KBS TECHNICAL REPORT

121

KBS Technical Reports 1 – 120 (1977 – 1978)

SUMMARIES

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KÄRN-BRÄNSLE-SÄKERHET

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FOREWORD

A Swedish law of 1977 stipulates that the owner of a nuclear reactor has to show how safe final storage of spent fuel or high level radioactive waste from the reprocessing of spent fuel could be effected before permission for fuel loading can be granted by the government.

The Swedish nuclear utilities started early in 1977 the KBS-project (kärnbränslesäkerhet = nuclear fuel safety) to study the high level waste problem and report on how and where a safe final storage could be arranged in Sweden.

The result of the work performed by KBS has been presented in two main reports corresponding to the two optional alternatives given by the law. The first report deals with the final storage of vitrified waste from reprocessing and is entitled

> Handling of Spent Nuclear Fuel and Final Storage of Vitrified High Level Reprocessing Waste

It was published in December 1977. The second report deals with the final storage of spent fuel which is not reprocessed. It is entitled

> Handling and Final Storage of Unreprocessed Spent Nuclear Fuel

and was published in June 1978.

The documentation produced by the project during 1977 and 1978 has been collected in a series of technical reports numbered from 1 to 120.

The series of technical reports were intended to provide a suitable way for systematic information distribution within and between the different working groups of the KBS-project and for the offical reviewers appointed by the Swedish government. They have been printed only in a limited number of copies and are generally not available any more. A considerable international interest has been paid to the work and results of the KBS-project, and several technical reports are now out of print. Therefore, all the English summaries of the technical reports have been collected in a separate volume, No. 121, which is hereby presented.

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

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

Most reports are in Swedish with an English summary but some are completely in English or have been translated into English as indicated in the listing, page 3.

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KBS' TECHNICAL REPORTS 1977–78
KEY WORD REGISTER

KBS Technical Report No 01

EMISSION RATES IN SPENT FUEL AND HIGH-LEVEL WASTE FROM A PWR, CALCULATED USING ORIGEN

Nils Kjellbert, AB Atomenergi 1977-04-05

Summary

Radionuclide inventories in PWR spent fuel and high-level waste have been computed. Apart from fission products and heavy nuclides the activation products emanating from the cladding have been included for the spent fuel case. As times-to-reprocessing 1, 3 and 10 years of cooling have been used and the inventories have been calculated for times from 1 year through 10^7 years after shut-down.

For specification of the reference alternative please look under 2. below.

Separated fractions in the reprocessing:

U	99.9 %
Pu	99.5
H	100
Kr, Xe	100
C	∿93

A special treatment of the carbon 14 inventories has been made in 5. below.

KBS Technical Report No 02

MEMORANDUM CONCERNING THE THERMAL CONDUCTIVITY OF SOIL

Sven Knutsson, Roland Pusch,
Luleå Institute of Technology, 1977-04-15

SUMMARY

A comprised survey of the available literature concerning the thermal conductivity of soil materials is done. Dry, moisty and water saturated masses of both artificial and natural materials are described. For calculation of the thermal conductivity, a technique proposed by Ø Johansen at NTH (Norway) is used. For a dry mass, with the porosity (n), the thermal conductivity can be calculated by using the equation (7) if the masses consist of natural soil deposits. If the masses originate from crushed products equation (9) should be used. For water saturated soils the porosity is of much less importance than in dry masses. The mineralogical composition has the utmost influence on the conductivity in saturated soils. Computations should therefore be performed by the equations (10) and (11) where q is the volumetric content of quartz.

If the soil is nonsaturated the conductivity can be calculated by the expression (12) where K_e is the "Kersten number". This is zero for dry material, one for saturated material and for nonsaturated material it follows the equations (13) and (14). The expression (14) is most suitable for clay and silt. Equation (13) should be used when dealing with sand and coarser materials.

KBS Technical Report No 03

DEPOSITION OF HIGH-LEVEL RADIOACTIVE WASTE PRODUCTS IN BORE-HOLES WITH BUFFER SUBSTANCE

Roland Pusch, Anders Jacobsson
Luleå Institute of Technology, 1977-05-27

English summary

Scope of investigation

The present investigation comprised a compilation of available literature data concerning the possible use of clayey masses as buffer substances in bore-holes (in rock) with canisters containing radioactive waste products. The aim was to find a suitable composition of the buffer mass and to recommend a suitable storing technique.

Main questions

The criteria concerning the function of the buffer substance were:

- Sufficient mechanical supporting power to prevent rock pieces from the bore-hole walls to enter the mass. Also, sufficient bearing capacity to carry the heavy canisters.

- Suitable mechanical properties to guarantee a homogeneous character of the buffer mass in case of small differential movements in the surrounding rock. This requires a plastic (non-brittle) behaviour of the buffer substance.

- Prevention of free circulation of ground water. This requires a homogeneous character and a very low permeability of the buffer substance.

- Ion-adsorption ability. This requires a sufficiently high base exchange capacity and a suitable originally adsorbed ion type.

- Sufficiently good heat conduction properties to prevent high temperatures to develop in the buffer mass.

Composition of the buffer substance

These criteria suggest that a buffer substance containing Na-montmorillonite (Na-saturated smectite) would be suitable. A close examination of the mechanical behaviour, especially with reference to the application of the mass, showed that homogeneous condition cannot be obtained unless the bentonite is in a liquid or semi-liquid state. This would require a water content of more than 150% which, however, reduces its bearing capacity to an unacceptable value. An air-dry condition at the application, on the other hand, would give the mass a homogeneous character and an acceptable bearing capacity (at least if the canisters are carried by some device until the mass is entirely embedded by the powder). There are, however, two important problems: 1) The water uptake, which will take place when ground water enters the mass, will create a very high swelling pressure. If free swelling is allowed, the water content will increase considerably and cause a rapidly decreasing shear strength of the mass. The bore-holes must therefore be sealed. 2) The heat conduction capacity will be very low even if the bentonite is fully water saturated.

It is concluded therefore that the mass should consist of a moderate amount of dry bentonite and a fairly high amount of silt/sand-sized quartz powder. A 10% (by weight) bentonite and 90% quartz fraction seems to fulfil the geotechnical and physical requirements. The general idea is that the holes for deposition (5-10 m deep) should be bored from the tunnel base. Since the buffer mass in the holes, due to the fairly low content of active clay substance, will have a fairly low base exchange capacity it is necessary to create a second effective barrier. This is obtained by filling the tunnels with a buffer substance consisting of the same components but with a higher content of montmorillonite (20-50%). The buffer substance in the tunnels should also be applied in an air-dry condition. In the course of water uptake it will swell and exert a considerable pressure on the tunnel periphery, thereby tending to close open cracks and fissures.

Stability of the buffer substance

Since the canisters will have a temperature of about 100°C and the water pressure will be of the order of 5 MPa at 500 m depth (where the deposition tunnels will be located) it is necessary to consider any possible chemical or mechanical change of the mass with special referance to pH and salinity changes in the system and to the influence of strong radiation from the radioactive objects. Two possible changes were considered: 1) The risk of crystal distortion leading to mechanical destruction of individual particles and to a severe decrease in base exchange capacity and 2) the risk of precipitation of various components leading to cementation, i. e. a transfer from plastic to brittle behaviour of the buffer substance. Literature studies and own experience show that montmorillonite is permanently stable at 100°C temperature and 5 MPa pressure when pH is within the range of 6.5-10 while quartz is stable at pH <9. If the salinity is increased due to intrusion of salt water zeolites may form and if free K-ions occur mixed layer minerals (such as illite/montmorillonite) may be created. The only possible influence of radiation would be a mechanical degradation which would in fact only have two (positive) effects: an increase of the base exchange capacity and a reduction of the permeability. Considering the moderate pressure and the very low water conversion in the system, the buffer substance in the bore-holes as well as in the tunnels will hardly be affected at all. This is in fact proved by the existence of many natural plastic bentonites consisting of montmorillonite and quartz which have been affected by high pressures and temperatures of the given order for geological periods of time. Preliminary tests also show that structural changes such as piping due to gas movements in sealed bore holes will not occur.

Conclusions

The authors conclude that the suggested principle of storing the canisters in sealed bore-holes filled with a 10% bentonite/90% quartz (silt, sand) mass is suitable provided that the tunnel system, from which the holes are bored, is sealed with a dense buffer mass consisting of quartz (silt, sand) and 20-50% bentonite powder.

KBS Technical Report No 04

DEPOSITION OF HIGH-LEVEL WASTE IN TUNNELS CONTAINING BUFFER MATERIAL

Arvid Jacobsson
Roland Pusch
Luleå Institute of Technology, 1977-06-01

The buffer substance required for the tunnel concept as well as for the authors' previously suggested concept with bored deposition holes, should have the following properties:

1. Plasticity
2. Very low permeability
3. Sufficient bearing capacity (high shear strength and very high viscosity)
4. Sufficient ion exchange capacity
5. Sufficient heat conductivity
6. Temperatures lower than 100°C and strong radiation must not produce noticeably changes of the properties even after several thousand years.

These requirements suggest a buffer mass composition of 10% Na-bentonite and 90% silt/sand-sized quartz. The canisters with the radioactive wastes can be deposited directly in a buffer mass in tunnels or enclosed in containers (with buffer substance) which are placed on supports in the tunnels and which are surrounded by buffer substance. The first-mentioned technique yields a number of problems, the homogeneity and difficulty of obtaining a sufficient density of the mass in the top part of the tunnels, being the main ones.

The container concept is recommended and the 10% Na-bentonite/90% quartz composition is suggested for the buffer mass compacted in the precast concrete containers in order to form the required impervious, heat conducting zone between the container wall and the canister. The mass should be applied in a fairly dry condition to give a satisfactory density.

The use of concrete containers yields positive and negative effects, the last-mentioned being dominant. pH may increase which can lead to a montmorillonite-zeolite transition (analcite). Also, Ca can replace Na which produces aggregation and an increased permeability and a risk of cementation (brittleness). The corrosion of steel reinforcement may also produce cementation (iron compounds).

The container principle has the disadvantage also, that the restricted life-time of the containers requires effective barrier functions of the tunnel system as well. This is difficult to achieve since complete filling of the slot between containers and rock with a bentonite-rich buffer material can hardly be obtained. The authors therefore claim that the "container technique" should be analyzed in greater detail. It is concluded, after a detailed discussion of the lay out of the "77-05-05" proposal, that bore hole deposition is superior to tunnel deposition.

