewpoint it would be more advantageous to locate the repository directly below the horizontal zone, where the groundwater flow is extremely small. However, investigations to date in Kamlunge, have not included the bedrock beneath the zone.

The character of the site is such that supplementary investigations may be expected to show that additional rock volume suitable for the deposition of spent fuel, either in a one- or two-storey repository, exists.

18.4.9   Model calculations

Model calculations of groundwater conditions within Kamlunge cover an area of 3.3 km^2 (Fig. 18-31). Kamlungekölen's strongly varying topography with bordering valleys at a level about 100 m lower means that there are high hydraulic gradients at the margins of the site. A regional model calculation embracing Kamlungekölen and the bordering valleys (34 km^2) was carried out to shed light on these factors and to obtain groundwater pressure as a boundary condition for the local modelled area.

In the model calculations, the horizontal zone has been assumed to exend over the entire area being modelled.

The results of the model calculations show that the groundwater flow at repository depth varies between 0.003 and 0.06 $1/(m^2 \cdot y)$ (Fig. 18-31). The groundwater recharge to the bedrock within Kamlunge, according to the calculations, is about 20 mm/y. This means that the main part of the groundwater flow takes place through the soil.

Transport and uptake paths in the biosphere have been calculated /18-9/. The model system for Kamlunge is shown in Fig. 18-32.
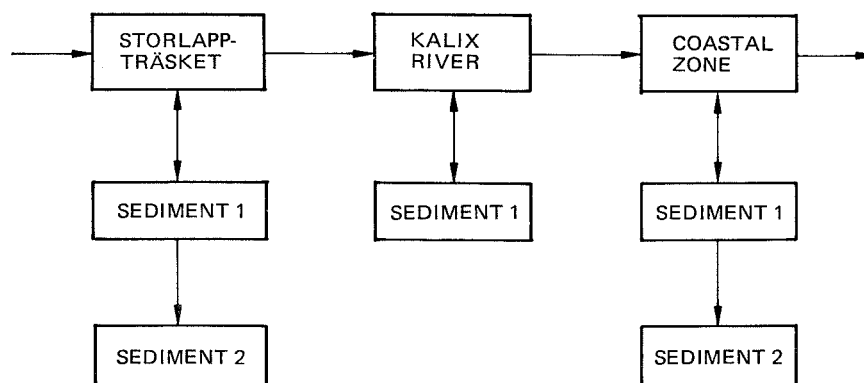
Figure 18-32. Flowchart for BIOPATH model's reservoir system for Kamlunge.

18.4.10 Concluding remarks

The study site at Kamlunge is situated within a 16 km^2 plateau in which the surface topography shows local relief of up to 30 m. Local fracture zones are narrow, and the rock mass has low hydraulic conductivity.

The local fracture zones are widely separated, which provides good possibilities for a single-storey repository. The horizontal fracture zone encountered at about 550 m depth means that a repository has to be located at 450 or 650 m depth beneath the surface of Kamlungekölen. Since no investigations have been carried out in bedrock beneath the horizontal zone, all calculations apply to a repository located at 450 m depth.

The local fracture zones within the area have a hydraulic conductivity at 450 m depth of 1 x 10^{-9} m/s (40 times greater than that of the rock mass). The model calculations show that the groundwater flows at 450 m depth within the site of a conceivable repository are small, and that the large hydraulic gradients found in the marginal zones of Kamlungekölen do not affect the groundwater conditions within the conceivable repository site. As on the sites already discussed, the groundwater at Kamlunge is of a composition that does not affect the safety of an intended final repository. The site is considered to posses good potential for the location of a single-storey repository at 450 m depth.
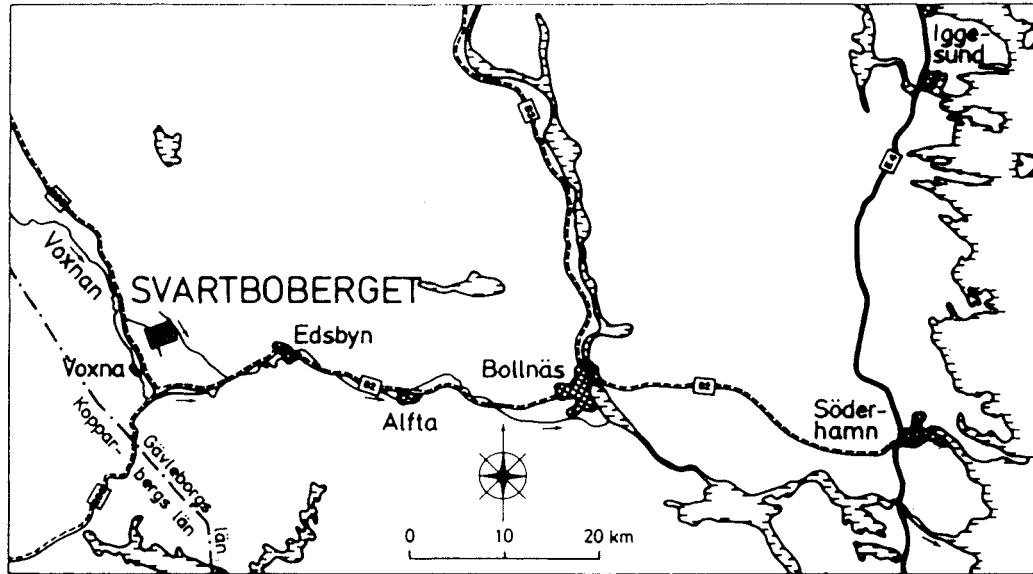
Figure 18-33. Locality map for study site at Svartboberget.
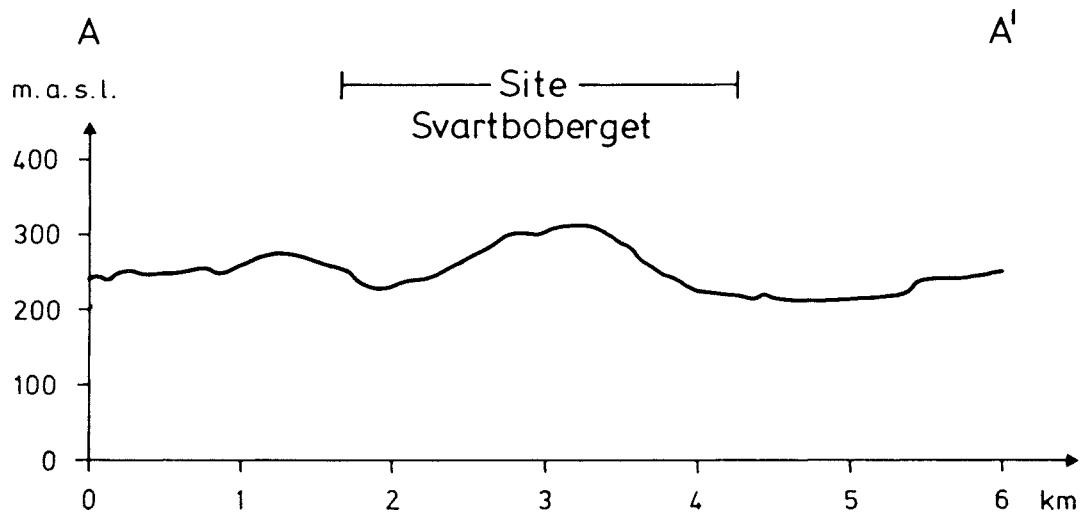
Figure 18-34. Topographical profile through the site at Svartboberget. Location shown in figure 18-35.

18.5 SVARTBOBERGET

18.5.1 Location and topography

The study site at Svarboberget is located in western Hälsingland c. 12 km W of Edsbyn (Fig. 18-33). The topography in the area between Voxnan and Edsbyn is characterized by NW-oriented fracture valleys located 2-2.5 km apart. The site consists of a ridge about 2.5 km
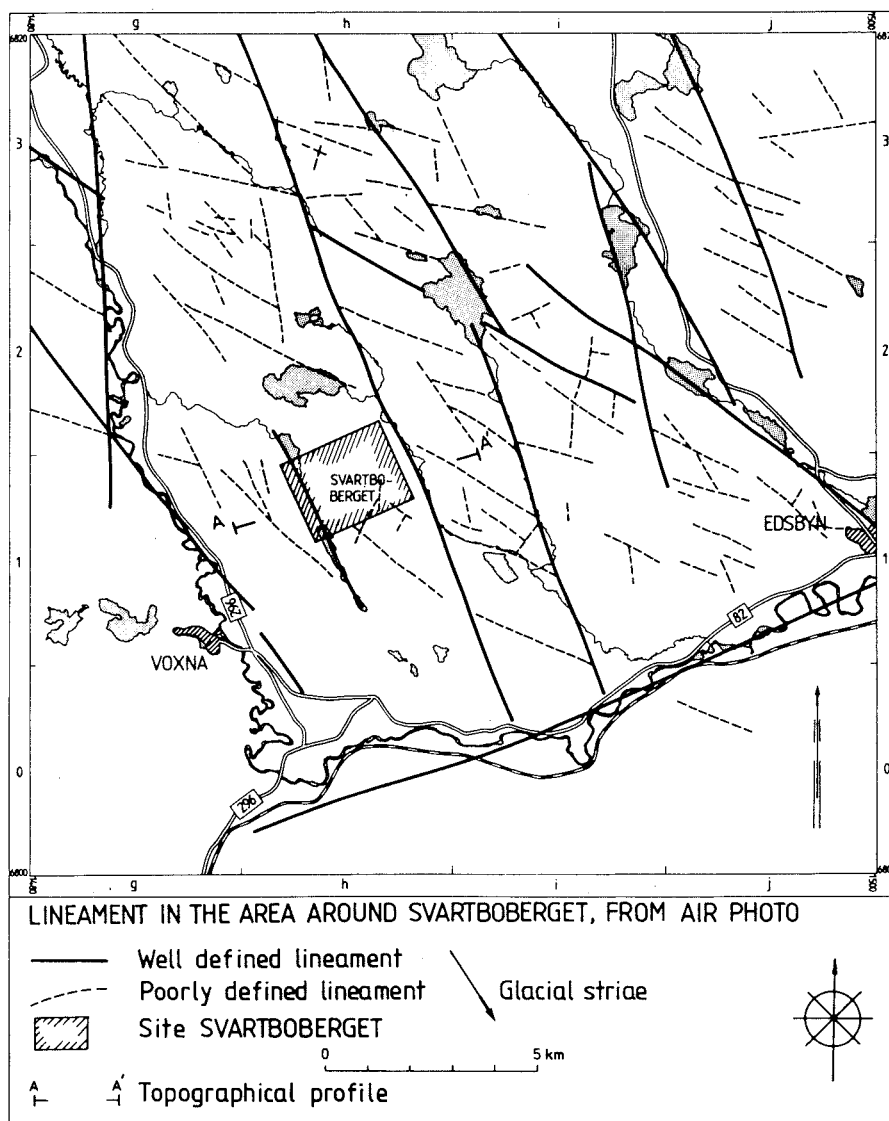
Figure 18-35. Interpreted lineaments in area around Svartboberget.

wide and 5 km long. The relief between Svartboberget's upper part, about 305 metres above sea level, and the bordering valleys to the E and W is 85 m and 75 m, respectively. The topography of the site is shown in Fig. 18-34.

The site is forested. The Quaternary deposits are bouldery moraine. The soil thickness is 0-2 m up on the ridge and 6-8 m on the hill-sides. Despite the thin soil cover, the amount of exposed rock is low, less than 1%.

A 2 x 2 km area within Svartboberget was investigated in detail (Fig. 18-35). The results of the investigations are presented in /18-16/.

18.5.2   Bedrock geology

The main rock type within the site at Svartboberget is a highly metamorphosed gneiss, migmatite. This rock type was formed in the same way as Fjällveden's veined gneiss, and is of the same age /18-17/. However, Svartboberget's bedrock is more highly metamorphosed than that in Fjällveden. That means that the gneiss contains a higher proportion of melted and recrystallized portions and fewer veins compared with Fjällveden's gneiss. There is a 150 m wide band of graphitic gneiss (9-13 wt. % carbon) in the northern part of the site. There are 2 narrow diabase dykes in the western part. There are, in addition, small bodies of amphibolite and pematite.

The essential minerals in the veined gneiss are quartz, plagioclase and biotite. The proportions of the essential minerals vary. The graphitic gneiss commonly has a high content of sulphides, up to 10%. The dominant sulphide mineral is pyrrhotite, but iron pyrites and chalcopyrite (less than 0.8 wt. % copper) also occur. The amount of ore minerals is low, but the possibility cannot be ruled out that the graphite gneiss could become of economic interest in the future.

18.5.3   Fracture zones

Seven core (diamond) boreholes and 16 percussion boreholes have been drilled in Svarboberget. The dip and length of the core boreholes are given in Table 18-14, and their locations are shown in Fig. 18-36.

Svartboberget is bounded on the NE and SW by NW-oriented valleys (Fig. 18-36), which coincide with regional fracture zones. The valleys consist of several parallel fracture zones up to 30 m wide. These zones have a total width of 150 m and dip of about 40° to the SW. One part of the NE zone (zone 18) has been encountered at great depth in 3 of the core boreholes (Table 18-15).
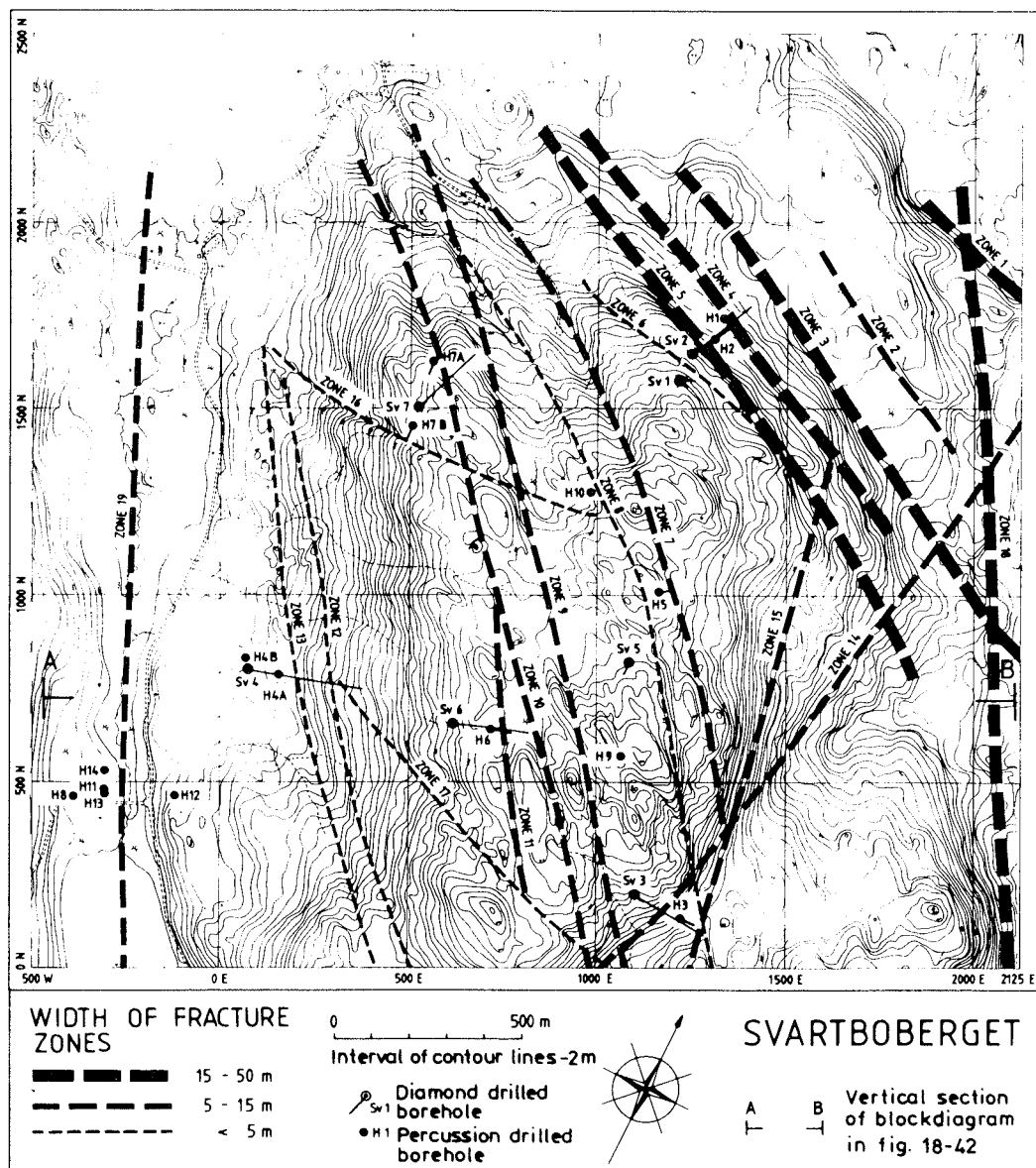
Figure 18-36. Fracture zones at the surface within the study site at Svartboberget.

Table 18-14.   Summary of length and dip of core (diamond) drill
                boreholes on site at Svartboberget.

Boreholes	Dep (degrees)	Borehole length (m)	Borehole depth (m)
Sv 1	90	787.9	786
Sv 2	58	400.4	340
Sv 3	62	450.5	398
Sv 4	60	641.5	569
Sv 5	86	801.5	800
Sv 6	74	750.3	714
Sv 7	60	396.9	339

Table 18-15.   Resumé of fracture zones in Svartboberget.

Fracture zone	Position in borehole (m)	Dip (degrees)	True width (m)	K value m/s
1	SV 1 (682-713)	40 S	25	6×10^{-12}
2	Sv 1 (458-467)	45 S	10	5×10^{-8}
	Sv 2 (295-311)	45 S	15	not measured
3	Sv 1 (275-312)	45 S	25	5×10^{-8}
	Sv 2 (144-177)	45 S	25	not measured
4	Sv 1 (169-223)	40 S	50	2×10^{-8}
	Sv 2 (50-119)	40 S	50	not measured
5	Sv 1 (45-104)	30 S	40	4×10^{-7}
	Sv 2 (0- 22)	30 S	20	not measured
6	Sv 1 (35- 37)	30 S	2	2×10^{-7}
7	Sv 5 (371-401)	65 SV	10	5×10^{-10}
8	Sv 3 (245-250)	60 SV	5	3×10^{-11}
	Sv 5 (128-136)	60 SV	5	1×10^{-8}
	Sv 6 (666-685)	60 SV	5	$<8 \times 10^{-11}$
	Sv 7 (383-390)	70 SV	5	4×10^{-9}
9	Sv 6 (666-685)	80 SV	10	$<8 \times 10^{-11}$
	SV 7 (383-390)	85 SV	5	4×10^{-9}
10	Sv 6 (591-633)	85 SV	10	2×10^{-10}
	SV 7 (140-142)	85 SV	5	$<1 \times 10^{-10}$
11	Sv 6 (475-506)	90	10	7×10^{-9}
12	-	75 Ö*	5*	-
13	Sv 4 (545-549)	75 Ö	4	$<3 \times 10^{-11}$
14	Sv 3 (58- 83)	40 V	25	2×10^{-7}
	Sv 5 (371-401)	40 V	10	5×10^{-10}
	Sv 6 (666-685)	40 V	10	$<8 \times 10^{-11}$
15	Sv 3 (100-170)	30 V	30	9×10^{-8}
	Sv 4 (643-648)	30 V	5	2×10^{-9}
	Sv 5 (254-270)	30 V	15	2×10^{-8}
	Sv 6 (475-506)	40 V	10	7×10^{-9}
16	-	75 N*	5*	-
17	Sv 4 (435-437)	85 S	2	2×10^{-7}
18	Sv 1 (736-788)	35 SV	>30	1×10^{-9}
	Sv 3 (441-448)	35 SV	> 5	8×10^{-9}
	Sv 5 (727-737)	35 SV	15	3×10^{-8}
19	-	35 SV*	>30*	-

*Calculated from geophysical observations.
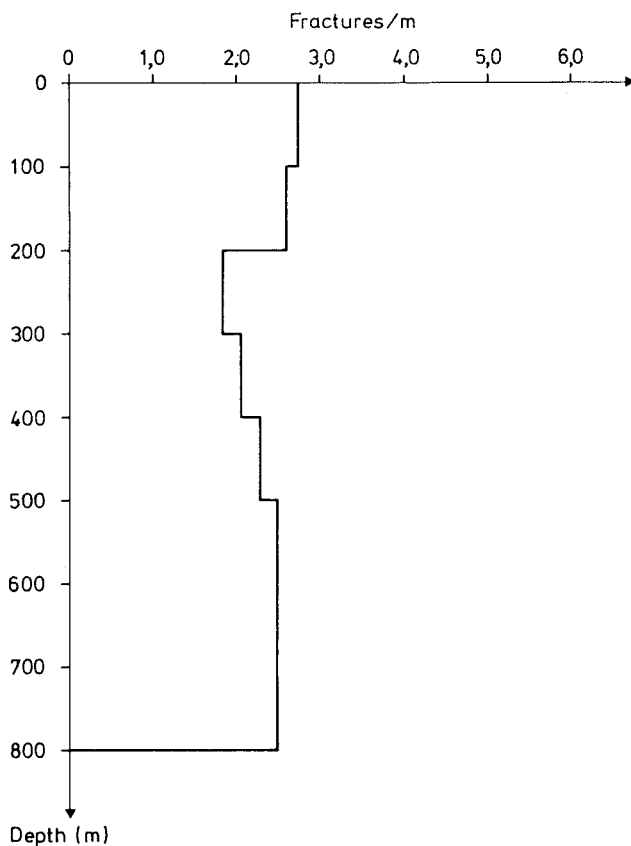
Figure 18-37. Fracture frequency in the rock mass within the site at Svartboberget.

The regional fracture zones confine an approx. 5 km^2 block. Within this block there are fracture zones of a more local nature. These zones are generally located 100-500 m apart. The majority of the zones dip to the W (Fig. 18-42).