KBS Technical Report No 05

PRELIMINARY TEMPERATURE CALCULATIONS FOR THE FINAL STORAGE OF RADIOACTIVE WASTE IN ROCK, REPORT 1

Roland Blomqvist, AB Atomenergi, 1977-03-17

SUMMARY

This report gives details concerning pre-requisies, assumptions and results for some introductory temperature calculations regarding final disposal of radioactive waste in hard rock. The calculations, which illustrate the influence of dimensional parameters, age of the waste etc, include high active waste from reprocessed nuclear fuel and waste in form of non-reprocessed spent fuel.

The calculations mainly deals with a repository geometry, where the waste containers are placed in vertical holes and surrounded by a layer of quarts sand mixed with bentonite.

For non-reprocessed fuel calculations have also been performed for an alternative geometry. In this case, the waste containers are placed directly in horizontal funnels filled with a quarts sand - bentonite mixture.

Especially for the reprocessed waste, several parameter combinations result in too high temperatures. One way to lower the temperature level could be to choose a smaller waste container diameter compared

to those 450 mm which on basis of foreign statements has been assumed for most of the calculations. The non-reprocessed fuel gives, in general, lower max. temperatures. The reason for this is the much lower volumetric heat generation.

The report points out that the maximum temperature of the waste is strongly depending on the thermal conductivities of those materials surrounding the waste. However, due to different material composition, water content, temperature etc, those properties could be strongly varying. It is of importance to verify by experiment the thermal conductivities of the materials used.

KBS Technical Report No 06

GROUNDWATER MOVEMENTS AROUND A REPOSITORY, PHASE 1, STATE OF THE ART AND DETAILED STUDY PLAN

Ulf Lindblom, Hagconsult AB, 1977-02-28

INTRODUCTION

This report summarizes the conclusions and recommendations reached in the Phase 1 studies as outlined in the Hagconsult proposal entitled "Radioactive Waste Repository for Sweden - Long Term Containment Safety: Study based on rock mechanical, hydrogeological and thermal analyses", dated December 1976. The objectives of these studies are to provide a state of the art review of groundwater flow in the region of a repository in granitic rock, in order to provide a basis for long term containment assessments and to prepare a detailed study plan for the continuation of the project.

We present a general description of the problem, including a summary description of the spent fuel and expected site geology, together with a qualitative description of the expected response of the reference repository concept. In addition, the different processes affecting the groundwater situation for containment are given.

In section 3, we present a state of the art review of the fluid flow, geochemical, heat transfer and rock mechanics processes as they relate to containment. In each case, data availability, prediction and validation procedures and monitoring techniques are discussed.

In section 4, we present a detailed study plan to provide a comprehensive assessment of the hydrogeological regime around the repository during its lifetime. This necessarily requires treatment of many complex coupled processes and is based heavily on the use of mathematical simulation models which will be validated against available field test data. The groundwater flow fields will provide a basis for subsequent long term containment studies.

KBS Technical Report No 07

DECAY POWER STUDIES FOR KBS
PART 1 REVIEW OF THE LITERATURE
PART 2 CALCULATIONS

Kim Ekberg, Nils Kjellbert, Göran Olsson
AB Atomenergi, 1977-04-19

SUMMARY

The decay power of irradiated PWR and BWR fuel has been calculated in the decay range of 1-10000 years using the Studsvik computer code BEGAFIP and the ORNL code ORIGEN. Vitrified waste from reprocessed fuel - with 99.5 % U and Pu eliminated - has been followed to 1000 years. The two programmes show reasonable agreement for fission products, but ORIGEN is some 30 % lower for actinides. Recommended values, based also on complementary results from the depletion code CASMO, are given for unprocessed fuel and for vitrified waste, assuming reprocessing either 2 or 10 years after discharge. The recommended values are conservatively chosen and are based on ORIGEN fission product results and BEGAFIP actinide results. An error analysis is given.

KBS Technical Report No 08

LEACHING OF FRENCH, ENGLISH AND CANADIAN GLASS CONTAINING HIGH-LEVEL WASTE

Göran Blomqvist, AB Atomenergi, 1977-05-20

Summary

A survey has been made of typical leach rates for French and English glasses used for incorporating highly radioactive waste. For French glass, calculations have been made to find the time- and temperature tendencies of the leach rates of Cs and Sr in range 25 - 110^{o}C using published values for the diffusion coefficients up to 70^{o}. For the higher temperature range, the diffusion coefficients have been calculated according to the formula

$$D = D_o . e^{-\frac{E_D}{RT}}$$

For the time dependence it has been assumed that the leach rate is proportional to $\sqrt{Dt}$.
According to the calculations, leach rates of French LWR glasses are significantly less temperaturedependent than English glass for Magnox waste. Canadian nepheline-syenite glasses have leach rates several orders of magnitude lower than other glasses. This is probably at least partly due to the much higher formation temperature.

KBS Technical Report No 09

DIFFUSION OF SOLUBLE MATERIALS IN A FLUID FILLING A POROUS MEDIUM

Hans Häggblom, AB Atomenergi, 1977-03-24

Summary

The physical and mathematical foundations of diffusion in porous media are explained. Equations for diffusion of a radioactive nuclide chain are derived. The program DIWA for computing one-dimensional diffusion of one nuclide is described and results from such calculations are given. An underground deposition of wastes was assumed at a deep of 500 m. For a pessimistic assumption about the diffusion coefficient, no decay and no adsorption, the discharge at the ground level was negligible for at least 70000 years.

KBS Technical Report No 10

TRANSLATION AND DEVELOPMENT OF THE BNWL-GEOSPHERE MODEL

Bertil Grundfelt, Kemakta Konsult AB, 1977-02-05

Introduction.

The rate of radioactivity discharge from a repository for radioactive waste in a geologic formation to the biosphere is a very important variable in the evaluation of the safety of the ultimate disposal. A mathematical model of the migration of radioactive species becomes rather complicated if effects as chain decay, chemical sorption and dispersion are taken into account. This necessitates a proper tool for the calculations. Such a tool has been developed by Batelle Pacific Northwest Laboratories, USA, in the form of a BASIC language computer program called GETOUT (1). This program was obtained by the Swedish utilities' project "Nuclear Fuel Safety" in December 1976 and has thereafter been translated into FORTRAN.

The program calculates the discharge rate of radioactive nuclides at t top of a unidimensional column filled with an ionexchanging material assuming that the waste is situated at one end of the column and that the groundwater flows at a constant rate through the column.

The main extension of the code, that was made during the translation, a model for averaging the hydrodynamic and geochemical parameters for ›e case of non uniform packing of the column (e.g. considering a repository in cracked rock with crack width, crack spacing etc. in different zones).

2. Description of the mathematical model.

2.1 The Transport Equation.

The model is based on a differential mass-balance with a dispersion term which in dimensionless form can be written as follows for the i:th nuclide in a decay chain:

$$K_i \frac{\partial N_i}{\partial \Theta} = \frac{1}{Pe} \frac{\partial^2 N_i}{\partial \eta^2} - \frac{\partial N_i}{\partial \eta} - K_i R_i N_i + K_{i-1} R_{i-1} N_{i-1} \qquad (1)$$

where: N_i = number of atoms of nuclide i
Θ = dimensionless time = $t \cdot u/L$
  t = time (s)
  u = ground water velocity (m/s)
  L = column length (m)
η = dimensionless length coordinate = Z/L
  Z = length coordinate (m)
Pe = Peclet's number = $u \cdot L/D$
  D = dispersion coefficient (m^2/s)
K_i = nuclide retentivity = u/u_i
  u_i = nuclide velocity for nuclide i (m/s)
R_i = decay number for nuclide i = $\lambda_i \cdot L/u$
  λ_i = decay constant for nuclide i (s^{-1})

Eq. 1 has been solved analytically for two sets of boundary conditions (2) namely instant dissolution of the waste (impulse release) and constant dissolution rate (band release). The solutions of eq. 1 are based on the following assumptions:

1) Constant ground water velocity along the whole migration path.

2) Constant axial dispersion coefficient along the whole migration path.

3) As the model is unidimensional the radial dispersion is neglected.

4) Low enough ground water velocity to allow the migrating species to be in equillibrium with the surrounding geologic medium.

The assumptions 1 and 2 deviate significantly from the flow pattern in Swedish cracked rock. A model for averaging the hydrodynamic and geochemical parameters has therefore been included in the code in order to make it useful in the safety evaluation of a rock repository.

2.2 Averaging hydrodynamic and geochemical parameters.

The migration is governed by three parameters i.e. the ground water velocity, the dispersion coefficient and the nuclide retentivities. These are the parameters that are averaged in this model.

The model is based on the properties of the recidence time distribution (RTD) in the dispersion model. The RTD can be looked upon as a statistical frequency function expressing the probability of a volume element of the fluid having a specific recidence time. For small amounts of dispersion the RTD becomes Gaussian while for greater amounts of dispersion the RTD becomes increasingly scewed. The RTD can be written as eq. 2 with a mean and a variance according to eq:s 3 and 4.

$$C_\Theta = \frac{1}{2\sqrt{\pi\Theta/Pe}} \cdot \exp\left[-\frac{(1-\Theta)^2}{4\Theta/Pe}\right] \qquad (2)$$

$$\bar{t}_C = (1 + \frac{2}{Pe}) \frac{L}{u} \qquad (3)$$

$$\sigma_\Theta^2 = \frac{2}{Pe} + \frac{8}{Pe^2} \qquad (4)$$

where: $\bar{t}_C$ = mean recidence time (s)

σ_Θ^2 = variance (dimensionless)

The mean recidence time is more useful in solving the equations and has therefore been presented above instead of the dimensionless mean of the RTD $(= \bar{t}_C \cdot u/L)$.

The migration path can be described as a number of zones with different ground water velocities, dispersion coefficients and nuclide retentivities. For each of these zones the mean recidence time and the variance are calculated from eq:s 3 and 4. The mean recidence times are added up to a total mean recidence time and the variances are added up to a total variance. The totals are thereafter inserted into eq:s 3 and 4 together with the total migration path length and the equations are solved for the ground water velocity and the dispersion coefficient.

The nuclide retentivity is calculated by first calculating the nuclide velocity following the procedure described above. According to the definition the retentivity is calculated by dividing the ground water velocity by the nuclide velocity.

The computer code can accept up to 10 zones.

3. Input description.

The input requirements for a sample run with the FORTRAN version of GETOUT is listed below.

1) Source strengths.

   At least one file with the nuclide inventory in curies of the waste is needed. The inventory should be listed in the file for times corresponding to the time of leach incident used in the calculations. The file also contains the halflife in years for the nuclides.

2) Time of leach incident.

   The lifetime of the waste cladding is expressed as the time of leach incident and measured in years after reactor discharge.

3) Leach rate.

   The rate of the waste dissolution is expressed as the leach duration in years.

4) Migration path length.

   The total migration path length is expressed in meters.

5) The nuclides to be calculated.

To obtain maximum flexibility of the code the nuclides to be calculated are read in from cards.

6) Data for the hydrodynamic zones.

For each of the zones mentioned in chapter 2.2 the following data are read in:

a) permeability (m/s)
b) crack spacing (m) alternative porosity
c) hydrostatic gradient (m/m)
d) zone length
e) a logical variabel that is true if the porosity is to be used in the velocity calculation and false if the crack spacing is to be used
f) nuclide retentivities.

7) Diffusion coefficient.

A diffusion coefficient that is considered as representative for most of the nuclides is read in expressed in sq.meters per second.

4. Computer implementation.

The FORTRAN version is written in IBM FORTRAN IV and has been outtested on an IBM model 360/75 computer. The execution time needed is about 30 seconds, compilation excluded. The storage request of the intire code is about 208 K bytes. If a linkage program with an overlay feature is available the storage request can be reduced to about 104 K bytes.

KBS Technical Report No 11

STUDY OF SUITABILITY OF TITANIUM AS CORROSION-RESISTANT CLADDING FOR NUCLEAR FUEL WASTE

Sture Henriksson, AB Atomenergi, 1977-08-24

Summary

A literature study and inventory of experience has been carried out, aimed at assessing the possibilities of unalloyed and Pd-alloyed titanium withstanding corrosion for 1 000 - 10 000 years in contact with Baltic Sea water at 100°C and pH 4 - 10. In fact the risk of Baltic Sea water coming into contact with the canister is considered to be minimal; this implies that the actual environment on final deposition of the canister is in all probability much less aggressive than is considered here, nor should the temperature exceed 80°C.

Based on the original assumptions the following assessment can be made: -

1 Pitting, crevice corrosion, stress corrosion cracking and corrosion fatigue constitute no problem if the canister is made of unalloyed titanium corresponding to ASTM Grade 1. Titanium alloyed with palladium therefore need not be used.

2 Linear extrapolation of reported corrosion rates for oxidation and general corrosion gives a life of between 1 000 and 10 000 years for a 5 mm thick canister. This estimate must be considered to be conservative since oxidation in fact follows a logarithmic law.

3 Hydrogen embrittlement resulting from hydrogen pick-up from the deposition environment should not occur. Delayed failure caused by a redistribution of the hydrogen initially present in the titanium can be avoided if its concentration is maximized to 20 ppm. Pd-alloyed titanium is more sensitive than unalloyed titanium to hydrogen pick-up, especially in galvanic contact with less noble metals.

KBS Technical Report No 12

EVALUATION OF PROPERTIES AND FUNCTION OF CONCRETE IN CONNECTION WITH FINAL STORAGE OF NUCLEAR FUEL WASTE IN ROCK

Sven G Bergström, Göran Fagerlund, Lars Rombén
The Swedish Cement and Concrete Research Institute,
1977-06-22

SUMMARY

This report deals with the possibility of using concrete in conjunction with the permanent storage of nuclear fuel waste in rock storage facilities. The emphasis has been placed on properties such as strength and tightness and how these may be affected by internal and external causes of destruction during a filling stage of approximately 100 years and during the final storage stage of 1 000 - 100 000 years.

The report begins in Sections 1 and 2 with a presentation of concrete as a material. A survey of the physical and chemical properties of the constituent components and of the hardened concrete is provided here. Section 1 provides information on non-aged concrete with regard to structure, strength, water tightness, long-term deformations, temperature movements as well as information on hydration reactions and hydration products. Section 2 deals with the effects of time on the structure and properties of concrete. Experience obtained from antique buildings constructed with similar materials and results from experimental long-time investigations are presented here. The question of the inner stability of the reaction products is dealt with and it is established that a conversion to more highly crystallized products can be expected to take place with time and particularly in conjunction with raised temperatures. This will lead to an increase in porosity. The concrete can

also be aged due to external influences. A survey is provided of the influence of physical and chemical environmental factors, as far as these can be established from experience over normal time periods.

Section 3 contains a brief description of the environment in which the concrete is to be used.

Section 4 contains an assessment of the properties and function of concrete in this environment with regard to various conceivable causes of destruction.

It is established that spontaneous structural changes, which lead to a certain increase in porosity, cannot be precluded during the filling stage and during the final storage stage.

It is deemed possible to avoid cracking during the manufacture and during the filling stage if the concrete is kept moist. The risk for cracking during the final storage stage is difficult to assess.

Attempts are made to estimate the tightness of aged concrete during the various stages. The tightness during the final storage stage is difficult to assess due to the fact that the scope of the cracking cannot be estimated.

Chemical attacks during the filling stage are deemed to be small and can be repaired.

Assessments of the maximum attacks, based on attacks caused by Baltic Sea water and on certain assumptions concerning the porosity and permeability of the rock, are made for the final storage stage. A sulphate attack from Baltic Sea water is deemed to be improbable but even if it should occur, a maximum depth of attack of approximately 100 mm after 1 000 years and 700 mm after 10 000 years can be assumed. The corresponding values are 5 mm and 15 mm respectively for calcium-leaching attacks and similar values can be expected in conjunction with a base exchange attack by the magnesium content of the seawater. These values indicate that it should be possible to design the construction in such a way that its loadbearing function is not lost. The sealing along the rock wall is, however, a weak point.

The risk for destruction due to radioactive radiation is extremely small.

The strength of the sections not chemically attacked should, in the main, remain unchanged during the filling stage. The structural change and other long-term effects may entail considerable losses in strength during the final storage stage. Despite this, the concrete should function monolithically and retain a certain residual strength.

Reinforcement, if any, can be protected during the filling stage on condition that the concrete is kept saturated but all reinforcement will be destroyed during the final storage stage.

By way of conclusion, a number of general views on the choice of concrete and work methods are provided.

KBS Technical Report No 13

LEACHING OF SPENT NUCLEAR FUEL (IRRADIATED URANIUM OXIDE) FOLLOWING DIRECT DEPOSITION

Ragnar Gelin, AB Atomenergi, 1977-06-08

Summary

As a part of the KBS program to evaluate the radiological hazards of storing irradiated fuel in geological formations the literature of leaching irradiated LWR fuel in water has been studied.

There seems to have been made very few relevant experimental studies.

Leach tests are being performed at Batelle-Northwest, Richland, US and some of the results have been published. These results and conclusions are summarized and discussed.

The relative leachability of the elements decrease in the order of

$$Cs > Sb > Sr + Y > Pu > Cm$$

The cesium based periodic leach rate for irradiated fuel fragments are similar to the cesium based leach rate for borosilicate glass containing radioactive waste.

KBS Technical Report No 14

INFLUENCE OF CEMENTATION ON THE DEFORMATION PROPERTIES OF BENTONITE/QUARTZ BUFFER SUBSTANCE

Roland Pusch, Luleå Institute of Technology, 1977-06-20

The nature of silica solution and precipitation is not known in detail. The chemical environment, temperature, pH and ion strength are known to be controlling factors which combine to make possible alternating solution and precipitation of silica. Simultaneous solution and precipitation, that is the case considered to be a possible process in the buffer substance, can produce cementation as indicated by MARZOLF's (1976) surface texture studies of Navajo sandstone. This may be the cementing process when ground water percolation is negligible. However, as shown by the case survey and the presented theoretical treatment the amount of precipitated SiO_2 will not be able to produce a brittle behaviour of the buffer mass even after thousands of years.

KBS Technical Report No 15

PRELIMINARY TEMPERATURE CALCULATIONS FOR THE FINAL STORAGE OF RADIOACTIVE WASTE IN ROCK, REPORT 2

Roland Blomqvist, AB Atomenergi, 1977-05-17

SUMMARY

Introductory calculations of temperatures in a rock storage for final disposal of radioactive waste have been carried out since February 1977. Calculations performed until about the 15th of March have earlier been presented in KBS Technical Report 05. The now compiled report give details about further calculations up to about the 15th of May.

The first part of this report deals with an extension of earlier parametric studies regarding disposal of highly radioactive glass in vertical holes deeply in hard rock. The extension mainly deals with how the temperature of the waste is influenced by the active length of the holes, the diameter of the waste containers and the concentration of fission products in the glass. Besides is also shown some results of calculations regarding an actual storage geometry in 4 levels.

The report also presents a parametric study of a final storage containing non-reprocessed spent fuel. In this case the waste containers are assumed to be placed horizontally along the center lines of horizontal tunnels filled with a mixture of clay and sand. The storage is alternatively assumed to have one or five levels. By assuming several different values for time of disposal, size of waste containers, distance between tunnels, distance between containers and thermal conductivity of clay-sand mixture the influence of those parameters on the container temperature is shown.

KBS Technical Report No 16

REVIEW OF THE FOREIGN RISK ANALYSES AND PLANS AND PROJECTS CONCERNING FINAL STORAGE

Åke Hultgren, AB Atomenergi, August 1977

SUMMARY

Risk analysis as an instrument for systems safety evaluation and the calculation of consequences from various accidents related to the probability of their occurrence is under rapid development. Risk analysis of the back end of the fuel cycle is now being given increasing efforts in several nuclear power countries. A review of the major programmes abroad in this field, especially for terminal storage of high level nuclear waste, is given in the first part of this report.

The second part of the report reviews major projects and plans for terminal storage in America and in Western Europe, with a brief reference to co-operation in international fora. The most comprehensive programme is in progress in the United States. For Sweden it seems that also the programmes in Canada and France are of particular interest due to their concentration on terminal storage in crystalline rocks.

KBS Technical Report No 17

THE GRAVITY FIELD IN FENNOSCANDIA AND POSTGLACIAL CRUSTAL MOVEMENTS

Arne Bjerhammar, Stockholm, August 1977

We have analysed all availabe information in a study of the Fennoscandian crust with respect to present movements in the crust. The main object of our investigation has been to discriminate between tectonic and isostatic movements.

We have analysed all the evidence we have found in favour for the tectonic hypothesis and we found little support for this alternative. In fact, there has been practically no proof for the existence of a major tectonic movement in the Fennoscandian crust. Small tectonic movements cannot be excluded, but present observations are not sufficient for a conclusive statement.[x)]

The main interest of our study has been devoted to the analysis of the gravity field with the use of satellite observations and terrestrial gravity observations. This study has revealed that there is a very strong support for the hypothesis that the present vertical movements in the crust have an isostatic origin. We find it also justified to conclude that the movements are mainly of glacioisostatic origin.

x) Records from Rheingraben indicate a tectonic uplift of maximum 1.7mm per year. The seismic activity in this area is stronger than in Fennoscandia and we estimate the present tectonic uplift in Fennoscandia to be below 1 mm per year!