The fracture zones have been penetrated by core boreholes in 29 different places. The width of the fracture zones varies between 2 and 50 m, with an average width of 16 m. A resumé of the results of borehole investigations of fracture zones is shown in Table 18-15.

The most pronounced local fracture zones are in the northern part of the site, and are oriented in a NW direction, dipping 30-45° to the SW (zones 1-6). The width of these zones varies between 5 and 50 m, and they are located 30-80 m apart. Fracture minerals are calcite, chlorite, illite and, to a lesser degree, zeolite minerals. These minerals also occur in other fracture zones in Svartboberget. Locally, weathering indications of thrusting are seen along the gently dipping fracture zones.

Two thin diabase dykes run across the western part of the site. These occur together with fracture zones 12 and 13.

18.5.5   Fracturing of the rock mass

Fracturing in the rock mass was mapped partly from outcrops and partly from cores. The dominant fracture direction in outcrops is between N and NE, with a vertical dip. The fracture frequency in outcrops is 1.0 fracture/m. The fracture frequency varies little with depth, as is shown in Fig. 18-37. The fracture frequency in the rock mass is 2.5 fractures/m.

18.5.5   Hydrology, meteorology and recipient conditions

Svartboberget is drained to the W by Brynåsbäcken Creek, which runs into the Voxnan River about 7 km upstream of the village of Voxna. The eastern part of the site is drained by the Älmån River, which also runs into the Voxnan, about 15 km downstream of the village of Voxna. The mean water discharge in the Voxnan there is 24.2 m^3/s /18-18/. The Voxnan discharges about 40 km away into the Ljusnan River, after another 40 km into the Gulf of Bothnia.

Connected with the site investigated in detail are the lakes Råttjärnsjön (1.1 km^2), Norra and Södra Brynåssjön (0.15 and 0.25 km^2, respectively), and Älmsjön (0.30 km^2). Lakes cover about 4% of the region around Svartboberget. The catchment area for Älmsjön is about 6 km^2, and the theoretical water turnover time is about 6 months. Norrsjön Lake, situated about 20 km downstream of the Älmån River's inflow, has a theoretical turnover time of 2.8 days. At the outflow into Ljusnan, the turnover time is about 24 days.

The topography of the site is such that it forms an inflow area for groundwater. The bordering low-lying areas to the east and west comprise outflow areas for groundwater, including from deeper parts of the bedrock. Älmsjön is the first major recipient for surface water and groundwater from Svartboberget. The natural concentrations of uranium and radium in the lake are shown in Table 18-16.

Mean precipitation in Svartboberget is 715 mm/y. Of that total, 24% falls as snow. Evaporation is estimated at 390 mm/y /18-3, 18-4/. Runoff is estimated at 325 mm/y (about 10 $1/(s \cdot km^2)$).

Table 18-16.   Measured concentrations of uranium and radium at Svartboberget.

	Uranium	Radium	Unit
Sediment	50	13	Bq kg^{-1}
Lake water	<0.002	<0.001	Bq 1^{-1}
Soil	40	30	Bq kg^{-1}

18.5.6   Hydraulic conductivity of the bedrock

The hydraulic units in Svartboberget's bedrock comprise:

- regional fracture zones (2)
- local fractures zones (17)
- the rock mass

The location of the regional and local fracture zones is shown in Fig. 18-36. Hydraulic conductivity has been measured in 147 sections 25 m long and 87 sections 5 or 10 m long in boreholes Sv 1, Sv 3 Sv 7. All measured values representing the rock mass, regional and local fracture zones are shown in Figs. 18-38, 18-39 and 18-40, respectively. The calculated relationship between depth and hydraulic conductivity is valid down to 800 m for the rock mass and to 700 m for the fracture zones.

18.5.7   Groundwater chemistry

Chemical analyses have been carried out on water samples taken in the two boreholes Sv 4 and Sv 5 from altogether 8 different sections between 82-712 m, vertical depth. However, the sampling in Sv 5 was spoilt by technical mishaps.

HYDRAULIC CONDUCTIVITY (m/s)
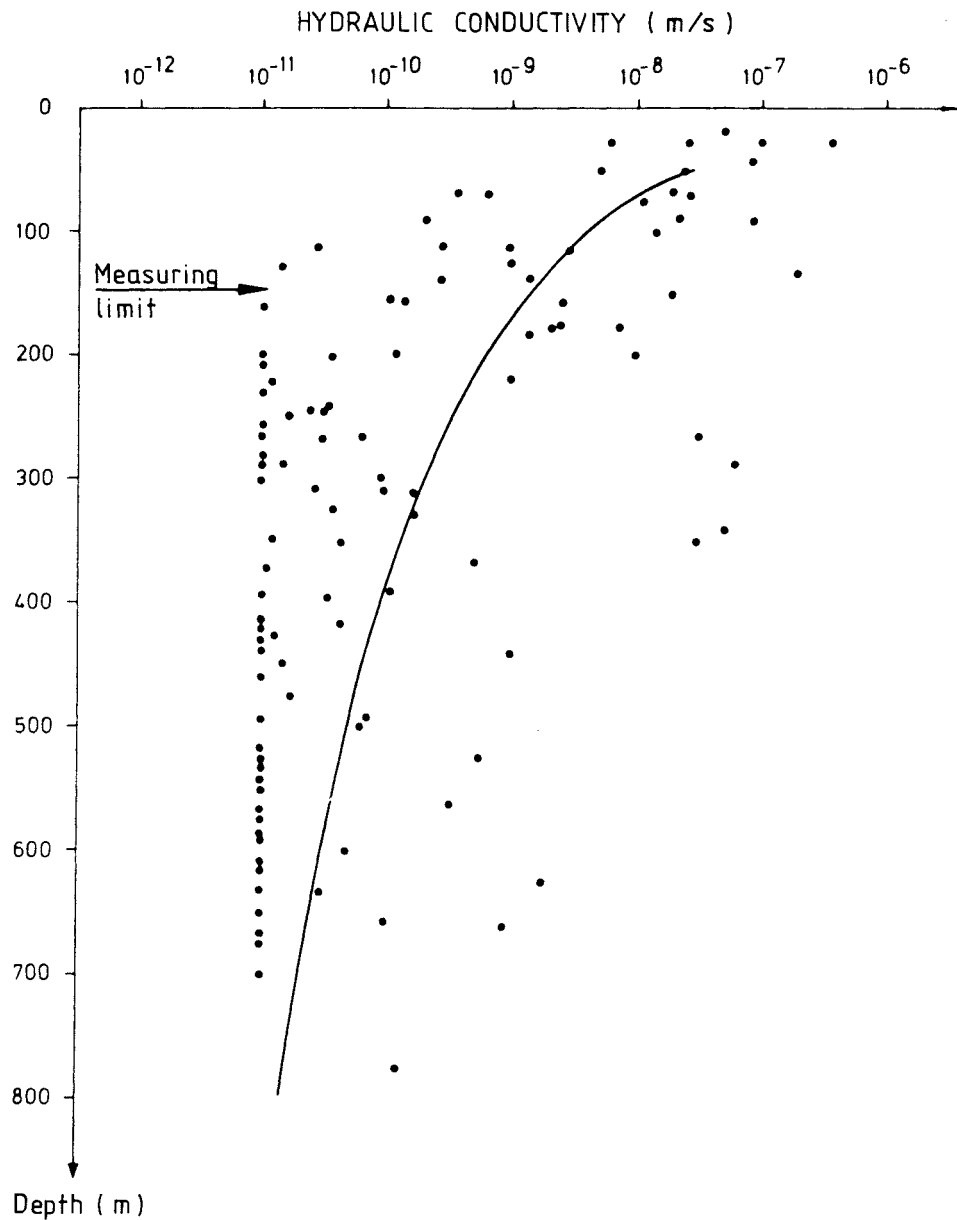
Depth (m)

Figure 18-38. Correlation between hydraylic conductivity and depth for the rock mass at Svartboberget.

The results of the analyses in Sv 4 are presented in detail in /18-7, 18-8, 18-19/. A number of analyses from sampled sections in Sv 4 are summarized in Table 18-17.

In general, the analysis results agree with what was presented in chapter 17. An exception is the content of organic carbon, which is unusually high in samples from the section at 324 m depth. High amounts of organic matter may, if they consist of high-molecular-weight humic and fulvic acids, increase the mobility of the triva-
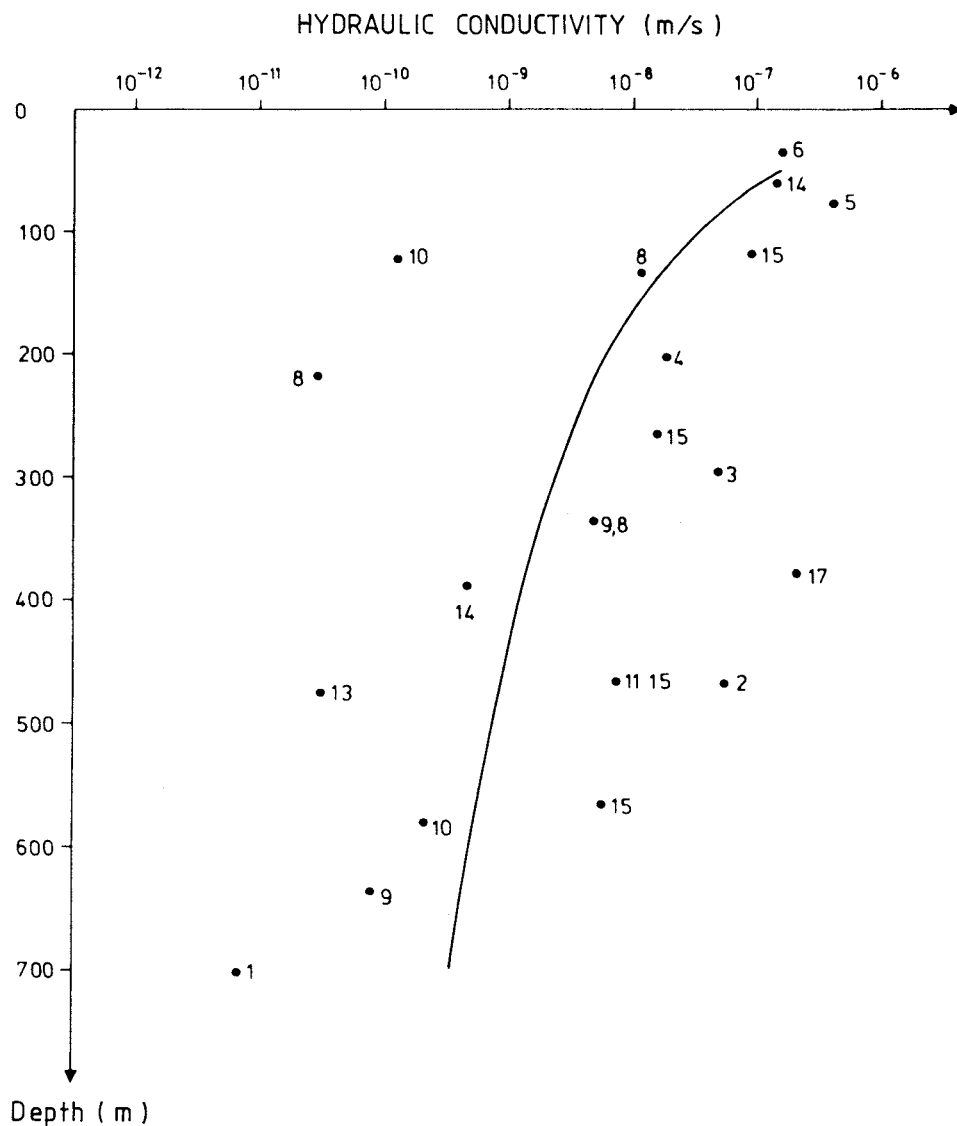
Figure 18-39. Correlation between hydraulic conductivity and depth for the local fracture zones in Svartboberget.

Table 18-17. Results of chemical analyses on groundwater from Svartboberget.

Borehole	Depth m	pH	Eh V	HCO_3^- mg/l	Cl^- mg/l	HS^- mg/l	Na^+ mg/l	Ca^{2+} mg/l	Mg^{2+} mg/l	Fe^{2+} mg/l	TOC^a mg/l
Sv 4	82	8.4	-	139	3	0.01	23	25	3	0.1	3
	324	9.6	-0.14	127	10	0.03	44	12	1	0.2	10
	373	9.1	-0.08	131	7	0.01	35	18	2	0.3	3
	551	9.0	-0.14	128	8	0.05	35	17	2	0.3	4

a Total organic carbon content of the water.

HYDRAULIC CONDUCTIVITY (m/s)
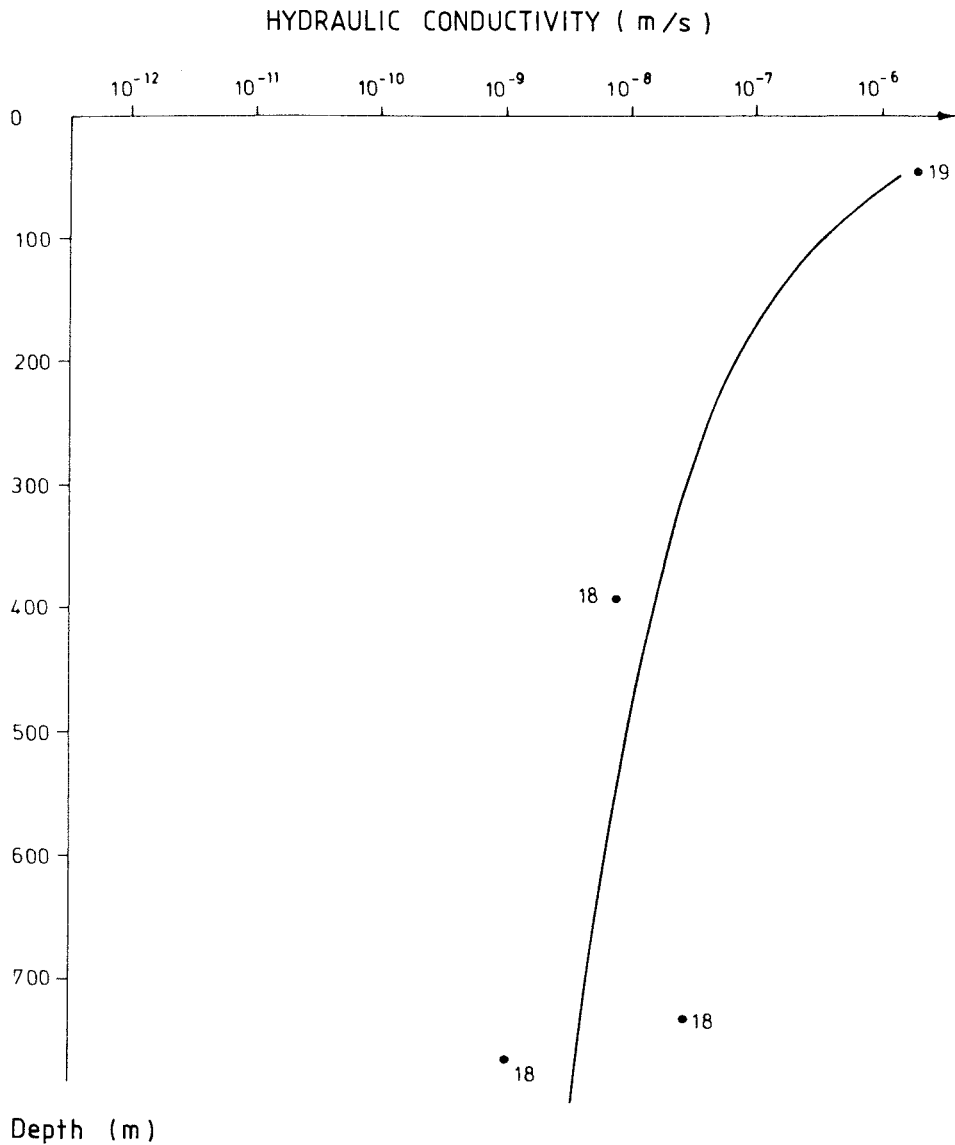
Depth (m)

Figure 18-40. Correlation between hydraulic conductivity and depth for the regional fracture zones in Svartboberget.

lent actinides in particular. Otherwise, the conclusions concerning solubility and sorption of radionuclides presented in chapter 12 for reducing conditions are applicable.

18.5.8   Model calculations

Model calculations of groundwater conditions have not been carried out for Svartboberget.
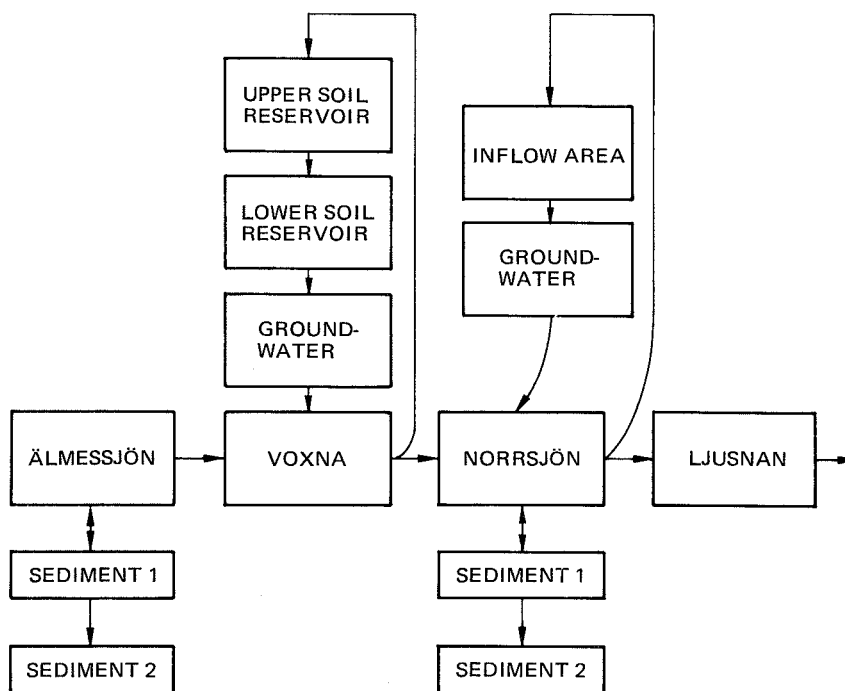
Figure 18-41. Flow chart for BIOPATH model's reservoir system for Svartboberget.

However, transport and uptake paths for radionuclides in the bio-sphere have been calculated /18-9/. The model system for Svartbo-berget is shown in Fig. 18-41.

18.5.9   Concluding remarks

The study site at Svartboberget consists of a ridge with a relief of 80 m above the surrounding valleys. The area is intersected by several local fracture zones. The rock mass between the fracture zones has low hydraulic conductivity.

The local fracture zones on the site have a hydraulic conductivity of about 8×10^{-10} m/s at 500 m depth (17 times greater than that of the rock mass). These zones dip about 45° and are located 30-300 m apart.

Analyses of groundwater indicate a composition which would not affect the safety of a final repository.
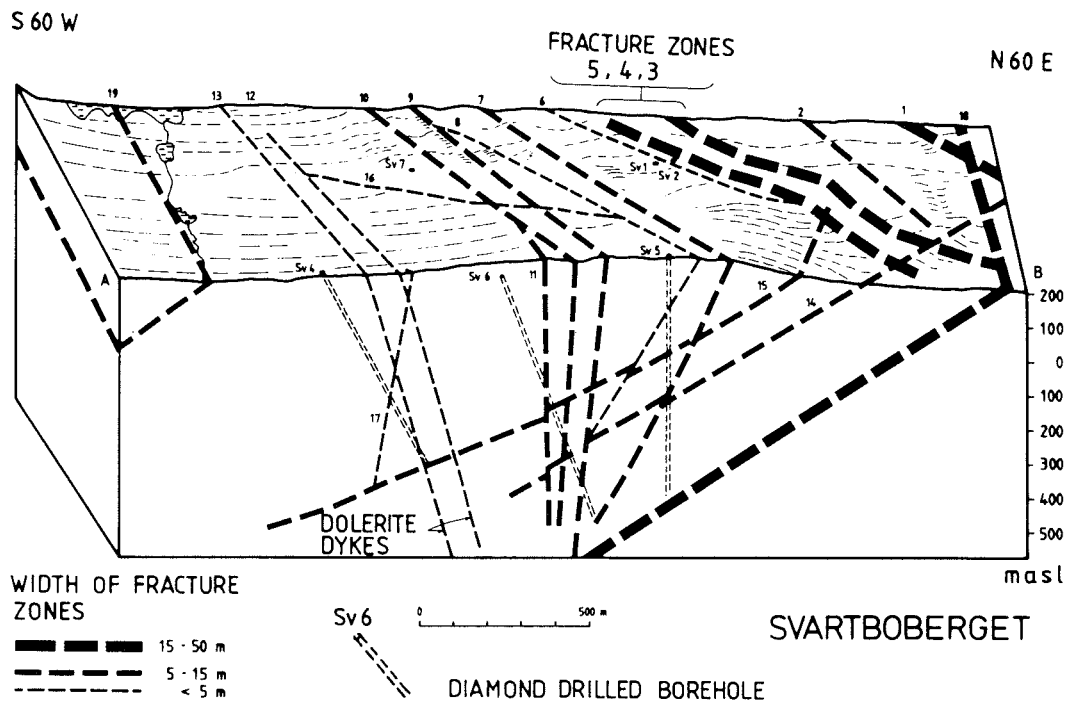
Figure 18-42. Vertical section through Svartboberget. Location of the profile is shown in figure 18-36.

The large number of local fracture zones with little separation (Fig. 18-42) means that the space for a repository is very limited. The western part of the site has a lower proportion of local fracture zones. The number of boreholes in that area is too small to obtain an acceptable interpretation of existing fracture zones, however.