This study has not considered the tidal movements in the crust.

The result of the analysis seems to imply that no alarming tectonic movements in Fennoscandia can be traced from the records now studied.

The problem of the optimal choice of a site for waste disposal inside the Swedish crust is very intricate and must be considered from a number of different points of view.

If radioactive waste has to be disposed in the crust, then the following priorities will be given.

1. Salt mines.
2. Permafrost-areas.
3. Selected areas, which are 'stable' from geodynamical point of view.

The two first priorities are fairly equivalent with respect to the environmental problems. The waste disposal should not endanger any life for any estimable time span. However, the first alternative is excluded in Fennoscandia. The second alternative is a possible alternative for some rare locations in northern Sweden. There is a practical problem with the heat dissipation, which means that the final disposal must be postponed considerable time (several hundred years).[x)] Meanwhile, a temporary waste disposal can be arranged above sea level in dry locations.

The third alternative has the lowest priority, but is the most convenient for the user. If this alternative is chosen, then it is recommended that an advanced geodynamical study is made over a time span of at least 10 years for the selected sites. Crustal studies of the horizontal and vertical movements should be made. The seismic activity in the neighbourhood of the sites should be studied. Elastic, viscoelastic and viscous parameters should be determined with great care. The final sites should be chosen in areas where the isostatic relaxation has been more or less completed.

Finally, it should be noted that large postglacial horizontal movements in Fennoscandia have recently been recorded by Lagerbeck and Lagerlund. These findings might indicate that the crustal movements are more complex than earlier anticipated.

x) Eventually restricted to transuraniums, in several small containers.

KBS Technical Report No 18

MOVEMENTS AND INSTABILITY IN THE SWEDISH BEDROCK

Nils-Axel Mörner, University of Stockholm, August 1977

PREFACE

First we trew garbage into the forest and believed that it hid everything and that no damage was made. Later we drained sewer water and chemicals into lakes, rivers and the sea and believed that it disappeared in this "enormous" recipient - it costs miljons every year now to try to restore better limnic and marine environments. We are now facing the final disition of storing nuclear waste in the bedrock under the prospect that the bedrock is "stable". The principle to hide and forget goes on but with all the more terrible material: from garbage in the forest, via sewer and chemicals in rivers, lakes and the sea, to nuclear waste in the bedrock. Of course, it is madness to put nuclear waste in containers in the bedrock, seal off and believe that nothing happens with the containers during centuries and millennia to come.

Like in the case of the Vietnam war, the problem can be tackled in two ways:

(1) directly emotionally - then the answer must be that it is madness, or

(2) via political/technical evaluations (where the key point almost immediately is pushed aside) - then the answer may be both positive and negative.

It is quite clear that it is basically madness to put nuclear waste into the bedrock - in order to hide and forget. As we, however, seem to live in a world that to a great extent is ruled by madness, I will give an objective geological evaluation of the Swedish bedrock and its movements during the last 20,000 years, which may serve as a base for further evaluations of the possibilities of storing nuclear waste in the bedrock.

SUMMARY

(1) The Swedish bedrock is by no means "stable". Like all other bedrocks it is unstable.

(2) The Swedish bedrock has an old and rich tectono-geodynamic inheritance.

(3) The total uplift is about 830 m, 725 m of which is caused by the parabolic purely glacial isostatic factor, which died out some 2000–3000 years ago (the astenosphere restored its old position).

(4) The maximum monentary rate of uplift reached 50–5 cm/yr during a short period at around the time of deglaciation or the end of the Younger Dryas Stadial.

(5) The linear factor in the uplift seems to have been induced at about 8000 BP and be caused by global cyclic changes of the geoid. The bend in the West Coast shoreline profile was formed by this change.

(6) Irregularities in the uplift in the form of shoreline bends and isobase irregularities have been established with ancient shorelines and geodetic data. They are in general all related to major faultlines and bedrock seams.

(7) Faulting, fracturing and seismic activity was shown to be linked to the deglaciation period (the maximum rate of uplift) and to be fairly frequently occurring.

(8) Major faultlines are generally related to old weak zones. Small faultlines (up to 2 m vertical displacement) and fracturing of the bedrock surface, on the other hand, are totally independent of these zones.

(9) Bouldery end moraines and bouldery ground in general register paleopseismic activity – (these areas must hence be excluded as alternatives for storage of nuclear waste in the bedrock).

(10) No extrapolation of presently measured mean values (e.g. for the seismic activity during the last decades) and no future predictions at all can be made beyond the next ice age.

(11) The next ice age is either on its way or it will, under the most favorable circumstances, have begun 20,000 years from now (AP).

(12) At the next ice age, all the seismic and neotectonic effects from the deglaciation period will be repeated.

(13) During an ice age, nuclear waste cannot be stored in the bedrock.

(14) The linear uplift factor is at each cyclic turning point likely to be linked to the same effects as those which were recorded for the period of about 8000 BP (and maybe also those linked to the peak rates in the uplift).

(15) If one succed in finding a Precambrian bedrock unit within an area of smooth uplift, absence of recent earthquakes, the bedrock surface of which shows few fractures and no faultlines, and where the surroundings exhibit normal moraine features and normal till composition, this area must still be evaluated with respect to that which will happen and may happen in connection with the next ice age and in connection with the cyclic gravitational changes in the present linear uplift.

KBS Technical Report No 19

STUDIES OF NEOTECTONIC ACTIVITIES IN CENTRAL AND NORTHERN SWEDEN, REVIEW OF AERIAL PHOTOGRAPHS AND GEOPHYSICAL INTERPRETATION OF RECENT FAULTS

Robert Lagerbäck, Herbert Henkel
Geological Survey of Sweden, September 1977

Summary

Several fault-lines of presumed late-glacial age in northern Sweden are described (fig. 2). The faults have been identified and investigated mainly by means of air-photo interpretation. Morphologically the fault scarps are very conspicious, the general appearance being a marked step in the till cover, which can be traced over long distances (fig. 3, 4).

The relation between fault-lines and late-glacial phenomena shows that movements occurred in late-glacial time in connection with the deglaciation of the area.

The faults are all developed in the Precambrian. Often the orientation of the faults are clearly influenced by the structures of the bedrock; e.g. gneissosity, and boundaries between different rock units. A comparison with aero-magnetic maps shows that the Quaternary movements frequently coincide with older fault zones.

The amount of displacement on the dislocations is of the order of up to about 30 metres and the most extensive fault-line is about 150 km long. Down-dip slickensides and local irregularities in the fault lines suggest only dip-slip movements.

The fault-lines have a general NE-SW trend parallel to the Caledonian mountain chain and to the seismically active zone along the Swedish coast of the Gulf of Bothnia.

The synchronism between the deglaciation of the region and the fracturing offers a natural explanation of the phenomena. The isostatic rebound after the glaciation apparently did not take place simply as a regular updoming. At least in the case under consideration, breaks in the Precambrian crystalline bedrock occurred, resulting in differential movement. The direct reason for the close connection between deglaciation and tectonic movements may have been that the deglaciation proceeded rapidly. A comparatively sudden release of the pressure on the earth's crust may be the reason for faulting instead of slow updoming. The occurrence of the fault-lines suggests that fault movement plays a part in the post-glacial, regional uplift of Scandinavia.

The regional connection between the neotectonic structures and the recent seismic pattern within the area is obvious (fig. 2). This indicates that the forces which produced the faulting are still active. Nevertheless it is evident that the faulting was more active in late-glacial times than today. Although there are no records, as yet, of movements after late-glacial time, the possibility of present activity within the fault zones should be investigated.

KBS Technical Report No 20

TECTONIC ANALYSIS OF SOUTHERN SWEDEN, LAKE VÄTTERN - NORTHERN SKÅNE

Kennert Röshoff, Erik Lagerlund
University of Lund and Luleå Institute of Technology,
September 1977

SUMMERY

This report gives the conclusions of an investigation in southern Sweden and includes the description, the analysis and the valuation of

1. three different lineament methods for the representation of the tectonic pattern
2. the tectonic pattern in the area studied and the quantifiation of tectonic reactivation
3. neotectonic activity

The area studied is depickted on Fig 1.

Three lineament methods are tested namely morphological lineament (1:50 000), airial photographs (1:33 000 and 1:66 000) and satellite photographs.

It is found that in a test area no single method will fully represent and depicked the tectonic pattern as described in the geological litterature. Several factors have to be considered of which the following are the most important; the scale factor, the morphology, frequency, length and clearness of a single lineament. Therefore different and combined lineament analysis are required for an acceptable result.

The area analysed (Fig 1) is subdivided into 7 subareas depending on the character of the morphological lineaments (Enclosure 1). Table II gives the order of importance of the lineaments within each subarea, based on a subjective judgement of the lineaments from the three methods.

A description of the tectonic pattern as outlined in the litterature within 10 subareas is given in Figs 5-20. The main tectonic zones are seen on Fig 4. The studied area consists of two main blocks bounded by the main tectonic zones. The eastern block is bordered to the north by the E-W faults in Östergötland, the N-S fault zone from lake Vättern to Skåne, the NW-SE fault zone through Skåne and a probable fault to the east

through Kalmarsund. The western block is bounded to the east by an approximate N-S fault zone from Skåne to Värmland, the NW-SE fault zone through Skåne and to the west a marked fault zone in the sea along the Westcoast.

Within the eastern block the tectonic pattern has main directions in N, NW and WNW. Its southern parts are more complicated with several important directions.

The western block has main directions in N, NE and ENE.

Fig 25 gives the directions, approximate frequency and order of importance of the tectonic pattern, based on the lineament analysis and geological data.

In order to quantity the reactivation of old tectonic zones during a deformation a special study was undertaken with OPAB:s seismic material as base data. The result is presented in Figs 21-24 and Table I, and shows that old tectonic zones were reactivated between 60-100 %. This study also gave the possibility to calculate the deformation rate along two faults (Appendix 1 p.4)

No important neotectonic activity is observed by the method used. However, the quarternery beds, as analysed at a scale of 1:33 000, display a lineament pattern, which corresponds to that found in the underlying hard rock, even if the quarternery beds are thick (> 10 m). These lineaments thus may have neotectonic origin.

Enclosure 11 and 12 give the results of a lineament study at scale of 1:66 000. Only lineaments, which can be traced from hard bedrock into the quarternery beds, are depicked. Those with a F have special interest. Field investigations are here recommended.

KBS Technical Report No 21

EARTHQUAKES OF SWEDEN 1891 - 1957, 1963 - 1972

Ota Kulhánek, Rutger Wahlström
University of Uppsala, September 1977

A general brief description of seismic activity in Sweden is given. Geographical distribution of epicentres, seismicity, magnitudes of Swedish earthquakes and their relation to source dimensions are included.

For the time period 1891-1957 macroseismic data, including 110 events with magnitudes M ≥ 3, are employed. The largest earthquake from this ensemble is that of March 9, 1909 (M = 5) which occurred in the northern Gulf of Bothnia. A map of macroseismic epicentres gives an indication of areas with higher seismic activity in south-western Sweden including also the southern Baltic Sea and in the eastern coastal area along the Gulf of Bothnia. A diffuse seismic area may be traced in northern Norrland. Areas with practically no seismic activity are broadly the western part of the country, say, north of 61°N and the area of south-eastern Sweden.

During the first years of the 1960's the seismic network in Sweden expanded so that it was possible to locate regional events from Swedish seismograph readings. The time interval 1963-1972 is used. Epicentral coordinates are given with an accuracy of ±10 km and times with an accuracy of ±1 s. 73 earthquakes with magnitudes $M_L \geq 2.0$ were identified and located. The geographic distribution of instrumentally determined epicentra differs somewhat from that determined macroseismically. A belt with higher seismic activity starts at the area of lake Vänern and extends in the NE direction to the northern Gulf of Bothnia. Another distinct active region can be seen in the northern Norrland. The aseismic areas resemble more or less those determined from macroseismic data.

Source dimensions are estimated by means of formulae derived empirically for California. Differences and limitations with respect to earthquakes in Sweden are discussed.

KBS Technical Report No 22

THE INFLUENCE OF ROCK MOVEMENT ON THE STRESS/STRAIN SITUATION IN TUNNELS OR BOREHOLES WITH RADIOACTIVE CANISTERS EMBEDDED IN A BENTONITE/QUARTZ BUFFER MASS

Roland Pusch, Luleå Institute of Technology, 1977-08-22

The reports gives the author's main ideas concerning the possible occurrence of large unexpected movements in Swedish pre-Cambrian rock and the theoretical basis for the calculation of stress and strain in the canisters and the buffer mass.

The procedure comprises the assumption of a reasonable regional stress situation, a superposition of a stress change leading to general failure of a large rock volume, and a determination of the associated deformation pattern.

A condition of general failure is shown to require a largely increased lateral stress in the upper earth crust. Applying MOHR/COULOMB's criterion, the theory of plasticity, and plausible values of the cohesion and friction parameters, it is shown that fairly good granitic rock with a moderate joint frequency is very far from a critical condition at depth of about 1000 m. This also goes for very bad fissured rock. For continuous, very weak zones (shear zones, clayey or chloritic zones) on the other hand, as well as for the upper 100 m of any Swedish bedrock, the conditions may be critical even today. It could mean that displacements are likely to occur any time along a number of pre-existing, easily identified (and probably already identified) large, steep and weak rock zones. It also means that displacements and differential movements within the huge rock blocks situated between these zones are very moderate and that the formation of a slip plane in previously joint-free rock will not take place.

Yet, "the incredible case" of an unexpected shear failure through intact, high quality rock has been considered. It concerned a circular tunnel or bore hole with a centrally located stiff cylindrical canister. A pilot model test was made with a 50 mm "tunnel" and a steel "canister" (70 mm length and 8 mm diameter) embedded in a buffer material of 10% Na-bentonite and 90% Pite silt with a dry density of about 1.4 t/m^3. The device operated as a shear box and the internal displacement pattern was studied by using X-rays which revealed the motion of lead shots applied in the buffer mass. This material was compacted in an air dry condition and artificial ground water was absorbed for about a week to obtain water saturation before the shearing took place.

The X-ray study showed that not until the shear strain approaches about 10% of the tunnel diameter the homogeneity and permeability of the buffer mass is severely changed. The contact pressure distribution could be determined and the application of Meyerhof's and Berezantzev's theories for deeply buried foundations were then used for calculating the canister stresses produced by the displacement. It turned out that these stresses may be very high in stiff canisters and that they can probably not be sustained by copper at the investigated geometrical conditions. Changed canister dimensions can reduce the stresses considerably, however, but this requires additional tests to find out how the displacement is affected by the geometry.

KBS Technical Report No 23

WATER UPTAKE IN A BENTONITE BUFFER MASS
A MODEL STUDY
Roland Pusch, Luleå Institute of Technology, 1977-08-22

Safe deposition of radioactive waste products in metal canisters surrounded by bentonitic buffer masses requires a maximum buffer mass temperature of about 100°C. Sufficient initial homogeneity of the in situ-compacted mass requires a fairly low water content and such a dry condition could yield a critically low heat conductivity λ. The water uptake from the confining rock will successively increase λ but the existing thermal gradient produced by the hot canisters may cause a local low degree of water saturation at least adjacent to the canisters. The report describes two pilot model tests to investigate the degree of homogeneity of the water uptake.

Steel cylinders, 20 cm high and 25 cm in diameter, were filled with 10% Na bentonite and 90% quartz- or quartz/feldspar mixtures to a bulk density of 1.4-1.8 t/m³, the water content being about 5%. A ϕ 50 mm heated model canister was inserted in the center of the cylinders, the surface temperature being kept at 100°C. In one test artificial ground water was brought in contact with the perforated cylinder so that water could enter the buffer mass over the entire periphery. In a second test water could enter the mass through a 5 cm wide slot only.

The test with uniform access to water and with a bulk density of 1.4 t/m^3, was run for about a week after which the cylinder was opened and samples taken for water content determination. It showed that the water content had increased from 5% to 30-37%, the lowest value being found close to the heaters. Even there, however, the degree of water saturation had become as high as about 55%.

The "slot" test in which the bulk density was 1.8 t/m^3 due to a till-like grain distribution of the quartz/ feldspar component, was run for about 3 weeks. Samples taken close to the water inlet showed water saturation degree values of the order of 60-80%, while values similar to those in the first test were found far from the inlet.

One main conclusion from the tests is that no irregularities or piping were found in either test. Also, it was found that the water uptake is a time-consuming process and that speeding up this uptake to get high λ-values shortly after the deposition should be made. This could possibly be arranged by using perforated pipes through which water under pressure is introduced.

KBS Technical Report No 24

CALCULATION OF LEACHING OF CERTAIN FISSION PRODUCTS AND ACTINIDES FROM A CYLINDER MADE OF FRENCH GLASS

Summary

The probable total leaching of the most important fission products and actinides have been tabulated for a cylinder of French HLW glass with approximately 9% fission products. The calculations cover the period between 30 and 10000 years after removal from the reactor. The cylinder is of the type planned for the introduction of the HLW into Swedish crystalline rocks. All the components are supposed to have the same leach rate. The calculations also include the probable thickness of eroded glass layer/year.

KBS Technical Report No 25

THE BLEKINGE COASTAL GNEISS, GEOLOGY AND HYDROGEOLOGY

Ingemar Larsson Royal Institute of Technology
Tom Lundgren Swedish Geotechnical Institute
Ulf Wiklander Geological Survey of Sweden
Stockholm, August 1977

ABSTRACT

The present work deals with the geology and the tectonics of the Precambrian of Blekinge in the southeastern Sweden, with special reference to the relations of the western part of the region. - The Precambrian geology of Blekinge shows a broad chronological development. Five geological units can be recognized, which structurally, petrologically and chronologically are distinguishable from each other: 1, the Coastal gneisses: 2, the Older granitoids; 3. the Younger granitoids; 4, the Youngest granitoids; and 5, the dolerites.

The oldes rock units in Blekinge are the Precambrian supractrustal complex of Västanå and the Coastal gneisses. The former rock sequence includes metasedimentary and metavolcanic rocks. The metavolcanics pass into the surrounding Coastal gneisses. These rocks display a wide compositional and textural variation. Folding, recrystallization and partial migmatization make it difficult to identify the origin of this rock sequence. - The Older granitoid complex makes up a large part of Blekinge. They are in general regionally foliated and/or lineated. - The Younger granitoids occupy the northeastern part of Blekinge. - The Younges granitoids occur in several massifs, consisting of coarse-

grained megacryst-bearing granites - granodiorites (Karlshamn-Erings-boda granites) and small-grained granites (Spinkamåla granite). In contrast to the earlier plutonics, these are accompanied by numerous pegmatites and local migmatization. The contacts against the older plutonics are prevailing intrusive and cross-cutting. - The dolerites are the youngest rocks of the Precambrian in the Blekinge region. They form dikes striking approximately NNE - SSW. - The nature of the meta-morphism of the Coastal gneisses and the plutonic rocks has not been studied in detail. It must be emphasized that the interpretation of the metamorphic relationships within and between the granitoid groups of different age is very complicated, which is due to the effects of super-imposed igneous, tectonic and metamorphic events. The principal mineral assemblage development in the region reflects PT-conditions of the amphibolite facies. K/Ar age determinations on the Older granitoids the Coastal gneisses reflect a strong thermal influence upon the recorded isotopic ages (1 240 m.y. - 1 510 m.y.). These recorded metamorphic ages may be correlated with a period of igneous activity (the Youngest granitoids) occurring approximately between 1 500 - 1 380 m.y. - The Coastal gneisses have undergone several folding phases. The dominant folding phase, producing the dominant regional structures, has N - S fold axis, plunging gently to the north.

The coastal gneiss shows two main structural types: one with a well-developed schistocity (here termed SB-type) and another with typical lineation (here termed BS-type). The two types seem to have been deve-loped during the folding of the gneiss. The mica fabrics (perpendicular to the fold-axis) show the corresponding patterns of planar and acial control.

Local variation in the mica content suggest primary sedimentary structures. Transitions occur between the two types. Therefore the terms SB and BS are used. SB stands for dominating schistose structure and BS stands for dominating lineation.

The BS-type ("B-tectonite") correspond to anticlinal structures with intern/extern rotation. Tha SB-type (S-tectonite) is interpreted as the flanks of the folds with dominating sliding movement during the folding.

Tensile joints, perpendicular to lineation, are very frequent in the B-tectonite areas. In the S-tectonites this type of jointing is strikingly infrequent.