Hydrological model calculations of groundwater conditions in Svartboberget have not been carried out. A final repository in Svartboberget would involve practical difficulties regarding the repository's geometrical configuration, since the space between the existing fracture zones is limited. Moreover, there is a graphite-bearing gneiss in the area which could become of economic interest in the future. Of the sites considered, Svartboberget is therefore the least suitable for a final repository.
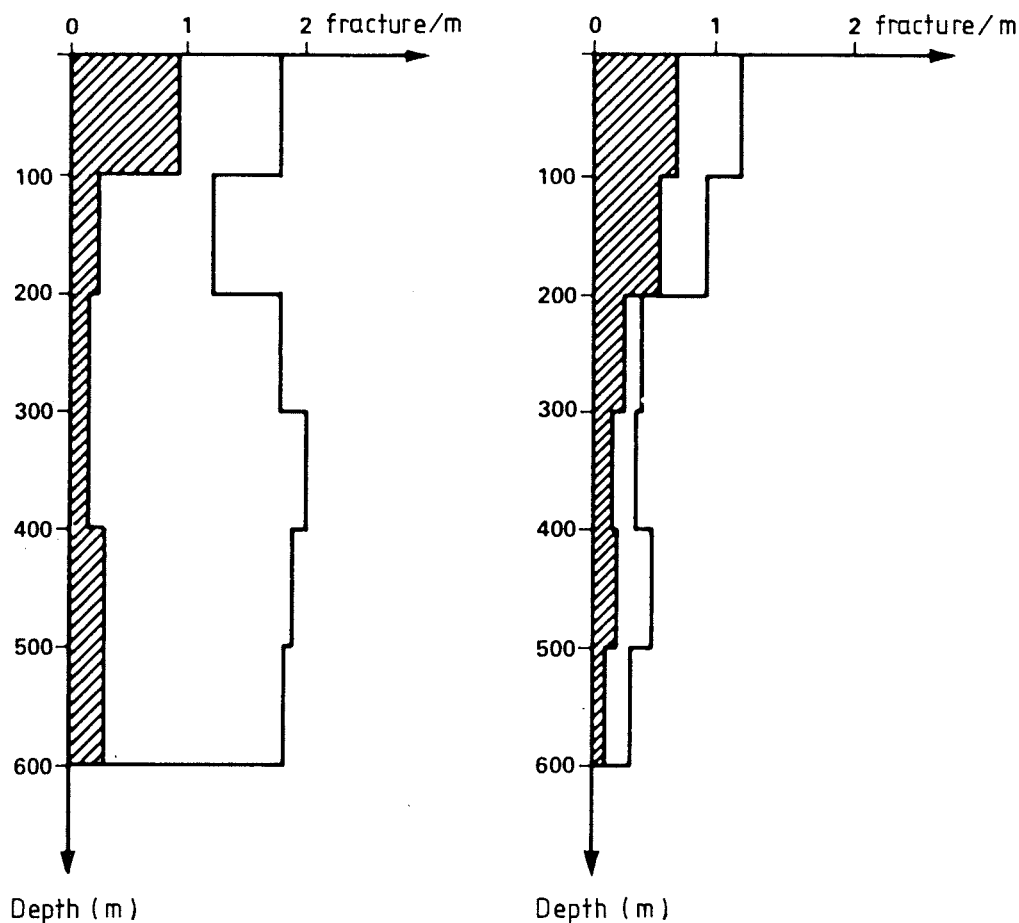
Figure 18-43. Total fracture frequency and hydraulic fracture frequency in the rock mass within the sites at Finnsjön (left-hand figure) and Sternö (right-hand figure).

18.6 RESULTS FROM INVESTIGATIONS IN OTHER AREAS

18.6.1 Scope

A large number of geological, geophysical and hydrogeological investigations have previously been carried out within the sites of Finnsjön, Kråkemåla and Sternö /18-20-18-28/ (Fig. 18-1). A summary for Finnsjön is given in /18-29/. In addition, certain investigations have been carried out in Studsvik in eastern Södermanland. The results are to some extent site-specific but at the same time they provide information on general properties and conditions in crystalline bedrock. These results include:

- Hydraulic fracture frequency in Finnsjön and Sternö
- Kinetic porosity in Finnsjön and Studisvik
- Hydrogeological model calculation of groundwater conditions in Finnsjön

18.6.2 Hydraulic fracture frequency in Finnsjön and Sternö

The hydraulic fracture frequency in Finnsjön and Sternö was determined by means of tests in 2 or 3 m sections in existing boreholes. These measurements have been used together with the results of core mapping to determine the proportion of fractures which form water conduits in the rock mass /18-30/. 467 measurements in Finnsjön and 399 in Sternö have been used for the calculations. The results are shown in Fig. 18-43. On both sites, the hydraulic fracture frequency decreases with depth. At levels below 300 m, the fracture frequency in Finnsjön is 0.2-0.3 fracture/m, and in Sternö 0.1-0.2 fracture/m. The measured data comprise mean values from 5 boreholes per site. The measuring limit for measurements in Finnsjön was 2.5×10^{-9} m/s, while in Sternö it was 4×10^{-10} m/s.

18.6.3 Kinematic porosity in Finnsjön and Studsvik

Kinematic porosity (see section 6.1.2) has been determined by establishing travel times for tracers added to the groundwater. The field tests have been carried out in Finnsjön /18-31/ and Studsvik /18-32/. The results are given in Table 18-18.

Table 18-18.   Kinetic porosity determined from field tests.

Rock type	Site	Porosity (%)
Granodiorite fracture	Finnsjön	0.08-0.09
Granitic gneiss fracture	Studsvik	0.2-0.5

18.6.4   Hydrogeological model calculation of groundwater conditions in Finnsjön

In connection with the development of the numerical model used for calculating groundwater conditions in Fjällveden, Gideå and Kamlunge, the model has been used to elucidate groundwater conditions in Finnsjön /18-24/. Both methodology and equipment have been improved considerably since these investigations were carried out. The model calculation gave flows of 0.2-1 $l/(m^2 \cdot y)$.

19 RADIOLOGICAL SAFETY DURING OPERATION

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┌──────┐   ┌──────────┐   ┌──────┐   ┌─────────┐   ┌─────────┐   ┌──────────────┐
│SPENT │──▶│FACILITIES│──▶│NEAR  │──▶│GEOSPHERE│──▶│BIOSPHERE│──▶│   SAFETY     │
│FUEL  │   │          │   │FIELD │   │         │   │         │   │┌────────────┐│
└──────┘   └──────────┘   └──────┘   └─────────┘   └─────────┘   ││OPERATING   ││
                                                                  ││PERIOD      ││
                                                                  │└────────────┘│
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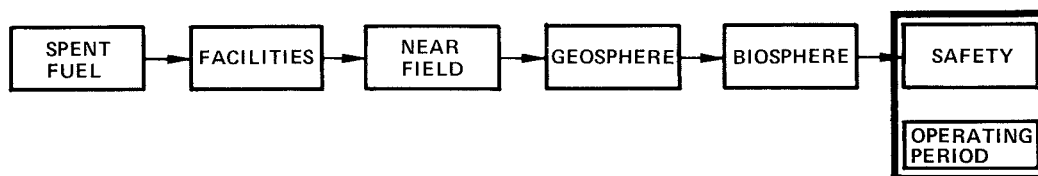
In this chapter, the radiological safety aspects of the operating period are analyzed. This includes all steps in the handling, transport and storage of the spent nuclear fuel, from the time it leaves the nuclear power station to the time the final repository is sealed. The operating period can be divided into a number of activities:

- Transports
- Intermediate storage in the CLAB
- Encapsulation and other treatment prior to final storage
- Emplacement in the final repository

The manner in which these activities can be carried out has been described in chapter 4.

19.1 GENERAL

The facilities for the handling, storage and treatment of spent fuel are designed in accordance with the safety principles described in chapter 17. Special importance is attached to:

- Minimizing the radiation doses to the personnel

- Preventing the fuel from being damaged due to accidents or improper handling

- Minimizing dispersal to the surroundings of any activity that may be released.

For parts of the handling system that have already been taken into operation or are under construction - namely the transport system and the central storage facility for spent fuel, CLAB - exhaustive safety analyses have been submitted to the authorities /19-1, 19-2/ and have received their approval. These parts of the system are therefore dealt with only briefly here. As a basis for an evaluation of the condition of the fuel when it is transported further to the encapsulation station and treated there, certain aspects of the long-term storage of spent fuel in water pools are described.

The encapsulation of spent fuel assemblies in copper canisters will be done at a special treatment station. It is assumed that this station will be co-sited with the final repository and built on groundlevel. The handling procedure within this facility is special for the proposed final storage scheme presented in this account. Its safety has therefore been analyzed and reported in special studies /19-3, 19-4, 19-5/, which are summarized in section 19.4 below.

Handling of the encapsulated fuel up to final storage is discussed briefly in section 19.6.

19.2　　TRANSPORTS

The main features of the Swedish transport system for spent nuclear fuel are described in section 4.2. The system is based on sea transport by a ship designed specially for the purpose. In addition, there are a special designed vehicle, load and lift devices etc. for the connecting overland transports. Safety analyses of the transports are presented in /19-1/.

The fundamental safety philosophy for the transports is defined by the IAEA's rules. The transport cask for spent fuel shall alone provide adequate radiation protection and shall constitute a leak-tight barrier between the spent fuel being transported and the environment. This applies both under normal conditions and in connection with accidents or failures.

In order to further limit the dose burden to the crew, the ship is equipped with permanent radiation shields. Normally, the individual doses to personnel who work with the transports will amount to about 2 mSv per person and year. The collective dose from the transport of 300 tonnes of fuel, which is the maximum annual transported quantity, has been estimated at about 0.2 manSv. The largest dose contributions are obtained during loading and unloading of the transport cask from the vessel.

The consequences of very severe but improbable accidents such as fire of long duration, forceful collision and loss of transport cask into the sea are also analyzed in /19-1/. The calculations show that the environmental consequences are very small in these cases as well. The maximum individual dose is less than 0.1 mSv.

The shipments of spent fuel from the CLAB to the encapsulation station take place 30-40 years later than the shipments to the CLAB. During this interval, the activity from important nuclides such as krypton-85 and cesium-134 has decayed for the most part, while the activity from cesium-137 has decreased by half. This means that the radiological consequences of the latter shipments are lower than those reported above.

19.3 CENTRAL STORAGE FACILITY FOR SPENT FUEL, CLAB

19.3.1 Radiological impact of normal operation and accidents

The design of the CLAB is described briefly in section 4.3 and in greater detail in /19-2/. This reference also provides a detailed safety analysis of the facility. Only a few results are reproduced here in order to facilitate a comparison with other links in the handling chain.

The storage section of the CLAB has been designed as a rock cavern facility, which provides good protection against external forces such as natural catastrophes of various kinds, acts of war and sabotage.

In the receiving section, which is located on the surface, fuel is only stored temporarily, but nearly all handling of fuel and transport casks takes place there.

All handling of fuel in the CLAB takes place underwater, which provides good radiation shielding. The dose burden to the personnel is therefore dominated by the various steps in the handling and discharge of the transport cask, which gives an average individual dose of about 3 mSv/y. The total individual dose for the most highly exposed individuals in the CLAB is estimated to lie below 5 mSv/y, on the average. The total collective dose to all personnel from work in the facility is expected to be about 0.3 manSv per year.

It has been estimated on the basis of pessimistic assumptions that normal operating releases from the CLAB will give individual doses of around 0.04 μ Sv per year and collective dose commitments of between 0.01 and 0.02 manSv. These are negligible values.

The consequences of various accidents such as handling accidents, loss of cooling of the water in the storage pools, fire etc. are also analyzed in /19-2/. The greatest consequence is obtained if a transport cask is dropped so that the fuel is damaged. The individual dose at a distance of 1 km from the facility is estimated to be 4 μSv for this case.

19.3.2 Long-term storage of spent fuel in water pools

The total time for storage of spent fuel in water pools prior to final storage is presumed to be 40 years.

Good experience of fuel storage in water pools exists from many years. No damage to the cladding or fuel has been observed in connection with this storage. Experience shows /19-6/ that general corrosion after 100 years of storage in neutral pure water scarcely amounts to more than 1 μm. Even with other reasonable water chemistry, general corrosion will not be altered in a manner that jeopardizes safety.

The chances of pitting, galvanic corrosion, crevice corrosion and galvanically induced hydration are also analyzed in /19-6/. The results show that these mechanisms, if they occur at all, do not entail any problem for the type of storage in question.

Delayed failure due to hydrogen embrittlement has not been observed on the zircaloy-clad fuel in connection with storage. Theoretical analyses of this process show low crack propagation that does not lead to failure.

During reactor operation, isolated fuel damages can occur due to stress corrosion cracking. This is caused by high stresses during operation under the simultaneous influence of certain substances (mainly iodine) released from the fuel. Since the stress in the cladding material is much lower during storage than during reactor operation, and the release of fission products declines greatly after discharge from the reactor, the necessary conditions for stress corrosion cracking do not exist during fuel storage.

Experimental studies of the inside of the cladding after many years of reactor operation and storage have shown the following:

-   The build-up of oxide on the inner surface was scarcely measurable (about $\mu 1$ m) after 8 years' irradiation in the Halden reactor.

-   Metallographic examination of fuel rods after 10 years' storage in England showed nothing that could be interpreted as incipient or ongoing degradation.

Damaged fuel is stored in different ways. Normally, fuel assemblies that contain leaking rods are placed in sealed water-filled containers in the pool. This method of storage has been used in WAK, Germany and Windscale, England. In Mol, Belgium, damaged fuel is stored in sealed dry containers in pools. Damaged CANDU fuel has been stored in Mol in this manner for 9 years. In General Electric's Midwest Fuel Recovery Plant in the United States, damaged fuel is stored in the same way as undamaged fuel without any problems. Spent fuel is stored in pools at the nuclear power plants without the use of special containers for damaged fuel assemblies.

At the CLAB, special containers will be used for damaged fuel starting with its transport from the nuclear power station. These containers are water-filled in the CLAB.

Small quantities of radioactive substances can be released from damaged fuel rods during storage. But continued degradation of the fuel or cladding has not been observed in connection with the storage of such rods.

Studies of long-term-stored fuel are being conducted in a number of countries. Since the publication of the report on long-term fuel storage in 1978 /19-6/, further data have been collected. They confirm and reinforce the conclusions presented above /19-7/.

19.4   ENCAPSULATION STATION

The encapsulation station consists of four main parts: A receiving section, a station for encapsulation of the fuel, a dispatch section that connects to the final repository and a station for the embedding of, among other things, BWR fuel boxes in concrete. The encapsulation station is described briefly in section 4.4 and in greater detail in /19-8/. A safety analyses of the facility is provided in /19-3, 19-4 and 19-5/.

Two procedures are described for encapsulation of the spent fuel in a copper canister: Filling with lead followed by electron beam welding of the lid and filling with copper powder followed by sintering with hot isostatic pressing of the canister.

19.4.1   The receiving section

Handling in the receiving section, which includes those parts of the facility where fuel and boxes are handled underwater, resembles in its essential respects the handling procedures in the CLAB. The important difference is that the CLAB is designed to handle fuel that is one year old, while the fuel that comes to the encapsulation station has decayed for 40 years. Many nuclides important for safety have therefore decayed completely or their activity has been

greatly reduced. The plant can therefore easily be designed to produce less environmental impact than that reported for the CLAB in these sections, during both normal operation and accidents.

19.4.2   Encapsulation in welded copper canisters

This process consists of the following main steps:

-    filling of copper canister with fuel assemblies

-    lead filling in vacuum furnace after the canister and the fuel have been heated to about $400^{\circ}C$

-    slow cooling

-    welding-on of lid by means of electron beam welding

-    inspection and decontamination of canister

The fuel is handled by remote control in different cells with concrete radiation shielding. It is supervised via radiation-shielded windows from control rooms. All areas containing radioactivity are under lower air pressure than control rooms and other areas in the plant. The ventilation air from the encapsulation cells can be passed through special filters that retain any leakage of radioactive substances.

All equipment in the encapsulation cells can be given service and maintenance by lifting it over to a special service cell.

After a cell has been emptied of fuel and decontaminated, the necessary maintenance can also be performed inside the cell.

During the lead casting step, the canister and the fuel are heated to about $400^{\circ}C$, which is well above the melting temperature of the lead ($327^{\circ}C$). This will cause the gas pressure inside the rods to increase. In PWR rods, this can lead to a considerable increase in internal pressure. However, the estimated stress lies well below the design value used for the cladding on the PWR rods (110 N/mm^2

compared to 300 N/mm^2). Oxidation of the fuel rods or the canister cannot take place, since heating and cooling are done under inert protective gas.

The fuel is handled dry throughout the encapsulation process. This handling will entail some contamination of the active areas of the plant. The dominant contribution to exposure of the personnel during normal operation is therefore obtained in connection with the service and maintenance work.

The annual dose to individuals is estimated to be lower than 2 mSv/y, on the average. The collective dose to the personnel is estimated to be on the order of 0.1 manSv/y.

The normal operating releases are very small and give rise to insignificant radiation doses.

Conceivable accidents that could result in releases that affect the environment are mainly within the category of handling accidents. The releases that can be obtained from these accidents are considerably lower than those reported for the CLAB, where much younger (more radioactive) fuel is handled. The probable maximum release is estimated at 200 GBq of krypton-85, 120 MBq of cesium-137, 80 GBq of tritium and 0.2 MBq of iodine-129. This would give rise to an environmental dose of less than 0.1 μSv.

Because all handling is done by remote control, the doses to the personnel in connection with an accident are also low. Since there is no hurry, the clean-up procedure following an accident can be planned and executed carefully and all special radiation protection measures can be adopted as needed.

19.4.3   Encapsulation including hot isostatic pressing of the copper canister

This process consists of the following main steps:

-    filling of canister with fuel assemblies

- filling of canister with copper powder

- evacuation and welding-on of airtight cover plate

- application of lid and welding-on of a second airtight lid

- hot isostatic pressing of canister (with lid, fuel and powder) at 500°C and 150 MPa

- cooling and inspection plus decontamination

The safety principles for the design of the plant are the same as those given in the preceding section.

Since pressing takes place under high external pressure (150 MPa), the problems with activity release due to the high temperature are, in general, less than in the case of the lead casting alternative. Incomplete filling of the canister with copper powder can, however, conceivably lead to such severe deformation of the canister that considerable contamination of the press furnace results. This highly unlikely event could conceivably give rise to some release to the environment, estimated at 1 300 GBq of krypton-85, 8 GBq of cesium-137, 300 GBq of tritium and 60 MBq of iodine-129. The radiation dose to individuals at a distance of 1 km for such a release is estimated at 3 μSv.

Personnel doses, normal operating releases and other accidents are judged to be equivalent to those for the alternative encapsulation process.

19.4.4   Casting of fuel boxes in cement

This handling is not expected to make any considerable contribution to the dose burden from the encapsulation station.

19.5   HANDLING AT THE FINAL REPOSITORY

The radiological safety problems involved in the handling of copper canisters in the final repository are essentially associated with direct radiation from the canisters. The surface dose rate amounts to less than 60 mSv/h of gamma radiation and 3 mSv/h of neutron radiation. The canisters must therefore be handled with radiation shielding. All equipment and all tools are designed with this in mind.

Before the canisters leave the encapsulation station, the surface activity will be removed. No dispersal of radioactivity from the waste is therefore expected to take place in shafts and tunnels in the final repository.

Conceivable handling accidents in the final repository are that the canister overturns or is dropped or that the elevator crashes. Only very severe stresses can cause such extensive damage to the canister that radioactive substances are released. The bottom of the elevator shaft has been designed to cushion a freely falling elevator cage so that the canister remains intact. Otherwise, there are no fall heights that are so great that an accident can be expected to lead to a release of radioactivity. If it is nevertheless postulated that such an accident occurs, the release of radioactivity will be considerably less than for the case described under 19.4.3 with unsuccessful pressing of a canister, since the temperature of the fuel is much lower.

19.6   SUMMARY

Safety analyses show that handling and transports of spent nuclear fuel can be carried out in a safe manner with the methods described in this report. This applies to the entire handling sequence from the nuclear power plant to the final repository.

In general, as the activity in the fuel decreases with increasing time from its discharge from the reactor, the problem of arranging a safe handling is simplified. Thus, it is possible in all steps to keep the personnel doses well below the permissible values stipu-

lated by the authorities. Releases from the normal handling procedure will be very small and will give rise to negligible environmental impact.

The probability of accidents or failures that give rise to large radioactivity releases is very small. The most difficult-to-identify case is an extensive fire of long duration on a ship transporting ten fully-loaded transport casks with spent fuel. Even in the event of this severe accident and with pessimistic assumptions concerning dispersal etc., small environmental consequences are obtained (13 μ Sv individual dose and 65 manSv collective dose commitment). All other identified or postulated accident cases in the handling chain give considerably less environmental impact.

20 THE SAFETY OF THE SEALED REPOSITORY

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┌──────────┐   ┌────────────┐   ┌─────────┐   ┌──────────┐   ┌───────────┐   ┌────────────────┐
│  SPENT   │──▶│ FACILITIES │──▶│  NEAR   │──▶│GEOSPHERE │──▶│ BIOSPHERE │──▶│   SAFETY       │
│  FUEL    │   │            │   │  FIELD  │   │          │   │           │   │                │
└──────────┘   └────────────┘   └─────────┘   └──────────┘   └───────────┘   │ ┌────────────┐ │
                                                                             │ │  STORAGE   │ │
                                                                             │ │  PERIOD    │ │
                                                                             │ └────────────┘ │
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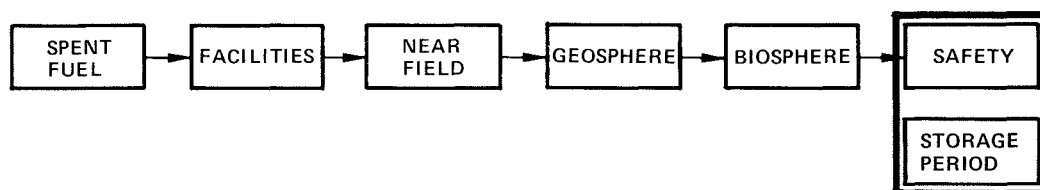
This chapter deals with the function and long-term safety of the final repository. The results consist of estimated radiological consequences for a number of postulated dispersal scenarios. Furthermore, the validity of the selected premises and the way in which variations in these premises affect the calculation results are discussed. Calculated environmental effects are compared with standards and guidelines issued by national authorities or international agencies as regards permissible doses from ionizing radiation.

An analyses of accidents and extreme events that could conceivably affect the final repository is presented in chapter 21.

20.1 GENERAL

20.1.1 Purpose

The main purpose of this report is to demonstrate that a final storage of spent nuclear fuel can be effected in Swedish bedrock and with present-day technology in a manner that meets very high demands on safety.

The high demands on safety mean that not only a probable sequence of events after sealing of the final repository must be analyzed, but also cases that show where the upper limit for environmental effects can be expected to lie. The function of the repository is examined over very long periods of time (millions of years).

The account is based on present-day knowledge. Wherever such knowledge is incomplete, this has been compensated for by the use of cautiously chosen premises.

It can be assumed that both the methods used to effect the final storage and the methods used to demonstrate the long-term safety of the repository will be successively refined and improved over the decades that remain before final storage is to be realized. It can therefore be assumed that the design that will later be chosen will be more optimal than the one presented here.

20.1.2   Outline of the safety account

The account of the safety assessments that is given in the following sections includes the following parts:

-    A description of the probable sequence of events after the repository has been sealed, section 20.2. No margins for uncertainties in evaluations of technical and geological systems over long periods of time are taken into account there.

-    An account of the premises for and calculated radiological consequences of postulated unfavourable dispersal scenarios, sections 20.3 to 20.7. These scenarios are selected to define the upper limit for the effects in the environment that might conceivably result from final storage of spent nuclear fuel.

-    An evaluation and disucssion of the safety evaluation and of the results of the calculations, section 20.8. Calculated environmental effects are compared with current standards and guidelines and with natural concentrations of radioactive substances.

20.1.3   Calculation procedure

The final respository consists of a system of barriers with two separate modes of function:

-    to contain and isolate the radioactive substances from the biosphere totally during such a long time that the greater part of the activity decays before the substances can escape from the repository

-    at a later stage, to limit and distribute in time and space any releases that could conceivably arise so that the concentrations that reach the biosphere will be acceptably low

Owing to various processes in the repository and its surroundings, the ability of the barriers to fulfil their safety-related functions in the long term may be altered. The different barriers will acquire varying importance during different stages of the final storage. The following stages have been defined in the calculation sequence.

-    Groundwater intrusion in the repository

-    Canister corrosion

-    Dissolution and dispersal of radioactive substances in the repository's near field

-    Dispersal of radioactive substances in the bedrock

-    Dispersal of radioactive substances in the biosphere and impact on man

The different stages overlap each other chronologically. This overlap becomes increasingly pronounced as more time passes from the sealing of the final repository.

Figure 20-1 shows a scheme of the different steps in the analysis of the function of the barrier system. It shows that a large quantity of input data and a number of mathematical models have been used for different parts of the calculation sequence. The data
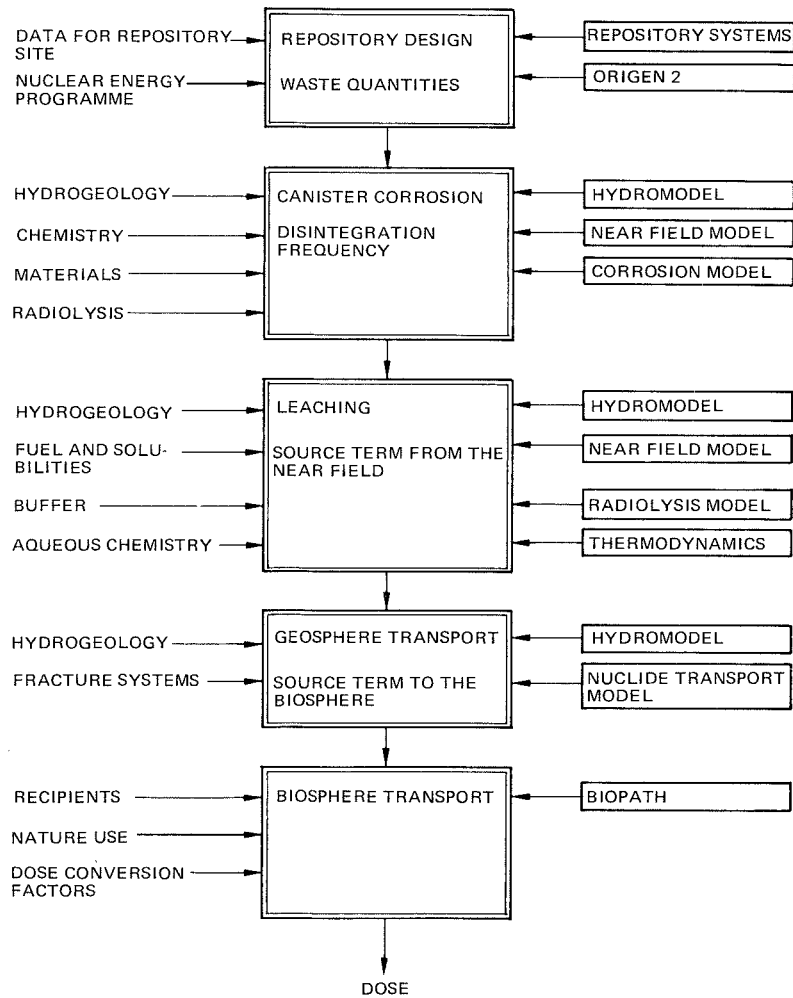
Figure 20-1. Scheme of different steps in the safety analysis.

base, the background of the models and the design of the models have been dealt with in previous chapters, mainly chapters 3, 6, 7, 10, 12-16 and 18.

20.1.4   Choice of scenarios

A release of radionuclides can only occur if the canister leaks. This could occur in three ways:

-   the canister is defective from the start due to faulty fabrication

-    the canister is damaged mechanically

-    the canister is damaged by corrosion

Mechanical damage to the canister during the handling stage is
equivalent from the viewpoint of safety to defects in fabrication.
Such damages and their consequences are analyzed below in a scenario
entitled "initial canister damage". Mechanical damage to the canis-
ter after sealing of the repository can only be caused by extensive
bedrock movements or other extreme events. Such events are discussed
in chapter 21.

Under the influence of corrosive substances that might reach the
canister surface, the canister can slowly corrode and eventually be
breached, after which the groundwater can come into contact with
the spent fuel. An analysis of the consequences this can have
constitutes the central part of the safety account.

The dispersal calculations have been carried out with different
input data to the models. The most important parameters and the
chapters where data for them are discussed are:

-    the flow of water at repository level, chapter 18

-    the solubility of the substances under oxidizing and reducing
     conditions, chapter 12

-    the sorption properties of the substances under oxidizing and
     reducing conditions, chapter 12

-    migration distance from the repository to the release point
     and the hydraulic fracture frequency of the rock along the
     transport path, chapter 18 and chapter 14

-    turnover in the biosphere and doses to human beings in connec-
     tion with exposure or intake, chapters 15 and 16

The scenarios have been selected on the basis of the following
principles:

- A central scenario has been defined. The input parameters for this scenario have been selected from the existing material in such a manner that the data represent unfavourable conditions on the study sites. However, canister life has been chosen so that it is more than ten times shorter than it is expected to be even under unfavourable assumptions.

- Other scenarios stem from the central scenario with variations of canister life, transport mechanisms and biosphere recipient. These variations will shed light on how the results are affected by significant deviations from the conditions in the central scenario.

A list of the scenarios and the most important parameters is provided in table 20-1. The background of the parameter values used is discussed in sections 20.3 to 20.7. In brief, the different cases can be characterized as follows:

A: Central scenario as described above. The biosphere recipients are assumed to be a well or a lake.

Table 20-1.   Important input data for described calculation cases.

Scenario	Canister disintegration	Water flow and migration distance	Matrix dissolution	Solubility and sorption data	Biosphere and dose data
A: Central scenario	At even pace during period 10^5 to 10^6 years	0.1 l/(m^2· year) and 100 m	360 mg/l for uranium	According to section 12.9, reducing conditions	According to chapters 15 and 16, well or lake recipient
B: Initial canister damage	1 canister leaky after 100 years	A	A	A	A
C: Oxidizing conditions	A	A	A	According to section 12.9, oxidizing conditions	A
D: Colloid transport	A	A	A	Irreversible sorption on particles (concentration 0.5 mg/l) which are transported without retardation	A
E: Peat bog recipient	A	A	A	A	Peat bog recipient

"A" means that the same input data as in scenario A have been used.

B: Initial canister damage, i.e. the barrier function of the canister is assumed to be lost 60 years after the repository

has been sealed. Otherwise, the parameters of the central scenario are used, corrected for temperature effects.

C: Oxidizing conditions prevail all along the transport path from the repository to the biosphere. Otherwise, the parameters of the central scenario are used.

D: A certain portion of the radionuclides are transported in the form of colloids without retardation in the rock. The parameters of the central scenario are used for solubilities and water flows.

E: The biosphere recipient consists of a peat bog, which is subsequently utilized as soil conditioner. Otherwise, the parameters of the central scenario are used.

Moreover, calculations have been carried out for several additional scenarios and combinations of different scenarios. A full account of all scenarios evaluated is given in /20-3/ and /20-6/. Certain results from these other calculations are also touched upon briefly in sections 20.5 to 20.7.

20.1.5   Safety margins

Section 20.2 below describes the probable sequence of events after the repository has been sealed. In the scenarios described after that, the different steps in the analysis incorporate safety margins, compared to the probable sequence of events. The safety margins can be of two fundamentally different kinds:

-    The assumed value of the parameter has a margin to the expected value. An example of such a case is the assumptions of water-flow at the repository level, which are 1 $1/(m^2 \cdot year)$ in the corrosion calculations and 0.1 $1/(m^2 \cdot year)$ in the dispersal calculations. Calculated values, based on measurement of the hydraulic conductivity of the rock, usually lie within the interval 0.002-0.06 $1/(m^2 \cdot year)$.

- Simplified assumptions are made and favourable processes or mechanisms are neglected. Such assumptions are justified where the available body of data is insufficient or models that provide a satisfactory quantitative description of a process are lacking. An example is the description of the release of radioactive substances from the fuel following a canister penetration. In the calculations, it has been assumed that the canister with its fill loses all of its function when the first point penetration occurs. The entire quantity of fuel is then assumed to be rendered accessible to leaching by the groundwater. In reality, only a limited area of the canister will be corroded through and only a small portion of the fuel content will come into contact with the groundwater.

A more thorough examination of the safety margins in the assumptions that are made in connection with the calculations is provided in section 20.8.

20.2   PROBABLE SEQUENCE OF EVENTS

20.2.1   Canister life

The investigations carried out by KBS within the study sites indicate the existence of continuous masses of impervious rock of sizes ranging up to several kilometres. The calculated groundwater flow at a depth of about 500 m in these rock masses lies between 0.002 and 0.06 $1/(m^2 \cdot year)$.

The deep groundwater at these sites has been found to contain low concentrations of substances that can attack copper (oxygen and sulphide).

If largely unchanged hydrological and geochemical conditions are assumed around the final repository in the future and the same cautious method of calculation is applied as that used by the Corrosion Institute's expert group /20-1/, a service life on the order of 100 million years is obtained for a copper canister with a wall thickness of 10 cm. Forecasts over such long periods of times are associated with uncertainty. Nevertheless, it is probable that

most of the canisters will remain intact for several million years and that they will be successively penetrated during a period of several million years.

20.2.2 Quantity of radioactive substances

After an isolation period of several million years, the radionuclide composition of the spent fuel will be roughly the same as that of natural uranium with daughter nuclides. The differences that persist are that small quantities of iodine-129 may be left and that the concentration of uranium-235 is about twice as high. This is of no importance for the dose compared to the natural isotopic composition.

20.2.3 Dissolution of fuel

The Swedish crystalline basement rock contains, on the average, about 5 mg of uranium per kg. This means that the rock cover over a repository contains a uranium quantity of the same order of magnitude as the uranium in the repository.

The concentration of dissolved uranium in deep Swedish groundwaters lies in the range 0.01-10 microgrammes per litre. The ratio between the isotopes uranium-234 and uranium-238 indicates that these concentrations correspond to the saturation concentration. Outside the near field, where natural conditions in the rock may be disturbed by the repository, the conditions for dissolution of uranium in the ground water are independent of whether the uranium comes from the fuel or is a natural deposit.

If all water that flows through the repository $(0.1 \ 1/(m^2 \cdot year))$ should become saturated with uranium from the deposited fuel (10 microgrammes per litre), approximately 500 mg of uranium would be transported away from the repository every year, assuming a repository area of 0.5 km^2. The equivalent dissolution time for all uranium in the repository will then be billions of years.

This agrees with the fact that the uranium in the bedrock has not been dissolved during the billions of years that have passed since the bedrock was formed.

20.2.4 Conclusions and discussion

The long-term processes that are judged to be most probable are thus the following:

- The copper canister will completely isolate the spent fuel from the envrionment for several million years. The composition of the fuel will then be close to that of natural deposits of uranium and uranium daughters in the Swedish crystalline basement.

- The conditions that determine the maximum concentration of naturally occurring uranium and uranium daughters in Swedish groundwater will also limit the concentration of the same substances from the deposited fuel. This means that dispersal from the repository to the undisturbed rock will take billions of years and the biosphere will not be affected.

The scenario described in this section is judged to be the probable one. Nevertheless, forecasts for such long periods of time always involve uncertainties. A number of scenarios based on more unfavourable assumptions are therefore described in the following.

The purpose of these calculations is to set an upper limit for conceivable consequences of the final disposal scheme and to show that these maximum consequences

- are well within the limits issued within the field of radiation protection

- do not essentially alter the radiological conditions in the repository's environment.

20.3   GROUNDWATER FLOW IN ROCK AND BUFFER

20.3.1   General

The chain of events that can lead to an exposure of human beings to radioactive substances from the repository includes canister degradation, dissolution of the fuel, transport through the geosphere and dispersal in the biosphere.

All processes before release to the biosphere are dependent to a high degree on the water flow rate and flow paths in the rock. The possibility that dissolved substances in the groundwater will affect the canister and the fuel is limited by the bentonite buffer that surrounds the canister.

Groundwater flow is dealt with below under the headings:

- Water flow in rock

- Function of bentonite buffer

- Summary

20.3.2   Waterflow in rock

On the basis of the geological and hydrological data obtained from the site investigations, three-dimensional model calculations of the flow of water in the rock have been carried out. A detailed account of these calculations is provided in /20-2/ and the results have been summarized in chapter 18.

The water flows in the rock are determined by

- The topography of the site

- The hydraulic conductivity of the rock and the fracture zones in different parts

-     The geometry of the fracture zones

The model calculations provide information on the flow of water in the different parts of the studied site as well as on the flow paths.

Model calculations of the water flow have been carried out for four sites.

Calculated groundwater flows at repository depth within the parts of the rock being considered for location of a final repository are tabulated in table 20-2, cf. chapter 18.

Table 20-2.   Calculated waterflows at repository depth at investigated sites.

Site	Range $1/(m^2 \cdot year)$	Typical value over repository $1/(m^2 \cdot year)$
Kamlunge	0.003-0.06	0.02
Fjällveden	0.002-0.05	0.03
Gideå	0.004-0.02	0.01
Finnsjön	0.2 -1	0.7

The water flow rate is very low in all cases. It is considerably higher within the Finnsjö area than within other areas, however. The Finnsjö area has been included to illustrate the conditions in an area with a relatively high groundwater flow. The highest calculated flow, $1 \; 1/(m^2 \cdot year)$, has also been used in the calculations of the rate of canister corrosion.

The highest calculated flows arise in pronounced fracture zones or under steep terrain sections. The final repository has therefore been located with a certain distance to the nearest fracture zone. This distance is about 100 m.

Influence of the fuel's heat flux

The water flow is affected by the hydraulic gradient. The rock in and around a final repository is subject to heating. The difference

in temperature in relation to the surroundings then gives rise to an upward thermal gradient in the repository. Calculations with a typical repository geometry and heat flux show only a marginal influence on the water's travel time from the repository to the ground surface, see section 6.4.2. The effect of heat on the ground-water flow is only of interest for the initial canister damage case. At a later time, when the canisters may be penetrated by corrosion, the temperature distribution in the repository area has returned to the natural one, see section 4.5.4.

Influence of glaciations

Since the Swedish bedrock has undergone numerous glaciation cycles, one additional glaciation should not involve any appreciable changes in the hydraulic conductivity of the deep bedrock.

The aggregate erosion caused by the ice sheets in most of Sweden is limited to a few tens of metres.

During a glaciation, the hydraulic gradient and thereby the ground-water flow in the bedrock will decrease. During the deglaciation period, however, elevated gradients and groundwater flows of rela-tively short duration might conceivably occur at the ice front. In all, a glaciation is expected to have only a marginal influence on the function of the final repository. See section 8.9.

20.3.3   Function of the bentonite buffer

The space between the copper canister and the wall of the deposition hole is filled with compacted bentonite clay. The geometric confi-guration of the deposition hole and the material properties of the compacted bentonite are described in chapter 9.

Since the rock nearest the deposition hole and the tunnel may have been affected by blasting and drilling, an elevated hydraulic conductivity and fracture frequency may exist locally around the buffer zone. This means that the groundwater's flow lines may be pulled in towards and densified around the deposition hole to some extent. However, the bentonite constitutes an impervious cylinder
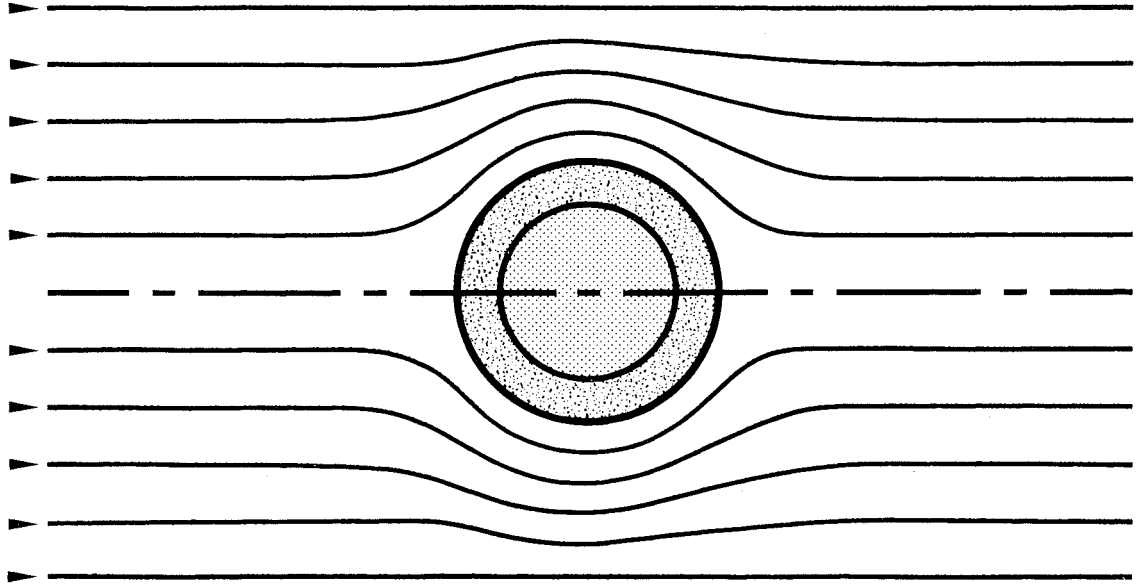
Figure 20-2. Stream lines for groundwater flow around a deposition hole.

in the rock mass and prevents flow through the deposition hole, which causes the stream lines to be pushed out from the hole. The shape of the stream lines is shown in figure 20-2.

The buffer's low hydraulic conductivity prevents groundwater flow between the canister and the rock. All transport of water-soluble substances through the bentonite buffer therefore takes place through diffusion. This limits the exchange of material between the canister and the groundwater, which retards canister corrosion and later fuel dissolution as well. The following factors are decisive in determining the transport velocity through the buffer:

- the geometric configuration of the buffer layer (chapter 9)

- the diffusion coefficients for the corrosive substances or radionuclides in question (chapter 9 and 13)

- the concentration gradient for the substances in question over the buffer layer

- the transport resistance in the groundwater's concentration boundary layer nearest the bentonite surface (section 13.2.2)

Data and results

The following data have been used in calculating the quantity of water that interacts with the canister via the buffer layer.

Geometric data	See chapter 9
Density of saturated bentonite	2.1 t/m^3
Diffusion coefficients for steady-state conditions in	
water	2×10^{-9} m^2/s
bentonite	4×10^{-11} m^2/s
Spacing between water-conducting fractures in deposition hole	1 m
Fracture width, average	0.1 mm
No penetration of bentonite into the fractures is assumed to take place	

For a given groundwater flow U_0 1/(m^2 · year), it is possible to calculated an equivalent groundwater flow Q_{eq} 1/(canister · year), which represents the volume of water with which a chemical exchange from or to a canister can take place via the buffer during one year. See section 13.2.2.

Table 20-3 gives related values of U_0 and Q_{eq} under the conditions stipulated here.

Table 20-3. Q_{eq} at different groundwater flows in the rock.

Groundwater flow in the rock U_0 1/(m^2 · year)	Equivalent flow per canister Q_{eq} 1/(canister · year)
0.01	0.19
0.03	0.32
0.1	0.57
0.3	0.94
1	1.57
3	2.41

During the initial stage for outward transport of radionuclides through the buffer, a concentration profile will be built up. This

takes place at different speeds for different substances, owing to the fact that they are retarded to different degrees by sorption in the buffer material. The time to obtain a penetration of radionuclides on the outside of the buffer (here defined as the point in time when 5% of the concentration on the inside has been reached) has been calculated in section 13.2.1. The penetration time varies between 10 and 500 years for fission products with limited retardation and amounts to several tens of thousands of years for highly retarded actinides. The time for nuclide transport through the buffer is of importance above all in the case of initial canister damage.