At comprehensive field studies of the tectonics of western Blekinge a tectonic-morphologic map was constructed. Statistical investigations concerning preferred orientation of fractures of the area was carried out. These investigations show two post-crystalline compression stages, l. striking N - 5^{o}E and N 20^{o} - 25^{o}E, respectively.

The fracture pattern of the Coastal gneiss differs from that of the surrounding bedrock areas. The most striking difference is the general lack of clear and distinct vertical tension joints parallel to the compression axis. A vertical joint set perpendicular to the compression axis is prevalent in the Coastal gneiss area and may be interpreted as a set of induced tension joints. Most of these joints are, however, synonymous with the local fracture pattern and probably not open due to the compression. Therefore, they are judged to be small potential aquifers of no regional importance.

It has been observed in drilled wells that the Coastal gneiss as a whole is a poor aquifer. Especially the S-tectonites show a very moderate yield of ground water, as they have very week development of jointing and fracturing. The B-tectonites have a somewhat higher order of magnitude of yield. As a possible explanation to this dryness of the Coastal gneiss it has been suggested that the gneiss has reacted as a very tough geological body against lateral stress in the crust.

In the Coastal gneiss several underground rock storages are constructed. They are in operation and the seepage into the chambers (by pumping) has been observed. A mean value of 0.03 $l/m^3/d$ has been recorded in nine chambers, filled up with oil. Empty caverns are observed to have a seepage of about four time this value.

The figures of seepage in the gneiss caverns may be compared to corresponding figures from a storage plant in middle Sweden. At this place the rock is a coarse, veined-gneiss with pegmatites and slabs of basic rocks. At normal precipitation a figure of 0.5 $l/m^3/d$ has been recorded.

As a conclusion it can be stated that the Coastal gneiss is well suitable for underground construction work.

KBS Technical Report No 26

EVALUATION OF RISK OF DELAYED FRACTURE OF TITANIUM

Kjell Pettersson, AB Atomenergi, 1977-08-25

Summary

The possibility of delayed failure in a titanium canister for radioactive waste has been analyzed. This analysis is based on a Canadian theory for delayed failure of zirconium alloys. Calculations carried out for different hydrogen contents of the material show, that in order to exclude the possibility of delayed failure, the hydrogen content should be kept below 20 ppm. This means that the hydrogen content is permanently below the solubility limit.

KBS Technical Report No 27

A SHORT REVIEW OF THE FORMATION, STABILITY AND CEMENTING PROPERTIES OF NATURAL ZEOLITES

Arvid Jacobsson, Luleå Institute of Technology, 1977-10-03

Zeolites have been suggested as suitable buffer mass components for storing highly radioactive waste products, since certain species have excellent ion exchange properties which are beneficial in the ultimate state when the canisters have corroded and radioactive matter is free to move towards the biosphere. This report deals with their genesis and occurrence as well as their chemical stability and mechanical properties.

The study showed that cementing effects are common in zeolites, which is a serious problem since it eliminates the required plastic, ductile behaviour of the buffer material. Also, zeolites are much too permeable to be accepted for this purpose. If their poor heat conductivity is considered as well, it must be concluded that zeolites are not well suited for the present case of producing a first-class buffer material.

KBS Technical Report No 28

THERMOCONDUCTIVITY EXPERIMENTS WITH BUFFER MATERIAL OF BENTONITE/PITESILT

Sven Knutsson, Luleå Institute of Technology, 1977-09-20

SUMMARY

The investigation reported here concerns the thermal conductivity of the bentonite/quartz buffer mass which has been suggested as embedding substance for radioactive canisters. The first part of the report presents the theoretical relationships associated with the various heat transfer mechanisms in moist granular materials. It is shown in this part that different mechanisms govern the heat transfer when the degree of water saturation is increased from a very low value to 100%. Ø Johansen's method of calculating the thermal conductivity of soils is referred to as well.

Chapter 3 describes the author's experimental determination of the thermal conductivity of the buffer mass. The tested mass consisted of 10% (by weight) bentonite and 90% natural silt. Four tests were made with different water content values and degree of water saturation (S_r).

The results from the tests with a low degree of saturation (5-15%) show that the conductivity was not time-dependent. The same tendency was found in tests where the degree of water saturation was high (73%). However, where the degree of water saturation was intermediate (32%), a time dependence was

observed meaning that the heat conductivity was found to decrease successively. This must be due to water transportation caused by the temperature gradient in the soil, which tends to dry out the mass at the warm side and to increase the water content at the cold side. When the degree of water saturation was low or high no such water movement took place. This phenomenon is explained in the following way. When the water saturation is low the water is very firmly bound to the mineral surfaces and is not readily moved. When the water content increases there will be an increasing amount of free water and, as a consequence, the water will be more easily transported. Thus, a dry zone is created close to the warm side. If the degree of water saturation is high and there is an access of external water the drying is counteracted by a capillary water uptake which brings the water back towards the warm side. Therefore, when the degree of water saturation is high, an equilibrium is established which yields a rather homogeneous water content in the mass. A comparison between the measured and calculated thermal conductivities is given in Fig 17. It is shown that the conductivity can be calculated with an accuracy of ±20%.

To prevent the development of dry zones around the canisters two different initial water contents can be chosen:

1. A low degree of water saturation (less than ~15%). This produces a low value of the thermal conductivity, e g about 0.3 W/m,K when the porosity is 47%.

2. A high value of the degree of water saturation (more than ~74%). This produces a much higher value of the thermal conductivity, e g about 1.7 W/m,K when the porosity is 48%.

KBS Technical Report No 29

DEFORMATIONS IN FISSURED ROCK

Ove Stephansson, Luleå Institute of Technology, 1977-09-28

SUMMARY

Simple shear is a common tectonics in nature. Stress distribution, deformations and failure of a rock mass in direct shear loading is examined. If the strength of the rock mass is in the form of the Coulomb-Navier criterion one set of tensile fractures and two sets of shear fractures arise. The sets of joints predicted by the theory have been used in the mathematical modelling.

Two computer models have been applied to the analysis of the jointed rock mass; the Discrete Block Method (DBM)and the Finite Element Method (FEM). DBM simulates the motion of a set of discrete blocks each one having the possibility of unlimited translation and rotation. Shear forces arise due to friction, damping forces are introduced to dissipate kinetic energy and kontact forces and displacements are determined.

Three blocky rock models have been tested. One model simulates the deformation in a shear box. The two others simulates a regular and unregular jointed rock mass subjected to direct shear. Displacements and total deformation is more flexible as the number of blocks increases for one and the same model.

Finite element analysis of a jointed rock mass is presented. The rigid blocks are assumed to be of linear elastic material and the joints are simulated by means of joint elements. Square blocks of dimension 4 x 4 km are cut by two set of joints and

the block is sheared an angle of 0.5^{0}. In models with a large joint spacing the rigid blocks reach contact in few points under high stresses. Models with small joint spacing show a more gentle and flexible mode of deformation and the stresses are much lower. Models loaded along the edges behave similar.

Results of the DBM and FEM analysis in this study indicate that a suitable rock mass for repository of radioactive waste should be moderately jointed (about 1 joint/m^2) and surrounded by shear zones of the first order. This allowes for a gentle and flexible deformation under tectonic stresses and prevent the development of large cross-cutting failures in the repository area.

KBS Technical Report No 30

RETARDATION OF ESCAPING NUCLIDES FROM A FINAL DEPOSITORY

Ivars Neretnieks
Royal Institute of Technology, 1977-09-28

Summary

A study has been made on retardation of radionuclides in various materials, which could be suited for use in the final repository. A literature survey has shown that except for Cs and Sr very little is known on ion exchange equilibria in ground water surroundings. Measurements were made to determine equilibrium data för Cs, Sr, Eu and U in five natural zeolites, which could be used as filling material. Diffusivities in zeolite particles and beds as well as clay beds were also determined.

The measured equilibrium data could not be predicted by using published data on binary equilibria. Bed diffusivities were of a magnitude that was expected.

With the aid of these data the function of the ion exchange barrier was investigated. The barrier is so short that the nuclide transport is by diffusion. An 0.2 m barrier of a zeolite will delay Cs and Sr so long that they will decay totally. Am^{241} will also be considerably delayed. An 0.2 m clay-quartz barrier will have very little effect on these nuclides. A 1 m clay-quartz barrier will have about the same effect as an 0.2 m zeolite barrier. Most other nuclides have so long lives that they will only be delayed, but not sufficiently long to decay.

The rock itself interacts with many of the radionuclides. A simple model has been made to describe the nuclide retardation and dispersion in fissured rock. With the aid of this, tracer experiments in actual underground rock have been analysed. For Sr at least, laboratory measured equilibrium data on rock, predicted the in-situ runs fairly well. A retardation factor 3-6 was predicted and a value of 6 was measured.

This is expressed as a mean retardation. Due to the fact that some water flows faster than other, the first nuclides arrive at about one fifth of the mean time. This dispersion effect is much larger than what has been observed in sand and similar materials. Using laboratory data, the model and expected water velocities and dispersion, a one km path in the rock would give time enough for Cs and Am^{241} to decay totally. Sr and Am^{243} are considerably delayed. The above must be confirmed by further experiments.

KBS Technical Report No 31

EVALUATION OF CORROSION RESISTANCE OF MATERIAL INTENDED FOR ENCAPSULATION OF NUCLEAR FUEL WASTE. STATUS REPORT, 1977-09-27, AND SUPPLEMENTARY STATEMENTS

Swedish Corrosion Research Institute and its reference group

Summary

Within the KBS project the Swedish Corrosion Institute has got the task to evaluate the corrosion resistance of different materials proposed to be used in canisters for nuclear waste. For this purpose the Institute appointed a reference group of specialists, mainly within the fields of corrosion and materials.

KBS has proposed three different alternatives of canisters for evaluation of which two are made of metallic and one of ceramic materials. The canisters will be buried deep in crystalline rock and surrounded by a buffer material consisting of quarts sand and bentonite. The corrosion environment has been specified by KBS, although some remaining uncertainties will be further investigated. This status report from the Swedish Corrosion Institute is supported unanimously by the reference group. Supplementary comments have been given by some of the group members.

Titanium lined with lead as a canister material for reprocessed and vitrified waste

The corrosion resistance of titanium is depending on its capability to maintain a protective oxide film which is selfhealing in the case of accidental damage. The 6 mm thick titanium casing is lined with 100 mm lead which is a complementary protection against corrosion penetration and is thought to increase the life of the canister considerably. With the assumptions and knowledge we have to-day the life of the leadlined titanium canister will be at least 1000 years according to some members of the reference group or at least 500 years according to others. To minimize the corrosion risk the site and the arrangement of the deposition should be chosen so that the canister will not be exposed to ground water with extreme contents of salt.