Discussion

Most of the transport resistance through the bentonite buffer is attributable to the concentration boundary layer in the nearly stationary water nearest the bentonite surface. As long as no water flow takes place through the buffer, the inward transport of corrosive products and outward transport of radionuclides through the buffer are relatively insensitive to the thickness and diffusion constant of the bentonite.

If the buffer material should penetrate into the rock fractures that lead into the deposition hole, a considerably higher transport resistance will be obtained in these fractures.

An important question is whether the bentonite buffer can be expected to retain its properties over a very long period of time. This has been discussed in section 9.2.4. At temperatures below $100^{\circ}C$, the predominant proportion of the smectite mineral mass will remain chemically intact for more than one million years.

The maximum temperture in the buffer is determined by, among other things, the canister's residual heat output and the thermal conductivity of the highly-compacted bentonite. As was shown in chapter 4, the quantity of fuel in the canisters is limited so that the residual heat output 40 years after discharge from the reactor does not exceed 850 W/canister. This means that the maximum temperature in the buffer will be around $80^{\circ}C$.

A prerequisite for the mechanical stability of the buffer is that large quantities of bentonite do not penetrate out into the fracture system in the rock. In fractures that open out into a deposition hole (fracture width less than 0.5 mm), the obstacle to water transport offered by the bentonite that fills the fracture and friction against the fracture walls will cause the penetration to cease after a few decimetres. The outer part of the intruding bentonite forms a gel that is sufficiently stable in the ground-waters in question to make the loss of clay particles through groundwater erosion negligibly small, see section 9.2.6.

The bentonite's capacity to bear the canister is a function of the density of the material. Even at a bulk density of about 1.5 t/m^3, the bearing capacity of the bentonite is still sufficient to prevent the canister from sinking through the bentonite, even over a very long period of time, see section 9.2.6.

20.3.4   Summary

The groundwater flow in the rock mass around a suitably located repository has been estimated to lie within the range 0.002-0.06 $l/(m^2 \cdot year)$. A flow of 0.1 $l/(m^2 \cdot year)$ is then a pessimistic value for the worst located canisters in all areas studied except Finnsjön. For a site with Finnsjön's characteristics, flows of up to about 1 $l/(m^2 \cdot year)$ can be expected.

The transport of corrosive substances from the groundwater to the canister (and radionuclides from the canister to the groundwater if the canister is penetrated) is limited by the buffer material in the deposition hole. At a groundwater flow of 0.1 $l/(m^2 \cdot y)$, an "equivalent" water volume of 0.6 l/y will interact chemically with the canister. At a flow of 1 $l/(m^2 \cdot year)$, the corresponding equivalent water volume will be 1.6 l/y.

20.4   CANISTER LIFE

20.4.1   Premises

The substances that dominate corrosion of the copper canisters are oxygen and sulphides, which are present in tunnels and deposition holes and are also brought in with the groundwater in dissolved form. In addition, some quantities of oxidants can be generated by the radiolysis of water nearest the canister surface.

Chapter 10 presents an evaluation of the minimum life of copper canisters of different wall thicknesses in the repository environment in question. The quantity of copper that has been oxidized at different times has been calculated, and the minimum time for corrosion penetration of canisters of different thicknesses has been given.

In determining the corrosion rate, use has been made of the concentrations of corrosive substances in the groundwater given in table 10-1, a water flow in the rock of 1 $l/(m^2 \cdot year)$, corresponding to an equivalent water volume of 1.6 l of water interacting with the canister every year, and radiolysis products according to /20-1/.

The evaluations are further based on the following assumptions:

- corrosion caused by sulphide, oxygen and radiolysis products is assumed to be able to proceed without mutual interaction

- the transport resistance that arises as a result of the volume increase stemming from the formation of the low-solubility corrosion products is neglected

- no barrier function is attributed to either the copper fill or the lead fill in the canister or the zircaloy material in the fuel rod cladding.

It is further assumed that all corrosive substances from the tunnel will concentrate their action on the upper 10% of the canister surface and that corrosion will take place irregularly with a

pitting factor of 5 in a probable case and 25 in an unfavourable case.

20.4.2   Results

The corrosion assessments have resulted in the following conclusions:

-   The corrosive substances that can affect the copper canisters in the final repository are oxygen and sulphides. Sulphates are judged to react with copper so extremely slowly that their action is negligible even over geological periods of time. (The supply of sulphates to the canister surface can, moreover, be effectively limited by plugging the storage holes with a granite plug.)

-   The maximum pit depth at different times and wall thicknesses of the canisters are given in table 20-4.

Table 20-4.   Maximum pit depth in mm at different times and for different canister thicknesses.

Exposure time (years)	10^3		10^4		10^5		10^6	
Pitting factor	5	25	5	25	5	25	5	25
Canister thickness:								
10 mm	0.6	3.0	0.7	3.3	1.1	5.8	5.5	penetrated
60 mm	0.6	3.0	0.6	3.0	0.9	4.5	3.6	18
100 mm	0.6	3.0	0.6	3.0	0.9	4.5	3.5	17
200 mm	0.6	3.0	0.6	3.0	0.9	4.5	3.4	17

As can be seen, even with the improbably high pitting factor of 25, a canister with a wall thickness of a few centimetres in the environment in question gives isolation times for the fuel on the order of one million years.

To simplify the calculations in the further analysis of the safety of the system, the very unfavourable assumption is made that the first penetration of a canister caused by corrosion occurs after 100 000 years. It is further assumed that variations in groundwater flow, in the supply of corrosive substances and in the pitting factor will cause the penetrations of all canisters in a repository to be distributed over a period of one million years.

As is evident from chapter 4, this account applies to the final storage of copper canisters with a wall thickness of 100 mm. Reported calculations of canister life show that a considerably smaller wall thickness should be adequate.

20.4.3   Initial canister damage

The possibility cannot be entirely ruled out that a fabrication defect that is not detected by the quality control procedures may result in a canister that is not completely leakproof being deposited in the repository. In order to shed light on the effect of such an event, the consequence of an initial canister damage is also dealt with in the further analysis. It is thereby assumed that dissolution of the fuel starts as soon as the groundwater has established a transport path out of the repository. The earliest point in time at which this can happen has been set at 60 years after deposition, i.e. 100 years after the fuel's discharge from the reactor.

20.4.4   Summary

In the calculations, it is assumed that the first canister penetration occurs 100 000 years after deposition. In view of the random variation in corrosion of the canisters, it has been assumed that the penetrations will be evenly distributed over the period from 10^5 to 10^6 years, i.e. an average of one corrosion-penetrated canister per 200 years.

The consequences of an assumed initial canister damage have then been calculated by assuming that all fuel in the canister will

become accessible for leaching 100 years after discharge from the reactor.

20.5   RELEASE OF RADIOACTIVE SUBSTANCES

20.5.1   General

The copper canister prevents contact between the groundwater and the fuel for a long period of time. In the calculations presented, it is assumed that groundwater contact with all fuel in the canister occurs when the canister has been penetrated. A slow dissolution of the various substances in the fuel can then begin. The rate of dissolution is determined by, among other things, the solubility of the different substances under the chemical conditions defined by groundwater chemistry, the chemistry in the deposition holes and radiolysis caused by, above all, alpha radiation. Radiolysis necessitates the assumption that oxidizing conditions will exist within a local area around the deposition holes, known as the near field, in contrast to the surrounding bedrock, where the environment is reducing.

The transport of radioactive substances through the buffer layer is determined by the geometry of the deposition holes and the diffusion velocity of the substances both in the buffer material and in the water in the fractures in the near field.

When the radioactive substances have been transported out to the front where natural chemical conditions are restored, i.e. reducing conditions, certain nuclides will be precipitated from the groundwater.

The transport process for the radionuclides from fuel to geosphere is dealt with below under the heading:

-    fuel dissolution

-   mass transport through the buffer

-   nuclide release at the redox front

20.5.2   Fuel dissolution

Premises

The water that penetrates into the canister is assumed to form a layer over all pellet surfaces of such thickness that alpha radiation is absorbed. The volume in the corrosion-penetrated canister that becomes accessible to intruding water will in actuality be greatly limited, since the corrosion products have a greater volume than the corresponding quantity of canister material. The above assumption therefore leads to a considerable overestimate of the effects of radiolysis.

Radiolysis causes the formation of reducing and oxidizing substances such as hydrogen and oxygen as well as hydrogen peroxide. The greater mobility and lower reactivity of the hydrogen causes it to diffuse out from the near field, leaving an oxidizing zone around the fuel that is slowly widened. The outward velocity of the zone boundary is determined by the radiation level in the fuel at the time of canister penetration, the diffusivity of the radiolysis products and their rate of recombination, as well as the presence of accessible redox-buffering substances in the bentonite and the rock matrix.

The dissolution rate of the fuel has been determined on the basis of the following assumptions (see chapter 13):

-   Most substances in the spent fuel are incorporated in the crystal structure of the uranium fuel and are liberated as the uranium dioxide matrix is dissolved. Exceptions are noble gases, iodine and cesium that have diffused out to the surfaces of the pellets during reactor operation - 100% of the noble gases, 10% of the iodine and cesium.

- Liberated substances are dissolved in the water nearest the pellet surface and transported further through diffusion. The maximum concentration to which different substances can be dissolved in the water is controlled by the solubility of either the uranium matrix or the substance.

- Radiolysis caused primarily by alpha radiation creates an oxidizing envrionment around the fuel.

- The groundwater flow rate for the scenarios is given in section 20.3 and table 20-1.

- The solubility of the different substances is given in section 12.9 (oxidizing conditions).

- An average canister contains 1.4 tonnes of uranium and other nuclides in the proportion given in tables 3-5 and 3-6 in section 3.4.2.

- For the case with initial canister damage, the inventory of radionuclides is assumed to have decayed in 100 years.

- For other cases, the time to the first canister penetration is 100 000 years and the penetrations are then evenly distributed up to the time one million years.

Discussion

The solubility of different nuclides in the environment in question is higher or lower than that of uranium (see chapter 12). For substances with lower solubility (thorium and plutonium), even if they are released at the same rate as the uranium matrix is dissolved, they will be partially precipitated on surrounding surfaces. The maximum concentration gradients that control the outward transport of these substances will therefore be determined by their own solubility. The outward transport of nuclides with a higher solubility than uranium is limited by the matrix dissolution, provided that they are evenly distributed in the uranium dioxide matrix. This applies to, among other elements, neptunium, technetium, strontium and most iodine and cesium. However, 10% of the iodine

and cesium is assumed to be easily accessible in the gap between the pellets and the zircaloy cladding (see section 11.3.1). Every penetration of a copper canister is therefore assumed to produce a pulse of these substances. The pulse release will be distributed in the total water volume present in the clay buffer. From there, the substances are carried away at a rate determined by the flow of water around the deposition hole. In the initial canister damage scenario, this is estimated to give rise to a dispersion of the pulse releases over a period of about 2 000 years. In other scenarios, the distribution of the canister penetrations in time leads to an equalization of the release of iodine and cesium from the entire repository. The effect of pulse releases of iodine and cesium is dealt with in greater detail in /20-3/.

Carbonate can form complexes with the uranium. The carbonate content of the groundwater under oxidizing conditions is therefore important for the solubility of the uranium. In the calculations, this has been based on an assumed carbonate content of 275 mg per litre of water, which is the upper limit of the expected interval according to table 7-4. Lower carbonate concentrations give longer times for matrix dissolution.

Variations within expected intervals of other aqueous chemical parameters have relatively little effect on the dissolution of the fuel.

20.5.3   Mass transport through the buffer

Premises and results

The transport of dissolved nuclides from the fuel to the mobile groundwater takes place through diffusion. The velocity is dependent on the geometric configuration, the diffusion constants of the nuclides and the concentration gradient. The flow of water through the buffer is negligible.

Table 13-1 shows the times it takes for the concentration on the outside of the buffer to reach 5% of the concentration on the inside. If in the case of initial canister damage, these times are

compared with the half life of the corresponding nuclide, it is found that the nuclide americium-241 decays completely before it passes through the buffer. Furthermore, a reduction is obtained of the releases to the geosphere of cesium-137, plutonium-240 and americium-243 to less than 1% and of strontium-90 and plutonium-239 to about 5% of the values that would have been obtained without retardation in the buffer. For other important nuclides, this retardation is of no importance. Retardation in the buffer has not been taken into account in the calculations.

Discussion

After a canister penetration, it is assumed that neither the canister, the fill nor the zircaloy cladding around the fuel pellets constitutes a transport obstacle for the outward diffusion of radionuclides. Nor is the fact that the canister material or other material in the deposition holes can consume the oxidants formed through radiolysis taken into account. Both of these phenomona, which limit the outflow of nuclides in reality, have been neglected due to difficulties in quantifying the effect.

During the first one or two of thousand years after sealing of the repository, the temperature in the repository area is elevated by up to $60^{\circ}C$. This is only of importance for scenario B, initial canister damage.

At higher temperatures, the diffusion constants for the nuclides are greater and the water flow in the rock can be somewhat elevated due to the thermal gradient. The transport resistances through the buffer thereby decrease and the time it takes to establish an equilibrium flow from an initially damaged canister will be correspondingly shorter. This effect is primarily of interest for the nuclides in table 13-1 that have travel times through the buffer of up to 1 000 years and simultaneously moderate half-lives. The changes caused by elevated temperature are less than those that reflect the reported variation of measured values of the diffusion constant (table 13-1, column 2). The same applies to possible minor variations in the density of the buffer material.

20.5.4   Nuclide release from the redox front

Premises

When the groundwater and substances dissolved in it come into contact with the iron(II)-containing rock, the redox-sensitive substances technetium, uranium, neptunium and plutonium are reduced. Solubility conditions are changed and precipitations can take place.

The nuclides that do not precipitate are transported with the groundwater flow at concentrations equal to the initial concentration, but this concentration gradually decreases as a result of admixture of water from the sides. The nuclides that are precipitated will be transported further at the lower concentrations defined by their solubility in the normal reducing environment in the rock. The extent of the redox front and the solubility of the nuclides determine the total outward transport of nuclides from the near field. The solubility data given in section 12.9 have been used in the calculations.

The extent of the redox front determines the water volume that is available for nuclide transport. A projected horizontal surface area of 100-150 m^2 per canister, equivalent to the total repository surface area divided by the number of canisters in a two- or single-storey repository, respectively, gives the largest quantity of water that can be available per canister.

Calculations based on the available quantity of oxidizing substances and the redox-buffer capacity of the rock show that only in very extreme situations will the redox front correspond to an area in excess of a few tens of m^2 per canister. Nevertheless, it is assumed in the further calculations that the oxidized area covers the entire repository area. This is equivalent to about 150 m^2 per canister and a water flow in the rock of 15 litres per year and canister.

The solubility of the uranium in reducing groundwaters is given in section 12.9 at 10 μg/l. Recently completed laboratory measurements and theoretical analyses give a solubility of about 7 μg/l at

equilibrium with UO_2 in solid phase. Measurements of natural ground-
waters in reducing environments point towards a saturation level of
around 10 µg/l or lower. See section 12.5.

Trivalent and tetravalent actinides at low concentrations will
probably be precipitated together with uranium at the redox front.
This affects the release of thorium, neptunium and plutonium to the
far field. See further section 13-4. However, this coprecipitation
mechanism has not been taken into account in the further analysis,
since it has only been demonstrated experimentally for plutonium.

The total releases of different substances are influenced by, among
other things, their decay due to radioactive disintegration and the
new daughter products created. This is illustrated by table 20-5.

Table 20-5.   Quantity of certain specified elements per canister
               at different times after discharge from the reactor.

Element	Quantity g/canister at time		
	100 years	100 000 years	1 million years
Technetium	1 200	880	50
Thorium	0.1	110	270
Uranium	1 340 000	1 350 000	1 350 000
Neptunium	1 000	3 000	2 200
Plutonium	10 000	1 300	180
Americium	2 400	0.02	0

Resulting releases to the geosphere

The releases of radioactive substances from the near field to the
geophere in the different scenarios according to table 20-1 have
been calculated in /20-3/ and are reported here in figures 20-3 to
20-6. Only a selection of the more important nuclides are included
in the figures.

Figure 20-3 shows the source term for release to the geosphere in
scenario A (the central scenario). The time for matrix dissolution
here is about seven million years, which determines the release
time for matrix-bound iodine-129 and cesium-135. Uranium, neptunium
and technetium precipitate at the redox front, which, especially in
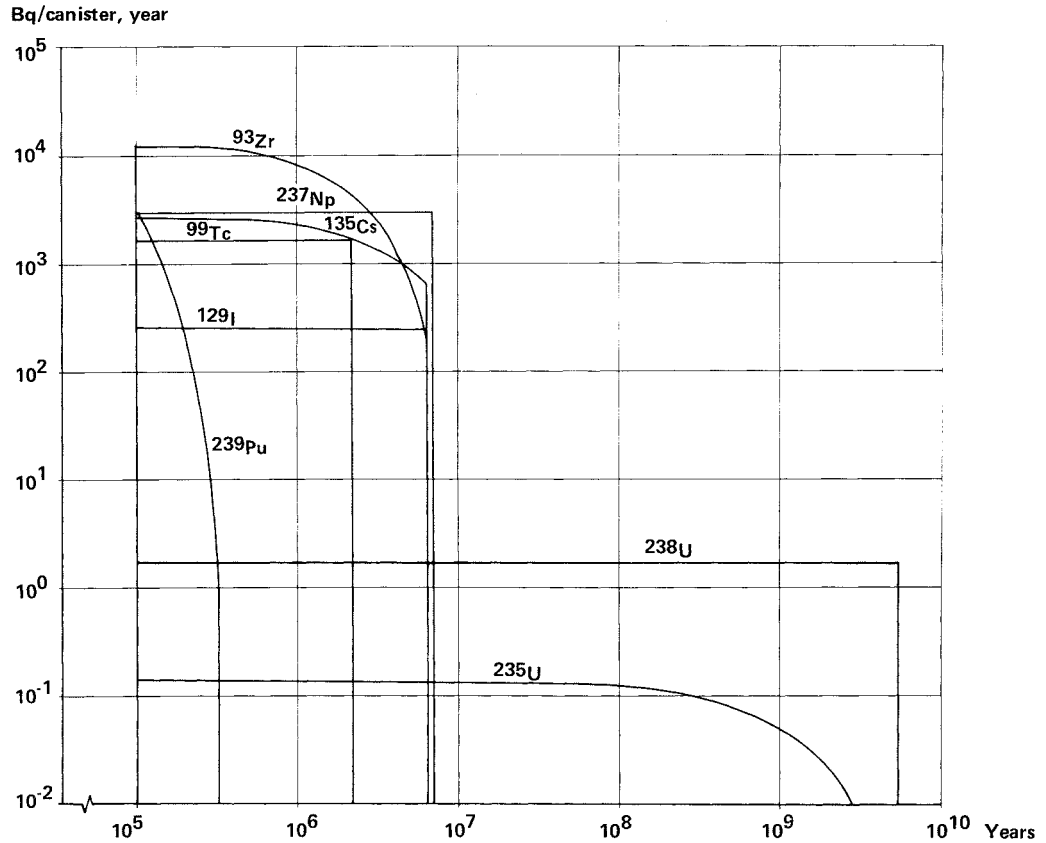
Bq/canister, year

Figure 20-3. Releases to the geosphere in Bq/canister and year as a function of time after fuel discharge. Scenario A (Central scenario).

the case of uranium, leads to an extremely slow release to the geosphere.

90% of the zirconium-93 exists as a fission product in the fuel. 5% is present in the metal components in the copper canister and 5% in the metal component repository, see chapter 3.

Figure 20-4 shows the source term for Scenario B (initial canister damage). The greatest difference in relation to scenario A is that the releases start already after 100 years and that a number of nuclides with shorter half-lives are therefore also included. Nuclides that have accumulated in the cladding gaps in the fuel rods will be released at an initially higher rate.

x = maximum level for pulse release of 135Cs

Bq/canister, year

(x) = maximum level for pulse release of 129I
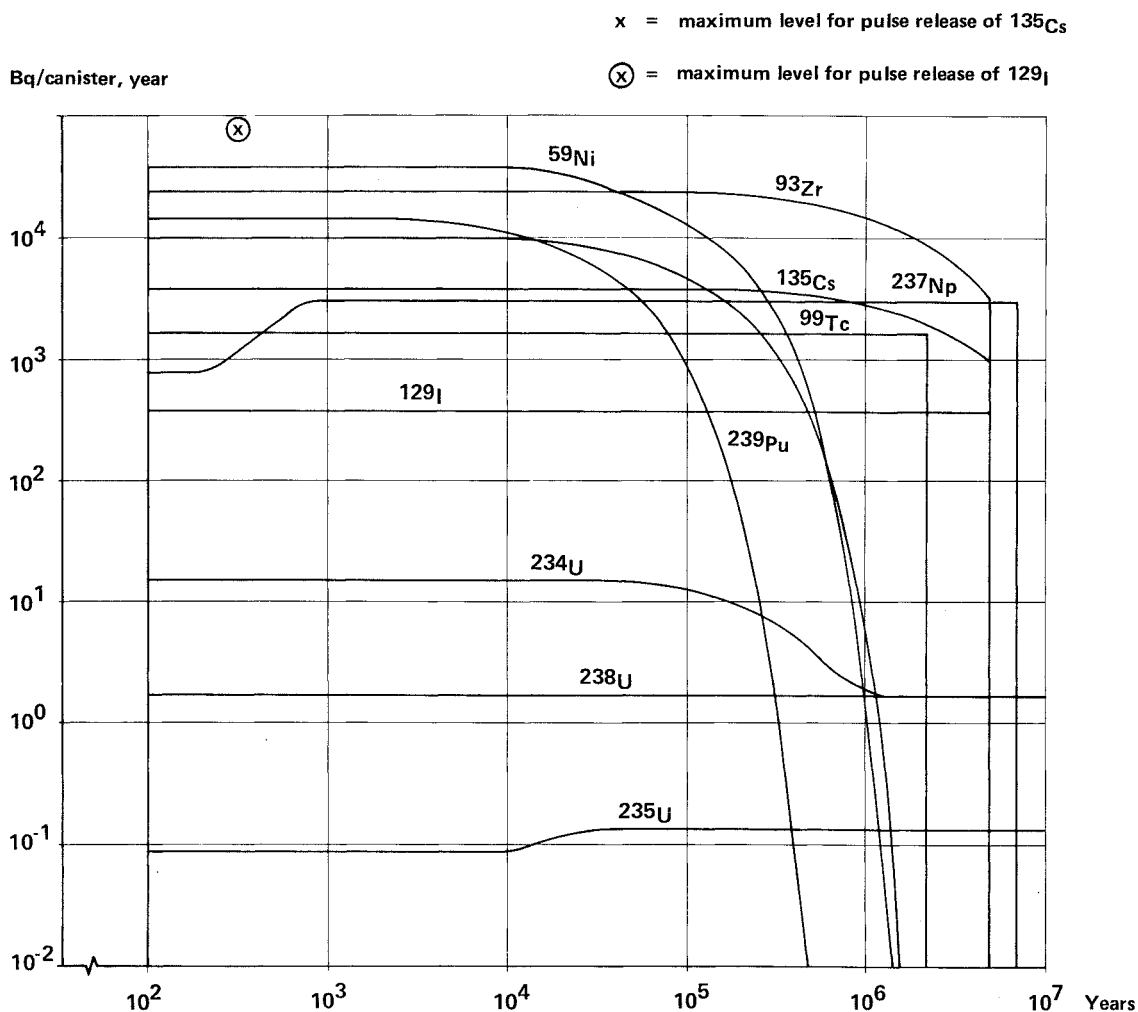
Figure 20-4. Release to the geosphere in Bq/canister and year as a function of time after fuel discharge. Scenario B (Initial canister damage).