Copper as a canister material for direct disposal of spent fuel

Copper is a comparatively nobel metal, stable in pure water free from oxygen. The amount of oxygen and other oxidizing agent produced by radiolyzis is very low in ground water surrounding the canisters. Thus a copper canister with a wall thickness of 200 mm is expected to have a very long life - at least 5000 years. There is, however, some uncertainty whether sulphate and/or sulphide in combination with bacteria are able to affect the corrosion of the copper canister. Such an effect, is, however, not very likely - due to the very low content of organic matter - but it should be further investigated.

Alumina as a canister material for direct disposal of spent fuel

The KBS-project has also proposed a canister in a ceramic material (alumina) manufactured by a method developed at ASEA for evaluation by the Institute. If such a canister can be produced in a material of satisfactory purity and quality it is likely to last for a very long time. Before the final evaluation it is desirable with further investigations of corrosion rates under adequate environmental conditions, especially with regard to the risk of delayed fracture.

General comment

For further evaluations the Institute points out the need of complementary investigations, e.g. on variations in ground water composition at a depth of 500 m, especially the content of oxygen, chloride, nitrite, sulphate and organic matter. Furthermore, the environment around the canister during the period following the deposition has to be specified. Among other things the effect of water evaporation and salt concentration on corrosion ought to be considered.

KBS Technical Report No 32

PROPERTIES OF BENTONITE-BASED BUFFER MATERIAL

Roland Pusch, Arvid Jacobsson
Luleå Institute of Technology, 1978-06-10

Summery

This report shows results from investigations concerning properties in bentonitebased buffersubstances which are suggested to be used when high level radioactive wastes from nuclear powerplants are to be stored finally. Recommended material characteristica of the bentonite to be used are summerized in tables 20 - 21.
In an attempt to find geological evidence for bentonite to loose its desireable properties there were no such findings at the temperatures, groundwater situations and pressures which are to be expected at the actual depositing depth (500 m) for a considerable period of time.
Concerning biological activity and then specially the mobility and activity of bacteria the conclusion is that there will be little or no influence from them either there is bentonite/sand or compacted pure bentonite in the buffer mass.

KBS Technical Report No 33

REQUIRED PHYSICAL AND MECHANICAL PROPERTIES OF BUFFER MASSES

Roland Pusch, Luleå Institute of Technology, 1977-10-19

Conclusions

So far all evidence given in the current investigation by the KBS buffer group shows that a combination of Na-bentonite and silt/sand-sized quartz components gives a buffer mass with optimum properties. The physical, physico/chemical, and mechanical properties of such masses are sufficiently well-known to allow us to use them in this very serious context. Furthermore, the handling of the masses at the deposition involves no new or unknown technique, only moderate changes or improvements have to be made. Thus, the practical deposition work will very probably be successful as far as the buffer mass concerns.

As to the bentonite/quartz ratio it can be stated that the bentonite should form at least 10 weight percent of the solid mass. This is where the bearing capacity or canister- or rock-supporting power is important. In other parts of the bore holes, tunnels and shafts the bentonite content can be increased to 20 or even higher percentages. The final choice of this ratio is considered to be a matter of design with special reference to required temperatur limitations and aspects which deal with the properties of the mass when transported, compacted etc.

KBS Technical Report No 34

FABRICATION OF LEAD-TITANIUM CANISTER

Folke Sandelin AB, VBB, ASEA-Kabel,
Swedish Institute for Metals Research
Stockholm, November 1977

Summary

The report describes a proposed design for a titanium-lead canister and its manufacturing process. This process is based on well-known technology. The strength of the canister has been considered ant it is evident that it will withstand the hydrostatic water pressure in the final storage.

The lead material and its behaviour are discussed. It is shown that high purity lead and an extruding process for the manufacturing of the lead shield can be used to obtain a high quality product.

The extruding equipment is described. The costs of the canister is estimated at 40 000 Swedish Crowns excluding the encapsulation work (which will be performed in a "hot cell") and the costs for the extruder.

KBS Technical Report No 35

PROJECT FOR THE HANDLING AND STORAGE OF VITRIFIED HIGH LEVEL WASTE

Saint Gobain Techniques Nouvelles, October 1977

SUMMARY

In this report the result of a study made by Saint Gobain Techniques Nouvelles for KBS is presented. The study was to propose methods for the intermediate storage, encapsulation and the final disposal of vitrified high level waste.

The waste is proposed to be encapsulated in a canister made of lead and titanium with a wall thickness of 10 cm for the lead and of 6 mm for the titanium. The lead provides radiation shielding whereby the radiolysis of the ground water is reduced to such a low level that its effect on the corrosion of the titanium is negligible. The titanium provides long term corrosion resistance. The encapsulation will take place in a hot cell.

Prior to the encapsulation and final disposal the vitrified waste is kept in an intermediate, air-cooled storage facility during 30 years in order to reduce the heat flux from the waste in the final repository. The intermediate storage is located underground with 30 meters of rock coverage and directly above the final repository. It will have a capacity of storing 6000 containers of vitrified waste (each containing the equivalent of 1 ton of uranium) in vertical pits in a concrete

structure. The design is very similar to a storage facility which is in operation in Marcoule, France. Final disposal will take place in a repository located in granite 500 meters below the rock surface, where the canisters will be placed in vertical holes which are backfilled with a mixture of sand and bentonite.

For the transport of the waste within the intermediate storage facility and to final disposal a radiation-shielded cast, specially designed for this purpose, will be used.

The report describes the facilities for intermediate storage and encapsulation as well as the handling equipment to be used. Approximate costs for the intermediate storage facility are indicated. Finally a safety analysis is presented.

KBS Technical Report No 36

COMPOSITION OF GROUNDWATER DEEP DOWN IN GRANITIC BEDROCK

Jan Rennerfelt, Orrje & Co, Stockholm, 1977-11-07

Summary

The effect of different types of dissolved solids in ground water on corrosion and leaching is discussed. A suitable composition of water for leaching tests is indicated. The technique for sampling of water in bedrock at large depths is discussed.

Water analyses from different investigations are presented and a probable interval of water composition as well as some maximum values are given. Very low oxygen concentrations, relatively high Fe^{2+} concentrations and low levels of organic substance can be expected. Leaching of bentonite can increase the concentration of organic substance.

KBS Technical Report No 37

HANDLING OF BUFFER MATERIAL OF BENTONITE AND QUARTZ

Hans Fagerström, VBB,
Björn Lundahl, Stabilator
Stockholm, October 1977

Summary

Storage system

The final storage of the radioactive waste from nuclear power plants will take place in a tunnel system located 500 m below the ground surface in tectonically undisturbed rock.

The waste is placed in canisters which will be embedded in a buffer material in order to prevent damages and minimize the flow of groundwater entering in contact with the canisters. Tunnels and vertical shafts to the ground surface will also be filled with similar material in order to limit groundwater movements.

Object of report

The intention of the field and laboratory investigations described in this report has been to investigate in which way a buffer material consisting of quartz and bentonite can be handled by means of known technics. In the tests, different mixtures have been studied with respect to function and handling. The test results have formed the basis for specifications, which step by step describe the practical handling of the buffer material when deposing the canisters and closing the tunnels.

Deposition schemes

Alternative deposition schemes have been studied for reprocessed as well as nonreprocessed high level radioactive waste (HLW). In the alternative with reprocessed material, canisters with vitrified waste are stored in vertical deposition holes in the bottom of the tunnels, while in the alternative with nonreprocessed waste the canisters are placed horizontally in the center of the tunnels.

The filling of buffer material around the canisters and in the tunnels will for the reprocessed waste alternative be divided in two stages. In the first stage, which will go on for about 30 years, canisters and buffer fill are placed in the deposition holes. The second stage, which comprises the filling of tunnels and shafts, will then take place during a three year period. In this stage the amount of filling per day will be about 1200 m^3.

In the nonreprocessed waste alternative, deposition and filling of the tunnels will be of more continuous character.

Advantages of quartz/bentonite materials

Field and laboratory tests have shown that a wellgraded mix of quartz and bentonite is the most suitable buffer material. The main advantages and disadvantages of the two components in the mixes can be summarized as:

	quartz material	bentonite
long time stability	+	+
bearing capacity	+	-
plasticity	-	+
permeability	-	+
swelling properties	-	+
heat conductivity	+	-
ion exchange capacity	-	+

Suitable mixture

According to the testings performed, the proportions of the materials can be adjusted to give the buffer fill desired properties in different zones. With mixes as stated below, the materials will fullfill technical requirements as well as those of handling:

10-20 % granulated Na-bentonite (0.07-0.8 mm)
10 % filler (0.02-1 mm)
60-70 % quartz sand (0.063-2 mm)
10 % quartzite gravel (4-8 mm)

The grain size distribution curves for the buffer material as well as for the different components are shown in Fig. 10.

Properties of buffer fill

The properties of the buffer fill as determined in tests are summarized as follows:

- strength and deformation properties are similar to a clayey moraine (boulder clay). The strength of the material is built up of a cohesion part and a friction part. An increase of the bentonite content tends to increase the cohesion and decrease the friction,
- permeability of the mixes is in the order of 10^{-10} m/s when swelling is restricted. Increasing content of bentonite gives lower permeability. The permeability is decreasing with time as the bentonite swells when exposed to water,
- heat conductivity has been determined to values between 0.5 and 2 W/m^{O}K when the water content of the mixes varies between 5 and 30 per cent. Higher water and quartz content as well as temperature increases the heat conductivity,
- compaction properties of the mixes are similar to a clayey moraine. Maximum dry densities are in the range 1.90-2.00 t/m^3 when determined in mod. AASHO tests. Corresponding optimum water contents are in the range 8-12 per cent,
- as regard swelling properties, the free swelling of mixes compacted at water contents in the range

10-15 per cent has been measured to between 15 and 35 per cent. With restricted swelling, pressures in the order of 30-200 kPa (0.3-2 kp/cm²) have been measured.

It is an advantage that the swelling properties of the buffer fill can be adjusted to confirm with the requirements in different zones. A high bentonite content is favourable in the upper portion of the tunnels in order to guaranty that possible voids and pockets will be filled up through swelling, thus minimizing any horizontal water movements. The variations in bentonite content will give somewhat different swelling properties to adjacent zones, but this is not deemed to cause inconveniences, considering the good bearing capacity of the mixes.