Figure 20-5 shows the source term for scenario C (oxidizing conditions). The rate of matrix dissolution is the same as for scenario A. However, since uranium, neptunium and technetium are not reduced at the redox front, the source term for uranium increases by about 10^3, for technetium by about 10^2 and for neptunium by about 10 compared to scenario A. Figure 20-6 shows the source term in scenario D (colloid transport). It has been assumed in the calculation that the particle concentration in the water is 0.5 mg/l and that the various substances are bound irreversibly to these particles in proportion to the distribution coefficient and the concentration in solution. The calculations are carried out for each nuclide for the fraction of the concentrations in oxidizing or reducing environment that gives the highest proportion of colloidally bound activity /20-3/. Since the colloids are not sorbed during their transport through the rock, the source term curve at the deposition holes largely coincides with the release curve to the biosphere in this case.
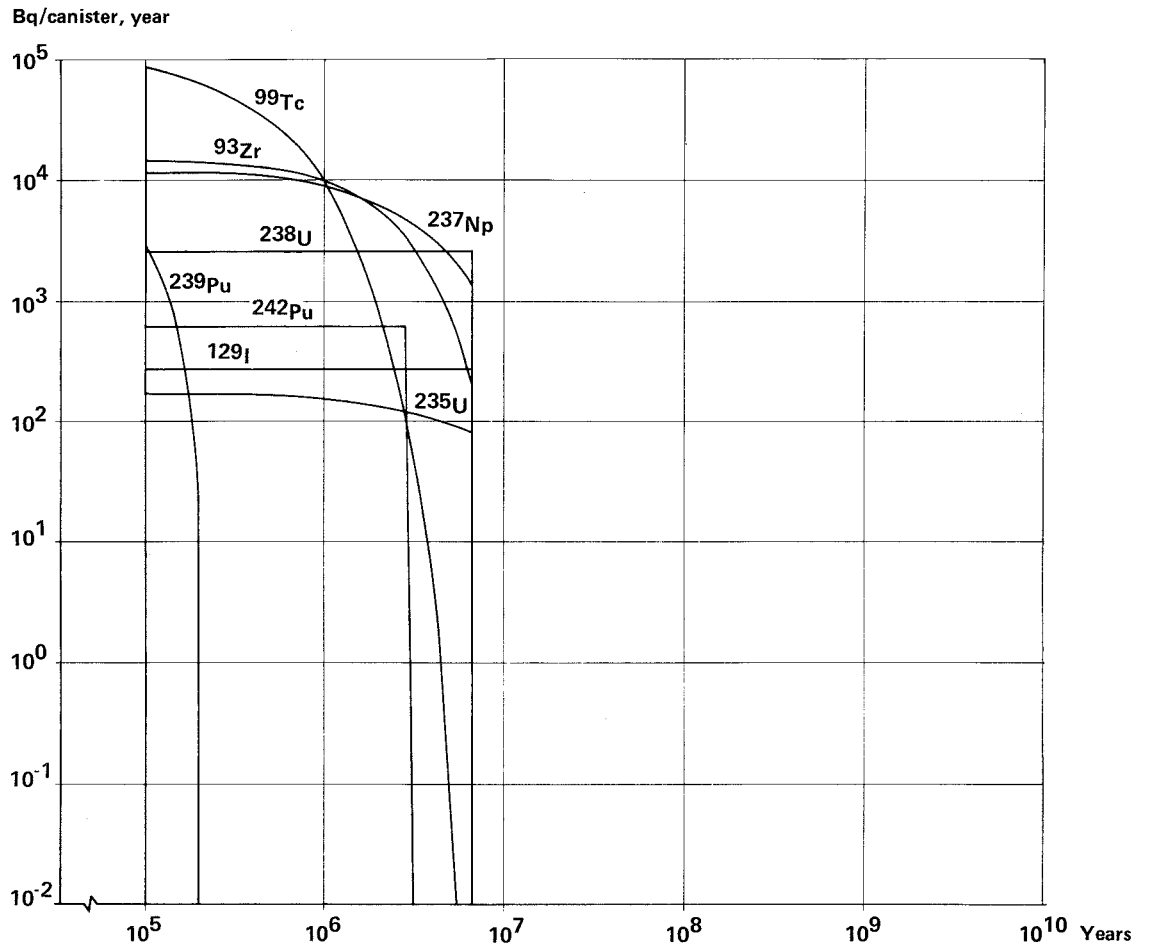
Bq/canister, year

Figure 20-5. Release to the geosphere in Bq/canister and year as a function of time after fuel discharge. Scenario C (Oxidizing conditions).

Discussion

The releases to the geosphere are determined by one of the following factors, cf. section 13.4.

1   Rate of dissolution of the uranium dioxide matrix (M)
2   Solubility in the near zone (oxidizing conditions) (N)
3   Solubility beyond the redox front (G)

Table 20-6 shows for the most important substances which of these factors is limiting in the different scenarios. As mentioned previously, coprecipitation between trivalent and tetravalent actinides has not been taken into account.
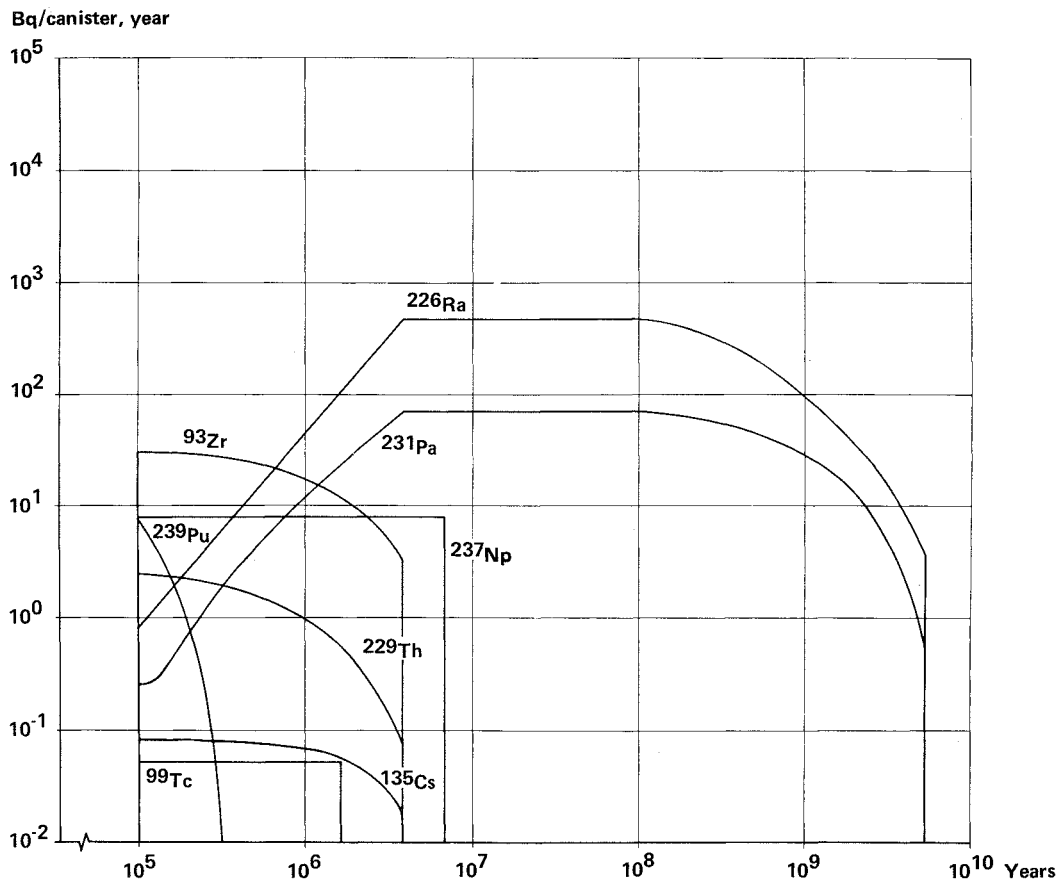
Bq/canister, year

Figure 20-6. Release to the geosphere and biosphere in Bq/canister and year as a function of time after fuel discharge. Scenario D (Colloidal transport).

Table 20-6.   Limiting factors for dispersal to the geosphere of different substances. A, B, C etc. are the different scenarios according to table 20-1.
M = Matrix dissolution
N = Solubility in near zone
G = Solubility in geosphere

Element	Limiting factor in scenario case	
	A, B, E	C
Iodine*	M	M
Cesium*	M	M
Technetium	G	M
Thorium	N	N
Uranium	G	M
Neptunium	G	M
Plutonium	N	N

*   Does not apply to easily accessible gap activity   10% of the total quantity.

20.5.5   Nuclide release from the repository for metal components

The content of radionuclides in boxes and boron glass rods is reported in Section 3.4. The design of the metal component reposi- tory is such that the chemical environment will be dominated by the large quantities of concrete cast around the waste. The water flowing through the repository will assume an elevated pH, limiting the solubility of zircaloy and steel in the repository, see section 13.5.

The release to the geosphere is dominated by nickel-59 and nio- bium-94, both of which come up to a flow of 10^6 Bq/y. The half-lives of these nuclides are 75 000 and 20 000 years, respectively. The quantity of nickel-63 in the repository is 100 times higher in the initial stage than the quantity of nickel-59. Its release can then amount to up to 10^8 Bq/y, but declines with a half-life of 100 years.

20.6   NUCLIDE TRANSPORT IN THE GEOSPHERE

Nuclide transport through the geosphere can be calculated with the aid of a mathematical model that takes into account

-    the source term from the redox front

-    the groundwater flow

-    dispersion effects

-    retardation due to inward diffusion and sorption in the rock matrix

-    radioactive chain decay

The model has been described in chapter 14. The calculations are reported in detail in /20-3/.

20.6.1   Premises

The medium through which the migration takes place is described as
a fractured rock with porous rock blocks between the fractures. The
radionuclides interact with the rock matrix in a number of ways and
the transport of a nuclide is retarded in relation to the water
transport by the following mechanisms:

-       Sorption on fracture-filling and -coating materials, surface
        sorption

-       Diffusion into the micropores in the rock matrix

-       Sorption on the inner surfaces of the rock matrix, volume
        sorption

Due to the long contact times between rock and water, the effect of
volume sorption is completely dominant. In the calculations of the
migration through the rock out to the fracture zone, surface sorp-
tion is therefore disregarded. This does not appreciably affect the
final results.

Simplified assumptions concerning the flow paths have been made in
the calculations. The flow in the rock mass is assumed to take
place over the shortest path from the waste canisters out to the
nearest fracture zone. See figure 20-7. It is further assumed that
the flow from all canisters travels this path. In the fracture
zone, the flow goes straight up to the biosphere. Only the retarda-
tion that takes place during transport in the rock mass has been
taken into account, whereas the retardation in connection with the
migration in the fracture zone has been neglected.

20.6.2   Data

The input data to the calculations of the nuclide transport in the
geosphere are:

-       source term data, i.e. inflow from the near field in Bq/y as a
        function of time. See section 20.5 and figures 20-3 to 20-6
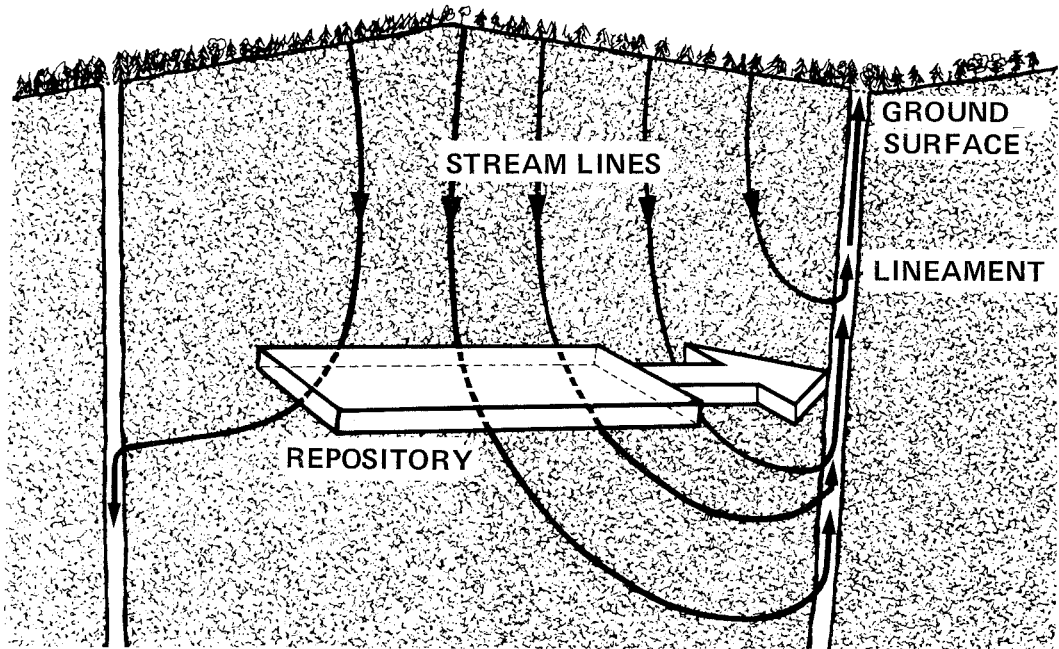
Figure 20-7. Schematic illustration of the flow field around a repository. The unfilled arrow shows the simplified model with the entire repository located at a distance of 100 m from the fracture zone and a direct horizontal flow from the repository to the fracture zone.

-   data on the groundwater movements in the form of water flow and migration distance. See section 20.3

-   data on the hydraulic fracture frequency in the rock. See chapters 14 and 18

-   data on dispersion (Peclet number). See chapter 14, sections 14.2.5 and 14.3

-   data on the sorption properties of the rock. See chapter 12

-   diffusivity and porosity data. See chapter 14

Table 20-1 in section 20.1 gives the data that have been used in the different scenarios. The hydraulic fracture frequency in the rock has been assumed in all scenarios to be one fracture every 5 m. High dispersion (Peclet number = 2) has been assumed. Calculations for low dispersion (Peclet number = 50) have also been carried out for some scenarios in order to check that less dispersion in time does not give higher release values.

One mechanism that can lead to radionuclides reaching the biosphere faster is that a certain portion is transported in the form of colloids or organic complexes. They are then retarded less or not at all during their transport through the bedrock. The premises for and effect of colloid-complex formation have been discussed in chapters 12 and 14. In the calculations, the colloids have been dealt with by assuming that the nuclides are irreversibly bound to the particles in the water. A concentration of 0.5 mg/l of particles in the water has been assumed. The formation of complexes has been taken into account by reducing the sorption coefficient for the trivalent actinides by a factor of 10.

20.6.3   Results

The outflow from the impervious rock to the nearest fracture zone as a function of time is obtained for each nuclide as the results of the migration calculations. The results are presented in detail in /20-3/. Only the releases of those radionuclides that are of the greatest relative importance for the dose burden are reported and commented on below. See section 20.7.

The calculation results are presented in figures 20-8, 20-9 and 20-10 in the form of annual releases in a fracture zone (biosphere) in Bq per year and canister as a function of time. Accordingly, in order to obtain the total release from the entire repository in scenarios A, C and D, the activity values given in the figures are multiplied by the number of canisters in question, 4 400.

Figure 20-8 shows the releases in Bq/year and canister for scenario A.

The releases in the central case are very low and are dominated by iodine-129 and cesium-135, which have no or very low sorption in the rock. The dissolution of uranium is so small in the central scenario and its transport in the rock so slow that the releases of uranium and its daughter nuclides do not emerge until after 100 million years and reach their upper values at between 1 and 10 billion years after deposition. It is hardly meaningful to discuss very small releases that are expected to emerge after such long periods of time.
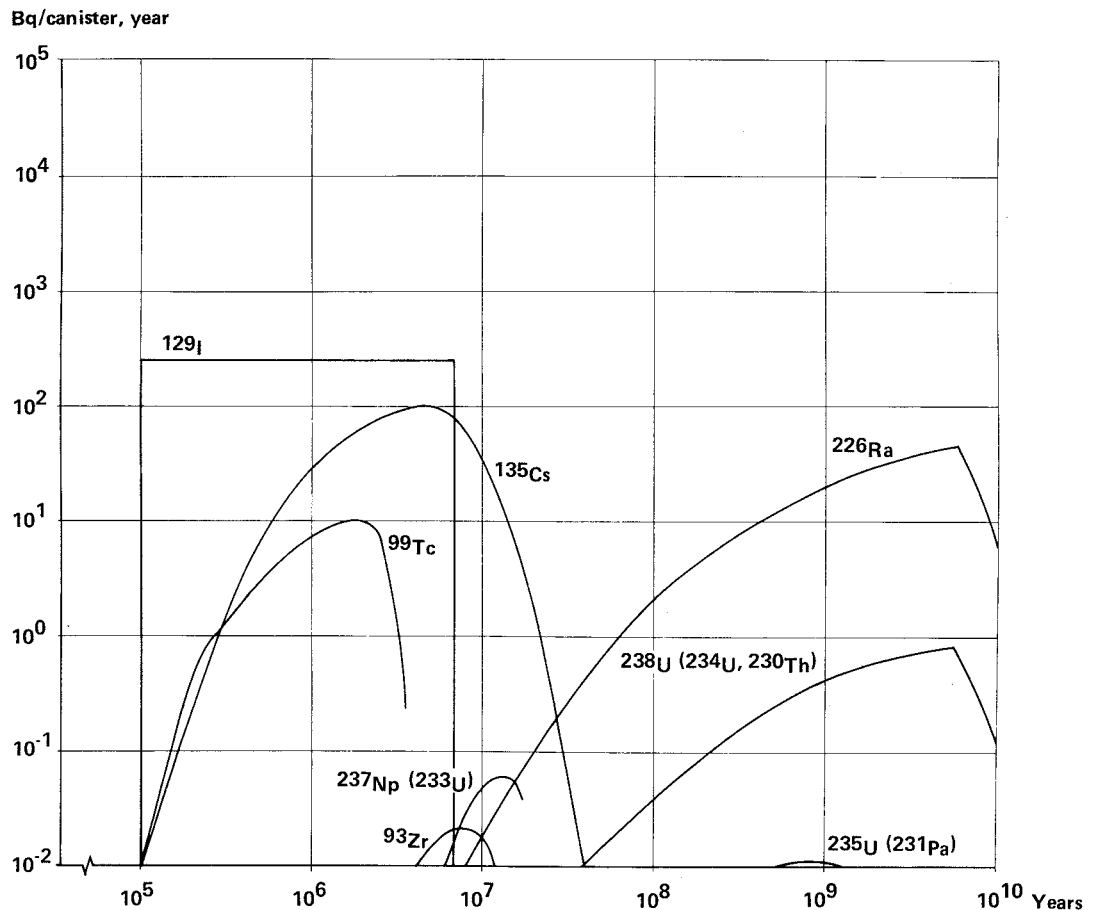
Bq/canister, year

Figure 20-8. Release to the fracture zone (biosphere) in Bq/canister and year as a function of time after fuel discharge. Scenario A (Central scenario).

Scenario B (initial canister damage) has been calculated using the parameters for the central scenario but with corrections for the influence of temperature on water flow and diffusion constants. The release of long-lived nuclides and their daughter products such as uranium-238, uranium-235, radium-226, neptunium-237 and technetium-99 are largely the same as in the central scenario figured for one canister. Plutonium and americium are retarded so much that the isotopes plutonium-239, plutonium-242, americium-241 and americium-243 decay to negligible levels before they reach the biosphere recipient. During the first 2 000 years, the pulse release of the 10% easily accessible iodine-129 and cesium-135 dominates.

Scenario C illustrates the effect of a situation where oxidizing conditions prevail all the way from the repository to the fracture
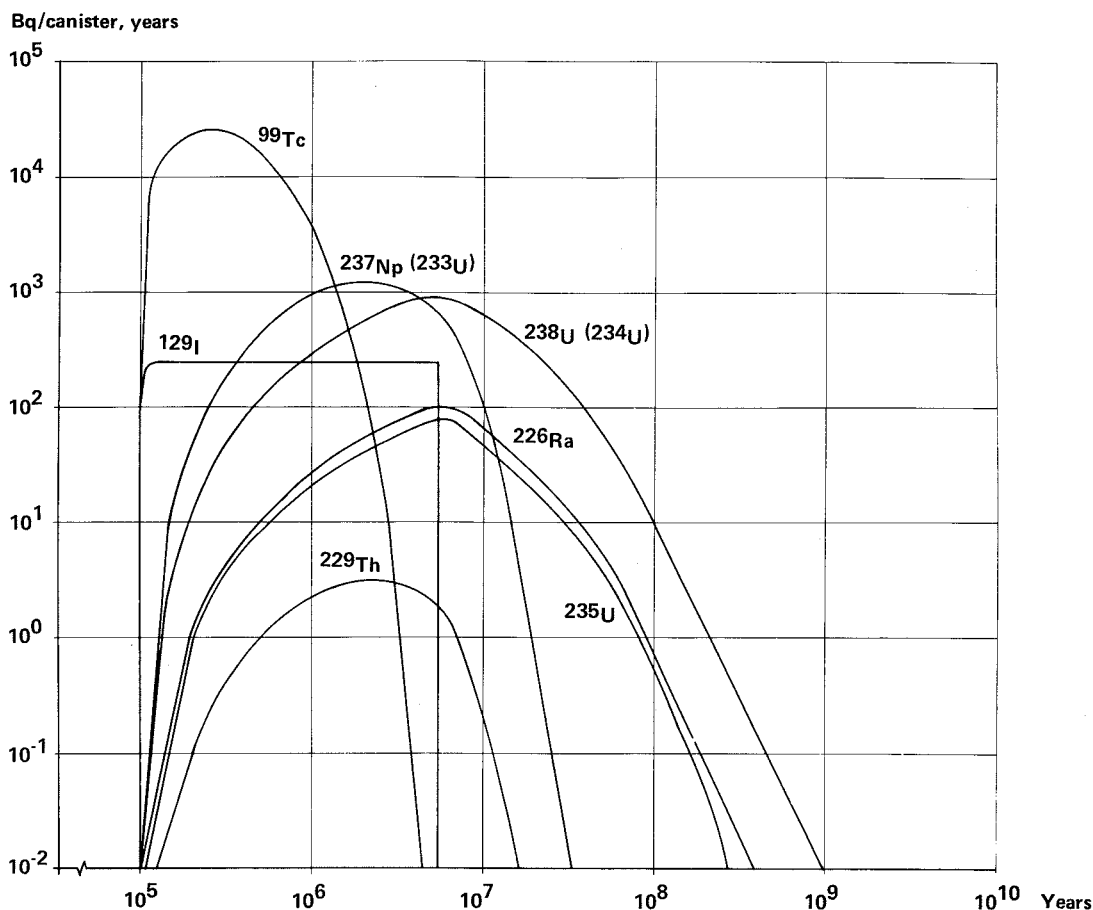
Figure 20-9. Release to the biosphere in Bq/canister and year as a function of time and fuel discharge. Scenario C (Oxidizing conditions).

zone. As explained in chapter 12, this affects both solubility and sorption data for technetium, neptunium, plutonium and uranium. The results of the migration calculations are shown in figure 20-9. It can be seen there that the assumption of oxidizing conditions leads to the result that the releases of uranium, radium, technitium and neptunium come earlier than in the central scenario. For uranium-238 and technetium-99, the release increases by a factor of about 10^3 compared to under reducing conditions. The release of radium-226 is on the same level. For neptunium-237, the influence of the faster migration is great, see figure 20-10; the release increases by a factor of about 10^4. This is due to the fact that in the central scenario, the transport from the redox front to the fracture zone is so slow that the release of neptunium-237 does not come until most has already decayed. Other nuclides are not affected by the assumption of oxidizing conditions.

Bq/canister, year
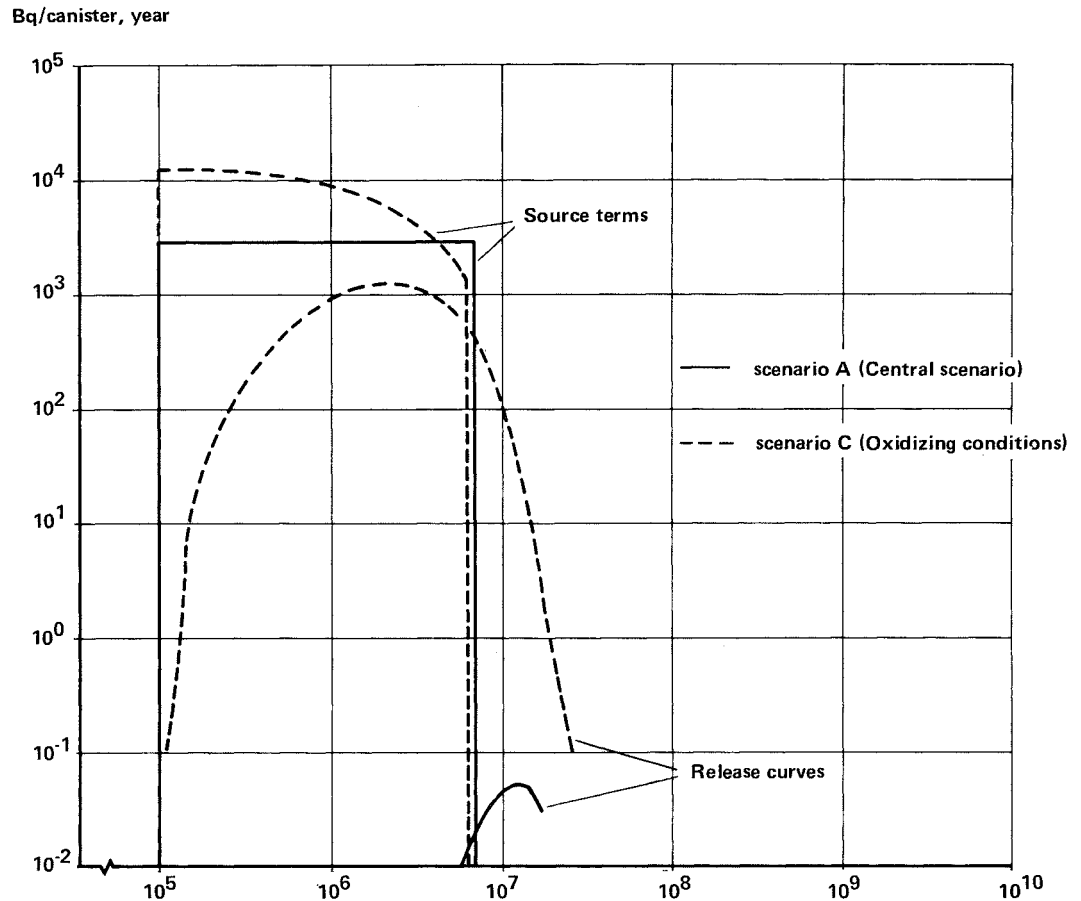
Figure 20-10. Source term and release of Np-237 in two different scenarios for nuclide transport in the geosphere.

The results in scenario D (colloid transport) are illustrated in figure 20-6, section 20.5.

The dominant nuclide under the assumptions made concerning colloid transport is protactinium-231, which is generated by the disintegration of uranium-235, and radium-226, which is generated by the disintegration of uranium-238.

20.6.4   Discussion

As was explained in section 20.1, the scenarios have been chosen to cover unfavourable possible situations and certain theoretical deviations from them. Naturally, they only represent some of the combinations of input data that could theoretically be postulated.

A calculation with extremely short migration distance, 5 m, has been carried out in order to illustrate the effect of the location of a water-bearing zone next to the repository. Such a situation can only be conceived as arising for a small fraction of all canisters in a repository. With an assumption of a water flow of 0.3 l/ (m^2 · year) in the rock mass and oxidizing conditions along the migration path out to the high-conductivity zone, releases that deviate less than one order or magnitude from the release in scenario C are obtained for all nuclides.

An example with extremely high water flow, 1 l/(m^2 · year), has also been calculated. Further assumptions in this case are rapid matrix dissolution, higher uranium solubility in the groundwater and a migration distance of 50 m. Naturally, the consequence is higher releases that emerge earlier than in case A. The release of radium-226 increases by a factor of 10^3 and reaches its maximum after 10 million years. The release of neptunium-237 is on the same order of magnitude as in scenario C (oxidizing conditions) /20-3/.

The maximum outflow to the biosphere of radionuclides from the metal component repository is 10^5 Bq/y and occurs 10^5 years after deposition. The outflow consists of nickel-59. Nickel-63 and niobium-94 have decayed due to their retardation in the rock.

In all reference cases, the retardation of the nuclides obtained during the transport in a fracture zone up to the biosphere is neglected. In reality, however, there will be a considerable retardation there as well. The fracturing in the fracture zone provides large surface areas and volumes for sorption. The amount of data available on this point is, however, insufficient to permit credit to be given to this effect. However, calculations under different assumptions of fracturing and block size indicate that a retardation is obtained during transport up through the 500 m long fracture zone that is of the same order of magnitude as that obtained during migration through 100 m of rock /20-4/.

In all scenarios reported, the hydraulic fracture frequency has been set at 0.2 m^{-1}, i.e. an average distance between water-bearing fractures of 5 metres. At the given groundwater flow, greater retardation is obtained the closer the water-bearing fractures are

spaced. The transport velocity for sorbing substances is inversely proportional to the fracture frequency, and a reduction of the fracture frequency by a factor of 2 has the same effect on the transport velocity as an increase of the water flow by a factor of 2.

20.7   DISPERSAL IN THE BIOSPHERE AND DOSE CALCULATION

20.7.1   Premises

The nuclide quantities that reach the biosphere via the groundwater in the rock comprise the source term for the dispersal calculations, which are carried out using a compartment model called BIOPATH. Two initial recipients have been studied for the biosphere: Well and lake. The nuclides are transported from these recipients primarily via surface water turnover to salt water reservoirs.

The well recipient has been assumed to be located in an outflow area 100 m from the edge of the repository (fracture zone). The total dilution is proportional to the entire repository's annual releases being mixed with 500 000 m^3 of water. Maximum well water consumption is assumed to be 6 m^3/d /20-2, 20-5/.

The lake recipient has been patterned on Morpa Lake in Fjällveden with a volume of 3.2 x 10^6 m^3 and a water turnover rate of once per 3.5 years. The lake thus has a very low rate of water exchange (about 30 litres/second).

The data on the transport of different nuclides in the food chains in the biosphere that have been utilized in the model calculations have been reported in chapter 15, tables 15-1, 15-2, 15-3 and 15-4. The conversion factors between intake of radionuclides and dose to man are given in chapter 16, table 16-3. The source term for re-leases of radionuclides to the biosphere for the different cases studied has been reported in section 20.6.3 and in /20-3/.

Sv/year
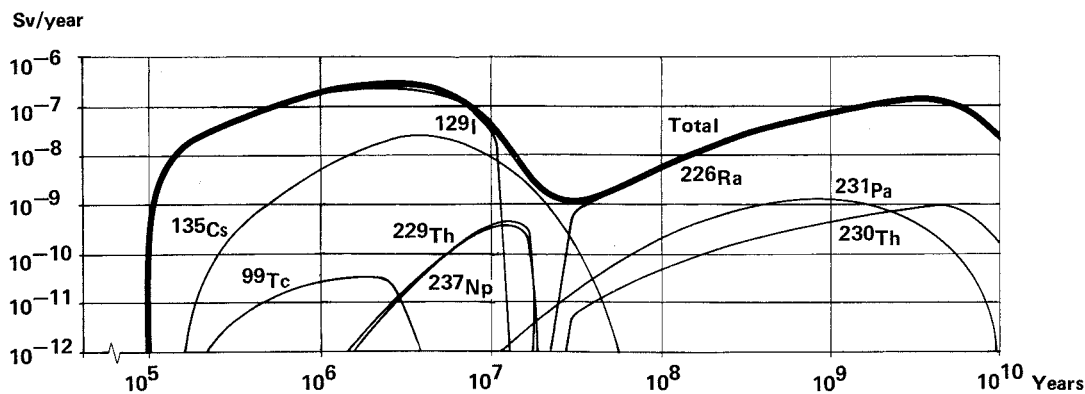
Figure 20-11. Calculated individual doses in scenario A (Central scenario) 4 400 canisters. Well recipient.

20.7.2   Calculation results and discussion

The results of the BIOPATH calculations for scenario A are shown in the form of individual doses from different nuclides for the initial recipients well, figure 20-11, and lake, figure 20-12, and in the form of collective doses in figure 20-13. The individual doses from scenarios A to E are also presented in table 20-7. A detailed account of the BIOPATH calculations is provided in /20-6/.

In scenario A (Central scenario), iodine-129 is the dose-dominating nuclide up to 10^7 years. Thereafter, radium-226 dominates the dose picture. The level for maximum individual dose is 10^{-7} Sv/y, i.e. completely negligible compared to background radiation.

The iodine and cesium doses apply for the 90% of these elements that is assumed to be released at the same rate as matrix disso-lution, i.e. during seven million years. In addition, 10% will be released at the time of the canister penetrations, which are distri-buted over 0.9 million years. Altogether, this entails an increase of the doses from iodine and cesium by a factor of less than 2 in the first million-year period.

A comparison between figures 20-11 and 20-12 shows that the maximum individual doses are affected by the recipient only to a very small extent. This is due to the fact that the lake recipient consists of a lake with a very low water turnover rate. Certain differences in dominant paths of exposure do exist, however. In the case of radium -226, intake via drinking water dominates in the well case, while
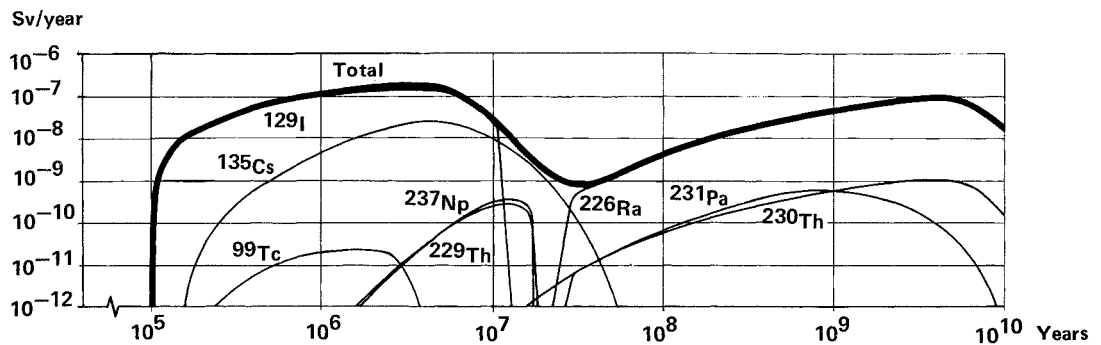
Figure 20-12. Calculated individual doses in scenario A (Central scenario) 4 400 canisters. Lake recipient.

intake via fish dominates in the lake case. For neptunium-237, intake via drinking water dominates in both cases.

Figure 20-13 shows cumulative collective dose for scenario A (Central senario).

The cumulative collective dose is completely dominated by the contribution from iodine-129. Owing to the slow release, the daughters in the uranium-238 and the uranium-235 chains do not contribute appreciably. The reported collective doses from iodine are only obtained if a change in diet takes place that entails a greater intake of marine organisms, algae and plankton.

Calculated collective doses are judged to be greatly overestimated for nuclides with half-lives of around one million years or more, since sedimentation in the global oceans can take the materials out of circulation in the global oceans more effectively than is assumed here.

The maximum individual doses for scenario B (initial canister damage, one canister) are shown in table 20-7. All nuclides except iodine-129 and cesium-135 are liberated here at the same rate per canister as in case A. Differences arise for those substances that are present in certain proportions in the fuel's cladding gaps, iodine and cesium. For the individual canister, the 10% present there will be liberated during a short time. The time is determined by the dilution of iodine that takes place in the bentonite buffer's water volume and its transfer from there to the rock's groundwater. The iodine and cesium release is hereby spread over more than 2 000
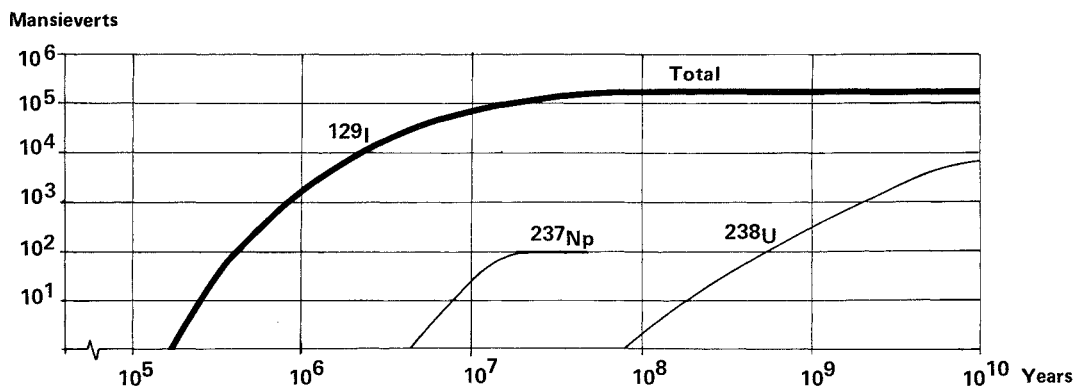
Figure 20-13. Calculated cumulative collective dose in scenario A (Central scenario).

Table 20-7.   Maximum individual doses in the different reference cases (Sv/y).

Nuclide	Scenario A (Central scenario) 4 400 canisters Lake recipient	Scenario B (Initial canister damage) 1 canister Lake recipient	Scenario C (Oxidixing conditions) 4 400 canisters Lake recipient	Scenario D (Colloid transport) 4 400 canisters Lake recipient	Scenario E (Peat bog recipient) 4 400 canisters Peat bog recipient
^{59}Ni	–	3×10^{-18}	–	–	–
^{93}Zr	3×10^{-13}	8×10^{-17}	–	5×10^{-10}	–
^{99}Tc	2×10^{-11}	7×10^{-12}	1×10^{-8}	1×10^{-13}	–
^{129}I	2×10^{-7}	1×10^{-8}	2×10^{-7}	–	–
^{135}Cs	2×10^{-8}	1×10^{-9}	2×10^{-8}	2×10^{-11}	–
^{226}Ra	7×10^{-8}	2×10^{-11}	2×10^{-7}	6×10^{-7}	2×10^{-7}
^{229}Th	2×10^{-10}	4×10^{-14}	1×10^{-8}	7×10^{-9}	5×10^{-10}
^{230}Th	8×10^{-10}	2×10^{-13}	2×10^{-9}	8×10^{-9}	–
^{233}U	1×10^{-11}	3×10^{-15}	4×10^{-7}	1×10^{-11}	–
^{234}U	2×10^{-10}	4×10^{-14}	2×10^{-7}	2×10^{-11}	–
^{235}U	2×10^{-12}	4×10^{-16}	1×10^{-8}	–	–
^{231}Pa	4×10^{-10}	1×10^{-13}	7×10^{-9}	3×10^{-6}	9×10^{-10}
^{238}U	2×10^{-10}	4×10^{-14}	2×10^{-7}	3×10^{-12}	–
^{237}Np	3×10^{-10}	7×10^{-14}	8×10^{-6}	4×10^{-8}	5×10^{-10}
^{239}Pu	–	–	–	4×10^{-11}	–
^{242}Pu	–	–	–	2×10^{-9}	–
Total dose	2×10^{-7}	1×10^{-8}	9×10^{-6}	4×10^{-6}	2×10^{-7}

years, which means that the doses from them during an initial stage will rise by a factor of 200 compared to scenario A. No credit has been given for the further dispersion in time that results from the fact that all fuel rods are not penetrated simultaneously.

In scenario C (Oxidizing conditions), table 20-7, the dose is dominated by neptunium-237, which has considerably higher solubility

under oxidizing conditions and is therefore transported out at higher concentrations in this case.

In the case of colloid transport, scenario D, table 20-7, protactinium-231 is the dose-dominating nuclide, approx. 3×10^{-6} Sv/y. This is due to protactinium's high sorption on solid surfaces and to the simplified but unfavourable assumption that sorption on colloidal particles is irreversible, while sorption on other surfaces is reversible.

In addition to the scenarios dealt with above, a case with a recipient consisting of peat bog has also been analyzed, scenario E.

The peat bog is assumed to be located in an outflow area and function in principle as an ion exchanger. After 10 000 years as a recipient, it is assumed that the material in the peat bog will be used as a soil conditioner.

With outflows from the geosphere as in scenario A, the doses given in table 20-7 from the dominant nuclides are obtained. The peat bog recipient gives a slightly higher dose from radium-226, but the total dose is the same as in scenario A.

The dose contribution from nuclides liberated from the metal component repository is dominated by nickel-59. With a conversion factor of 6×10^{-15} Sv/Bq, an outflow of 10^5 Bq/y gives individual doses of less than 10^{-10} Sv/y. Hence, the radioactive substances in the metal component repository do not affect the overall consequence picture from the entire repository.

20.8   SUMMARY

As stated in section 20.2, it is probable that the final repository will not give any doses at all to man in the repository's surroundings. In view of the very long time spans that must be taken into account and the unavoidable uncertainties this entails, a number of more or less unfavourable scenarios have been analyzed.

Table 20-8.   Summary of scenarios and results.

Scenarios	Main Characteristics	Summary of calculation results
A (Central scenario)	The canisters disintegrate successively during the period 10^5 to 10^6 years. The water flow is 0.1 l/(m^2 . year). Reducing conditions in the rock. Migration distance 100 m.	The matrix is dissolved in 7 million years. Uranium, neptunium and technetium precipitate at the redox front. The release of uranium to the geosphere takes 10^{10} years. The releases of uranium and radium are retarded until after 10^8 years. Neptunium is retarded so that it decays for the most part in the rock. ^{129}I and ^{135}Cs dominate the doses before 10^7 years, after which ^{226}Ra is dominant. The maximum dose level is around 10^{-7} Sv/y.
B (Initial canister damage)	Largely the same conditions as in A, but one canister is assumed to leak already after 10^2 years.	The releases per canister of the uranium isotopes radium, neptunium, cesium and technetium are nearly identical with A. Plutonium and americium are retarded so greatly that they decay to negligible levels before they reach the biosphere. The pulse releases, during about 2 000 years, of 10% easily accessible ^{129}I and ^{135}Cs dominate the dose. The level maximum level from one canister is 10^{-8} Sv/y.
C (Oxidizing conditions)	Same as in A but oxidizing conditions in the rock, which affect solubility and sorption for uranium, neptunium, plutonium and technetium.	The release to the geosphere increases for uranium (factor of 10^3), neptunium (factor of 10) and technetium (factor of 10^2) compared to A. The same substances are also sorbed less during their migration in the rock, which leads to 10^4 times higher releases of ^{237}Np than in scenario A. Maximum dose level is 9 x 10^{-6} Sv/y.
D (Colloid transport)	Irreversible sorption on particles (concentration 0.5 mg/l) in proportion to the concentration of the substances in the water and their sorption coefficients (K_d). Transport of the particles without retardation in the rock.	For the actinides, the particle-bound activity consitutes about 0.25% of the quantity of actinides in the water. The release of ^{231}Pa increases considerably and ^{231}Pa becomes the dose-dominating nuclide. The maximum dose level is 4 x 10^{-6} Sv/y.
E (Peat bog recipient)	Same input data as in A but the biosphere recipient is assumed to be a peat bog that is used as soil conditioner after 10 000 years.	The dose from ^{226}Ra is somewhat higher than in A. The total dose is the same as in A.

Table 20-1 in section 20.1 shows the input data that have been used in the different scenarios. Table 20-8 summarizes the results of the calculations for these scenarios.

STANDARDS AND NATURAL CONDITIONS **RADIATION DOSE (Sv/y)** **CALCULATED DOSES**

Recommended limit value for individuals (ICRP)

10^{-2}

Natural radiation in Sweden

10^{-3}

Design aim for nuclear power plants

10^{-4}

10^{-5} ← Scenario C (Oxidizing conditions)

← Scenario D (Colloid transport)

10^{-6}

10^{-7} ← Scenario A (Central scenario)
Scenario E (Peat bog recipient)

Probable doses

10^{-8} ← Scenario B (Initial canister damage, one canister)

Figure 20-14. Calculated doses in different scenarios in comparison with standards and natural conditions.

The doses from a final repository obtained in the different scenarios have been compiled in figure 20-14, together with certain recommended limits and guidelines as well as dose levels from natural radiation. As is evident from the figure, the doses in the central scenario are less than one thousandth of the natural background radiation. In all cases, the doses lie below applicable limit values.

As was pointed out in section 20.1, all scenarios incorporate safety margins in the assumptions that have been made concerning input data or certain mechanisms and processes. Table 20-9 gives some of the margins that are incorporated in the calculation assumptions for the central case.

Talbe 20-9.   Examples of safety margins and unfavourable assumptions.

Parameter or process	Assumptions used in calculations	Probable values and processes	Comments
Canister life	Gradual degradation of all canisters during the period 10^5 to 10^6 years. All fuel available for leaching after penetration.	Canister life much longer than 10^6 years. Only a small portion of the fuel becomes available for leaching.	Means that the releases are expected to come earlier than is probable. The decay effect for e.g. ^{237}Np is thereby neglected and the leached quantities are overestimated.
Water flow at repository depth	0.1 $1/(m^2 \cdot year)$	0.002-0.06 $1/(m^2 \cdot year)$	Means that the source term is overestimated, as are the migration velocities.
Dissolution of the uranium matrix	360 ppm uranium solubility	1 ppm (see chapter 11)	The rate of matrix dissolution is over-estimated. Affects the release rate for I and Cs, among others.
Precipitation of redox-sensitive substances at the redox front	Precipitation in relation to solubilities (table 12-8)	Coprecipitation with uranium of Pu, Np, Th and Am	Means that the source term to the geosphere is over-estimated for these elements. See table 13-5.
Sorption data	According to table 12-7, column for values used	According to table 12-7, column for best estimate	Migration velocity for actinides is overestimated.
Migration distance to biosphere	100 m distance between all canisters in the repository and the release point to the biosphere	From 100 m and up to several hundred metres migration in the rock mass. 500 m migration in fracture zone	The source term to the biosphere is greatly over-estimated.

21 ACCIDENTS AND EXTREME EVENTS

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┌──────────┐    ┌────────────┐    ┌──────────┐    ┌───────────┐    ┌───────────┐    ┌─────────────┐
│  SPENT   │───▶│ FACILITIES │───▶│   NEAR   │───▶│ GEOSPHERE │───▶│ BIOSPHERE │───▶│   SAFETY    │
│   FUEL   │    │            │    │  FIELD   │    │           │    │           │    │             │
└──────────┘    └────────────┘    └──────────┘    └───────────┘    └───────────┘    ├─────────────┤
                                                                                     │  ACCIDENTS  │
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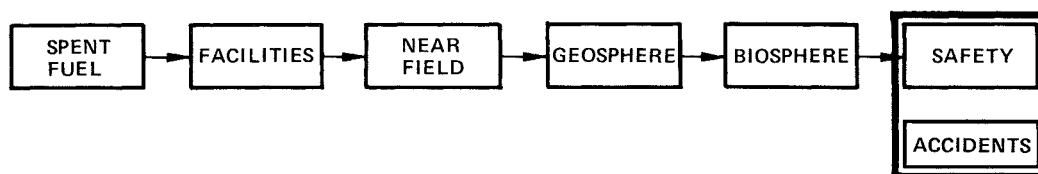
This chapter describes how certain accidents and extreme events can affect the safety of the final repository.

21.1 GENERAL

The preceding chapter deals with the slow processes that can eventually lead to the escape of radioactive substances from a final repository. It is also conceivable that radioactive substances will escape as a result of extreme or special sequences of events that are not covered by the premises that have applied for the preceding analysis. In the following, some extreme and special events are discussed. All of them are deemed to have very low probability.

The extreme or special events that could conceivably affect the final repository either have a natural origin or derive from human activities. The former group includes various kinds of bedrock movements, including earthquakes, as well as criticality in the final repository. The category "human activities" includes acts of war and sabotage as well as unintentional or intentional intrusion into the final repository.

21.2 BEDROCK MOVEMENTS

Bedrock movements could damage a final repository by creating new paths for groundwater flow and by damaging the canisters.

Damages to a limited number of canisters do not alter the conditions for the safety of the repository. The consequences have been analyzed in chapter 20 in scenario B, initial canister damage. The properties of the copper and the buffer material allow small

movements to take place without the integrity of the canister being affected /21-1/.

The probability of bedrock movements that could affect the safety of a final repository is discussed in chapter 8.

An idea of the frequency and size of bedrock movements over long periods of time can be obtained by studies of fracture displacements in exposed rock surfaces. To obtain information on movements in shallow strata of sound rock, observations have been made in different coastal areas. On the basis of the frequency, size and spacing of the displacements, the probability of a future displacement of 1 mm has been estimated. Applied to a final repository, this estimate would entail that a single canister might be affected during a period of one million years. The fracture movements at greater depth should be less than in shallower rock, and minor displacements on the order of a cm will not damage a canister.

As was discussed in greater detail in chapter 8, fracture movements and their frequency during certain periods many millions of years ago have been greater than the average. At the present time, Sweden is in an inactive geological period with a lower frequency of fracture movements.

As was more closely analyzed in the same chapter, fracture movements during the next few millions of years can only be expected to bring about insignificant changes in the hydraulic conductivity of the rock.

In summary, the studies made of bedrock movements have shown

- that the probability of movements in the Swedish bedrock that adversely affect the safety of the final repository is extremely low

- that within areas surrounded but not intersected by fracture zones, the probability that new flow paths will be opened is extremely low

- that neither the buffer layer nor canisters will be damaged, even if earthquakes of considerable magnitude by Swedish standards should affect the repository site.

21.3   CRITICALITY IN THE FINAL REPOSITORY

The possibility that criticality, i.e. a self-sustaining chain reaction, could occur with the fissionable plutonium-239 and ura- nium-235 which is present in the repository is virtually nil. The question has been thoroughly analyzed in a technical report /21-2/. A brief summary is provided here.

Criticality with plutonium-239

In principle, sufficient plutonium-239 is present in a single canister at the time of deposition in order for criticality to be reached. But first, some process must separate plutonium from uranium in the fuel and collect it in concentrated form in a manner suitable for criticality. This is not plausible for a number of reasons.

Since the expected life of the copper canisters is considerably longer than the half-life of plutonium-239, there will not be enough plutonium present for criticality if and when the copper canisters are penetrated. The case is therefore only of interest for a single canister which might have initial canister damage.

Criticality inside a damaged canister requires that the uranium present there be dissolved and transported out of the canister before plutonium-239 has had time to decay sufficiently.

Criticality outside a damaged canister requires that uranium and plutonium be dissolved together and that plutonium be selectively precipitated in the bentonite buffer outside the canister. Selective leaching of plutonium is not possible for chemical reasons. See chapter 12.

Neither internal nor external criticality with plutonium is a reasonable possibility, however, since it requires a more rapid

dissolution (on the order of 50 000 years) of uranium or of uranium and plutonium than is deemed to be possible (millions of years) under existing conditions of water flow and carbonate concentrations. Furthermore, the plutonium must be in a geometrical configuration that is highly unlikely.

In summary, the probability of plutonium criticality is extremely low. Furthermore, the consequence of a postulated criticality would be insignificant. The course of events is characterized by a slow heat generation and an increase in temperature to a theoretical maximum of the boiling point of water (around 265°C at the pressure prevailing at 500 m). If the temperature rises higher, water boils off and the reaction stops. The effects on nearby canisters will not jeopardize their integrity. The amount of long-lived radiotoxic nuclides formed in connection with criticality is small in relation to the amount already present in the failed canister.

Criticality with uranium-235

Criticality with uranium-235 is not possible inside the canisters owing to neutron physics considerations. Such criticality is only possible in tunnel systems and deposition holes outside the canisters. Owing to the long half-life of uranium-235, the risk of criticality in this case does not apply only to an isolated initially damaged canister. Calculations show that the minimum critical mass in the tunnels is around 4 400 kg, which means that all the uranium from at least four canisters must be accumulated within the critical geometry (a sphere).

As is evident from chapters 12 and 13, the solubility of uranium in groundwater is very low. Furthermore, the transport resistances in bentonite and tunnel fill are great. All transport takes place through diffusion and takes a very long time. Moreover, the chemical environment in the water-saturated tunnel fill is expected to be reducing after the initially entrained atmospheric oxygen has been consumed. This means that only very low uranium concentrations can occur dissolved in the water in the tunnel. It is difficult to imagine any process that leads under these conditions to the accumulation of a relatively large quantity of uranium from several

canisters within a limited volume with optimum geometry. If the geometry is not optimal, a larger quantity of uranium is required for criticality.

Even though the risk of uranium criticality is thus extremely low, the consequences of this hypothetical case have also been calculated. A sudden, heavy release of energy is out of the question. Criticality can only be built up by a slow accumulation of uranium. The thermal power output from a hypothetical critical mass inolving all of the uranium deposited in one of the tunnels would be lower than 130 kW. The formation of fission products would be equivalent to that of 900 tonnes of deposited fuel, which would not increase the radiation doses to any great extent in relation to the reported scenarios without criticality.

In summary, it can be concluded that a number of extreme or improbable prerequisites must be postulated in order to achieve a critical configuration. The probability of this occurring is deemed negligible. The consequences of hypothetical cases of criticality from both plutonium-239 and uranium-235 have, moreover, been calculated to be insignificant.

21.4   ACTS OF WAR AND SABOTAGE

In the long time perspective which is relevant for the final repository, acts of war cannot be considered to be "extreme events". On the other hand, the possibility that acts of war might lead to serious consequences for the safety of a finally sealed final repository at a depth of some 500 m in the Swedish bedrock must be considered to be remote.

Ground detonations of nuclear devices of 10-50 megatonnes create craters in the rock with a depth of roughly 110-180 m /21-3/. The geological barrier would thus not be penetrated, but might well be weakened. In such a situation, however, this would be of subordinate importance, since any release of radioactivity from the final repository would represent only a fraction of the radioactivity from the bomb, which would remain in the area for a long period of time.

Wartime damages to the final repository and the encapsulation station during the deposition stage are, naturally, conceivable. But the probability is low, since these facilities are not likely to be primary targets for military actions. The consequences of bomb hits and similar occurrences would also be limited compared to the other consequences of such acts of war.

Safeguards against sabotage are foreseen during intermediate storage, encapsulation and deposition in the final repository. After the final repository has been closed and sealed, effective acts of sabotage are out of the question.

Compared to other installations that experience has shown to be likely targets for sabotage and terrorist actions, the facilities described here are less attractive to potential saboteurs and are most closely comparable to other industrial plants where environmentally hazardous material is handled.

21.5   INTRUSION INTO THE REPOSITORY

Conceivable human intrusions that could affect the safety of the final repository may either be intentional or unintentional. An intentional intrusion into the final repository may have the purpose of recovering the useful materials that it contains, i.e. the copper or the spent fuel. An unintentional intrusion into the repository can only occur if knowledge of the location of the repository has been lost.

An inescapable principle is that each generation must take responsibility for its own conscious actions. At the same time, it should be regarded as an obligation for former generations to pass on to later generations the information that is required to avoid unnecessary risks and injuries. In other words, it is of the utmost importance that information on the final repository be carefully recorded and preserved so that future judgments do not have to be made on the basis of incomplete data.

If, with preserved knowledge of the repository and its content of radioactive substances, man decides in the future to intrude into

the repository area, it must be assumed that he also takes into account and is in control of the radiological risks that such an action might entail. A possible reason for such intervention is to recover useful materials in the area. Such materials may include the copper in the canisters, rare substances in the fuel, the spent fuel itself or the thermal energy that is stored in the rock mass during the first few thousands of years. Since the repository is deliberately situated in an area where there are no valuable minerals, the possibility of mining can be excluded. The great depth of the repository and the low water flow in the selected impervious rock precludes use of the area as a source of water supply on any large scale.

Knowledge of the final repository could conceivably have been lost at some point in time in the future, either as a result of some catastrophic event such as a global war of extermination or as a consequence of human life being rendered impossible during a given era due to a new glaciation. If the country is thereafter repopulated, it is conceivable that certain activities might violate the barriers of the final repository.

In order to rediscover the repository from the ground surface, for example by means of geophysical methods, such a future civilization must have access to advanced technology. They should then also have the ability to detect and handle the radioactive materials that are stored in the repository and whose radioactivity will then be considerably lower than today.

In this case as well, the area is unsuitable for mining operations or for use as a source of water supply. There is no logical reason for constructing underground storage facilities or the like at such great depth.

In order to make use of the heat that has been stored in the rock, heat recovery operations must be undertaken during the next few millenia. The probability that knowledge of the repository will have been lost within this span of time is naturally less than over longer periods of time. The radiological risks associated with such heat recovery can be overcome with technology that is no more advanced than that required for the heat recovery as such.

21.6 OTHER EVENTS

Serious mistakes in connection with the deposition procedure or sealing of the repository can also be included in the category "extreme events". Such mistakes are avoided by the use of a carefully planned and executed quality assurance programme. It should be rather simple to execute, since the activities connected with the actual final disposal procedure are uncomplicated and easy to oversee.

The most serious type of mistake would probably be leaving some open connection between the final repository and the biosphere. It is therefore necessary to keep a close check on all boreholes, shafts and tunnels that lead to the repository. During sealing, all such connections must be backfilled and sealed. The consequence of forgetting e.g. one borehole is, however, minor as a rule. Only if it leads from the vicinity of an initially damaged canister directly to an outflow area is there a risk for a large release of radioactive substances.

22   GENERAL SAFETY CONCLUSIONS

22.1   GENERAL

According to Swedish law, the nuclear power producers are responsible for the safe dispoal of radioactive waste from the nuclear power stations. The nuclear power utilities have contracted the jointly owned Svensk Kärnbränsleförsörjning AB (Swedish Nuclear Fuel Supply Company) to take responsibility for the necessary measures. This is done in accordance with a long-range plan, which is updated annually.

An important part of the work is to develop a system for the final and safe storage of spent nuclear fuel. Under Swedish law, a reactor owner must show that the spent nuclear fuel can be handled and finally disposed of in a safe manner before the reactor can be loaded with fuel for the first time.

The radioactivity in the spent nuclear fuel gradually declines with time. Some radioactive substances in the fuel, however, have such long half-lives that they can be equated in practice with stable toxic substances. As yet, there is no legislation requiring an accounting of how such substances can affect the envrionment over the very long run. It has therefore not been possible to evaluate the safety of a final repository for spent nuclear fuel in this report in the light of what general view society takes of safety problems that affect the environment in the very long run.

The account presented in this report is based on the state of knowledge achieved as of the start of 1983.

22.2   SAFETY IN THE HANDLING CHAIN

The handling and treatment steps that precede the final storage of the spent nuclear fuel have their counterparts within industry and nuclear power plant operation. The necessary safety measures can therefore largely be based on experience. No technical difficulties in meeting stringent safety and protection requirements are foreseen.

22.3 THE LONG-TERM SAFETY OF THE FINAL REPOSITORY

22.3.1 Time perspective

The radioactivity in the spent fuel declines very rapidly at first. Ten years after discharge from the reactor, the spent fuel contains only one hundredth of its original radioactivity. In order for the activity in the "ten-year-old" fuel in turn to decline to one hundredth, a period of 10 000 years is required. A further reduction of radioactivity to one hundredth takes place during the period from 10 000 years to 10 million years.

During the long spans of time that must be considered here, many changes will occur in our environment.

- During a period of on the order of 100 years, nature can be altered, for example by the eutrophication of lakes.

- During a period of on the order of 10 000 years, essential climatic changes can take place that could possibly lead to a new glaciation.

- During a period of on the order of one million years, evolution can lead to the origin of new species and the extermination of others.

- During a period of on the order of several tens of millions of years, major geological changes can take place.

Naturally, the uncertainties in forecasts and future assessments are greater the longer the periods of time they are applied to. In order for the safety evaluation of a final repository for spent nuclear fuel to be complete, however, it is necessary to shed light on anticipated conditions very far in the future. The properties of both the rock formation that will be selected for the final repository and the materials included in the engineered barriers are characterized by the fact that they have persisted over long geological periods of time. It is therefore meaningful to carry out evaluations of the barrier functions in the final repository even

over a perspective of some million years. In time perspectives beyond some million years, the isotope composition of the spent nuclear fuel will be largely the same as that of the uranium that occurs naturally in the Swedish bedrock.

22.3.2 Storage site

The Swedish crystalline bedrock is very stable geologically and no major changes can be expected in the hydrological or geochemical conditions of the bedrock at a depth of several hundred metres over the next million years or so.

The safety analysis is based on geological, hydrological and geochemical data that have been obtained from extensive investigations at different sites in Sweden. Conditions vary somewhat between the sites, but the safety analysis covers most of the variations that have been observed.

The site investigations and the safety analysis have shown that there are several sites in Sweden where the bedrock is well suited for the construction of a safe final repository. The recently investigated areas at Gideå and Kamlungekölen, and probably also Fjällveden, as well as the previously investigated area at Sternö are examples of such sites. The analyses have shown that even sites such as Finnsjön, with relatively higher groundwater flows than those mentioned above, should be acceptable from the point of view of safety.

Before the site of the final repository is definitely decided on at some time towards the close of the 1990s, additional areas will be investigated so that a sufficient body of data will be available for an optimal site choice.

22.3.3 Safety

The spent fuel is surrounded in the final repository by different barriers. They have been designed and chosen to isolate the fuel completely from the environment over a very long period of time and

to retard and dilute the radioactive substances that can eventually leak out from the repository over an even longer period of time.

A canister made of copper with a wall thickness of a few cm is expected to remain intact and leakproof for at least one million years, probably much longer.

The bentonite clay in the storage holes comprises a sealing layer as well as a mechanical and chemical buffer between the canisters and the rock mass. Geological observations show that bentonite is a durable natural product that retains it properties for at least some million years, provided the temperature does not exceed $100^{\circ}C$.

Both geochemical observations in nature and laboratory studies show that uranium and other actinides present in the spent fuel have very low solubility in water. Together with the very low rate of groundwater flow in the final repository, this ensures that the leaching of the radioactive substances that are left in the fuel when the canisters have been penetrated will be extremely slow.

In the chemical environment that prevails in the groundwater in the rock, the radioactive substances will migrate extremely slowly. Sorption in the microfissures in the rock will retard their transport.

An analysis of the function of the described barrier system and knowledge of how naturally occurring uranium moves in the rock indicate that the final repository will not affect the environment at all. However, sufficient knowledge and data have not been gathered to demonstrate this in an absolutely conclusive manner. Dispersal calculations have therefore been carried out for a number of postulated cases.

Unfavourable (pessimistic) premises have been chosen in the various calculation cases, for example with regard to the life of the canisters, the rate of water flow in the rock and the chemical retardation effects. Even then, the calculated doses are insignificant - on the order of between a thousandth and a hundredth of the dose from natural radiation - and they do not arise until a very distant future. Compared to current radiological standards, the

repository system described here provides a very large margin of safety.

The analyses show that considerable scope should exist to achieve, through continued research and development, a solution which, without compromising the high safety requirements, is much more favourable with respect to economy and resource utilization.

22.4   CONCLUSION

Spent nuclear fuel from the Swedish nuclear power plants can be handled and finally disposed of in a manner that satisfies very high demands on safety and radiation protection. This handling and final disposal can be carried out using technology currently known and available in Sweden. The bedrock at a number of places in Sweden possesses the quality required for a safe final repository.

The disposal method described here is flexible and can be adapted to local conditions. Continued research is expected to provide a basis for considerable improvements with regard to economy and resource utilization.

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