Varying requirements

The quality requirements on the sand/gravel material differs somewhat in different parts of the tunneling system. Pure quartz-material is required in:

- deposition holes
- bottom part of deposition tunnels in the reprocessed HLW alternative
- bottom and upper part of the deposition tunnels in the nonreprocessed HLW alternative
- lower 100 m of the shafts.

Natural sand and gravel containing feltspar and mica can be accepted in:

- transport tunnels and caverns
- upper part of the shafts
- upper part of the deposition tunnels in the alternative with reprocessed HLW.

Handling of buffer fill

The buffer material can be mixed and handled utilizing technics well known from fabrication of concrete as well as from earth fill works and road works. The technic for filling and compaction the material in deposition holes in shown in Fig. 11.

In the lower part of the tunnels, conventional earth fill equipment can be used. In the upper part, the buffer material will be filled and compacted with a technic similar to that used in shotcreting (the ROBOT-method), Fig. 13-14.

Specifications for buffer fill

The specific requirements on the buffer fill in various parts of the deposition system are listed below:

Reprocessed HLW alternative

Deposition holes:		below canister	around canister	above canister
bentonite content	%	10	15	15
water content	%	10±2	15-20	15±2
degree of compaction	%	90	80	80
layer thickness	m	0.2	0.1-0.2	0.2

Tunnels:		bottom part	upper part	shafts
bentonite content	%	10	10-20 1)	0-10 2)
water content	%	10±2	10-15	8-10
degree of compaction	%	90	70-80	90
layer thickness	m	0.2	-	0.2-0.3

1) The higher value of the bentonite content might be used in order to secure that pockets and voids along the roof are filled up due to swelling of the material.

2) The higher bentonite content is used in the lower 100 m. Above this level, natural sand/gravel materials or moraine can be used. Bentonite might be added to secure that the permeability is not higher than 10^{-8} m/s.

Nonreprocessed HLW alternative

Lower part of deposition tunnels, with zones according to Fig. 15 (page 52):

		zone I	zone II	zone III
bentonite content	%	15	10-15[3)]	15
water content	%	10±2	8±2	8±2
degree of compaction	%	90	90	90
layer thickness	m	0.2	0.2	0.15-0.2

3) The lower bentonite content is used in the part below the canister in order to secure a high bearing capacity.

In the upper part of the deposition tunnels, a mix of pure quartz material and bentonite shall be used. In other respects, the requirements for tunnels and shafts listed above in the description of the reprocessed HLW alternative are valid also for the nonreprocessed one.

KBS Technical Report No 38

DESIGN OF ROCK CAVERN FACILITIES

Arne Finné, KBS, Alf Engelbrektson, VBB
Stockholm, December 1977

Summary

The present report contains a compilation of data concerning the layout and design of temporary and final rock storage facilities for vitrified high level waste from nuclear plants.

In the introduction the basic principles for the design of the storage facilities are briefly summarized. The vitrified waste, in the shape of glass cylinders encased within containers of chrome-nickel steel will be stored for at least 30 years in a Temporary Storage. After this period the glass containers will be encapsulated in a lead-titanium canister and transferred to the Final Storage, which is located in rock underneath the Temporary Storage at a level of about 500 m below the surface. The rock together with the material used for filling the tunnels and other excavations will constitute the ultimate barrier against the migration of radioactive substances to the biosphere and provide protection against external forces such as acts of war, etc.

The Final Storage has a capacity to receive 9 000 waste canisters, each containing a quantity of waste corresponding to approximately 1 ton of reactor uranium. The canisters are distributed over a horizontal area of 1 km^2.

As shown in Fig. 1, the Temporary Storage and Encapsulation Plant is located in two adjacent rock cavities, housing the Reception and Encapsulation Facilities in one of the rooms and the Temporary Storage in the other. The max. span of both rooms is approx. 20 m and the covering rock at least 30 m in thickness. The rooms are connected to each other and to the ground by tunnels, and there are also tunnel connections with the main shaft and the waste transport shaft down to the Final Storage.

The layout of the Temporary Storage and Encapsulation plant is shown in further detail in the drawings A1-A5, enclosed in Appendix 2 to this report. The building structures are made mainly of cast-in-situ concrete. The design and the technical standard of the structures as well as the installation are similar to those of auxilary buildings and equipment in nuclear plants.

The handling of the vitrified waste within the plant is illustrated in Fig. II. The glass containers arrive at the reception hall inside a transport cask. After unloading and, when necessary, recanning, the glass containers are transferred to the Temporary Storage by a transfer cask with radiation shields. The containers are stored in vertical steel tubes, 10 cylinders in each tube. The tubes are located in four storage chambers, isolated from the surrounding spaces by thick concrete structures. The cooling of the waste is provided by means of a ventilation system consisting of two 100 per cent capacity units, the fans of which are housed in the ventilation rooms adjacent to the storage chambers.

The encapsulation of the glass cylinders in the lead-titanium canisters must be performed by remote operation. The glass containers are transferred to the Encapsulation Cell by the transfer cask. The cell is enclosed in radiation-shielding concrete structures.

After the final encapsulation the waste canister is placed within a transfer cask on rails and lowered by an elevator to the Final Storage. As shown in Fig. III, this facility consists basically of a system of parallel storage tunnels with access tunnels and shafts to the ground level. Vertical holes are drilled in the floor of the tunnels, and in each hole a canister is placed and embedded into a compacted buffer material of sand and bentonite.

The construction of the shaft and tunnelling system will begin with the sinking of a shaft from the ground level down to the level of the storage tunnels. From this shaft horizontal adits will be driven to permit the drilling of the other shafts with raise boring methods. Adjacent to the storage area, tunnels for service facilities will be constructed. Rock blasted will be transported to the ground level by a skip hoist.

Construction of the storage area tunnelling system will begin with the peripheral and the centre transport tunnels and with the ventilation tunnel. A good survey of the rock conditions is thereby obtained and the layout of the storage tunnels may then be modified if necessary. Construction of the storage tunnels will then commence with the use of careful blasting or possibly full face boring in order to minimize disruption of the rock. In the floor of these tunnels the deposition holes will be drilled. Before the holes are drilled to full size, exploratory holes will be drilled and the permeability of the surrounding rock leakage tested.

The deposition of a waste canister in the final storage is illustrated in Fig. IV. From the elevator, when at the final storage level, the transfer cask is moved on rails, pulled by an electric tractor, through the tunnelling system of the storage area and positioned above the hole in the tunnel floor in which the waste canister is to be placed. The canister is lowered down into the hole with the hoist of the transfer cask and placed on a pre-compacted bed of sand and bentonite. The transfer cask is then removed and the hole filled with a mixture of sand and bentonite, which is spread and compacted in layers of 10 - 20 cm (See Fig. V).

The Final Storage will have auxiliary systems for ventilation water supply, sewerage, electric power and lighting, compressed air, fire protection, telecommunications, transport of personnel and materials etc. Basically the systems are very similar to what is generally provided in conventional mining installations.

The ventilation system is designed on the basis of the free flow of air in tunnels and shafts. The basic design criteria for the system are the volume of fresh air needed for the construction works and for the dissipation of the heat release from the canisters. The temperature in tunnels where personnel are working shall not exceed 25°C.

The placing of canisters will begin when about one-fourth of the storage tunnels has been completed and will proceed for 30 years. The design of the facilities provides for a complete physical separation of construction works from the transport and the placing of canisters. The equipment for the transport and handling of the canisters as well as for the back-filling of the holes is operated on rails and pulled by electrically powered tractors.

Up to the time when the Final Storage will be sealed, the storage tunnels in which waste canisters have been placed may be inspected and monitoring of rock stresses, temperature gradients, leakage of ground water into the tunnels etc performed.

When the Final Storage is to be sealed the tunnelling system is filled with a sand/bentonite mixture similar to the material used for the backfilling of the holes for the waste canisters in the storage tunnels. The lower part of the fill will be applied in layers and compacted with vibratory equipment. In the upper part of the tunnels the sand/bentonite mixture is applied with shotcrete equipment adapted for this purpose. The shotcrete technique and the swelling properties of the bentonite will ensure that the tunnels will be completely filled with a high degree of compaction. A sand/bentonite mixture will also be used for the filling of vertical shafts and holes. In the upper part of the shafts a finely grained moraine may also be used. Compaction in shafts will be made with vibratory equipment.

In this manner all cavities made in the rock will be filled with a material having a permeability which is at least as low as that of the surrounding rock.

Drawings A11 - A18 showing the Final Storage in further details are attached in Appendix 2 to this report. (See List of Drawings, App. 2.)

The calculated cost for the construction of the Temporary and Final Storage facilities including process equipment as well as depositioning and sealing operations amounts to 1 300 Millions Swedish Crowns referred to the current cost level. Costs for encapsulation work and materials, surveillance of the facilities and similar operations are not included in this sum.

KBS Technical Report No 39

DESIGN STUDIES, DIRECT DEPOSITION

Bengt Lönnerberg, ASEA-ATOM, Västerås, September 1978

Summary

The second part of the KBS project deals with final storage of spent nuclear fuel without reprocessing. ASEA-ATOM has contributed to this part with a study of the incapsulation process before the final s.orage in bedrock.

The complete handling process is described in the main KBS report. On the following pages a short process description is given, to a certain extent supplementing the KBS report and commenting some of the chosen process stages. Additional detailed studies regarding the various process steps are reported in the enclosed appendices.

The description includes the following process steps:

- spent fuel cask reception from the central storage
- unloading of casks
- dismantling of fuel bundles
- insertion of fuel rods in copper racks
- placing of racks in copper capsules
- filling of capsules with lead
- welding of capsule lids
- capsule transport to the underground final storage

The handling of remaining metal parts from the fuel bundles are described as well as collection of lossed particles from damaged fuel in the process.

KBS Technical Report No 40

ECOLOGICAL TRANSPORT AND RADIATION DOSES FROM GROUNDWATERBORNE RADIOACTIVE SUBSTANCES

Ronny Bergman, Ulla Bergström, Sverker Evans
AB Atomenergi, 1977-12-20

Summary

The second part of the KBS project deals with final storage of spent nuclear fuel without reprocessing. ASEA-ATOM has contributed to this part with a study of the incapsulation process before the final s.orage in bedrock.

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The handling of remaining metal parts from the fuel bundles are described as well as collection of lossed particles from damaged fuel in the process.

KBS Technical Report No 41

SAFETY AND RADIATION PROTECTION IN THE FIELD OF NUCLEAR POWER. LAWS, STANDARDS AND GROUNDS FOR EVALUATION

Christina Gyllander, Siegfried F Johnson, Stig Rolandson
AB Atomenergi and ASEA-ATOM, 1977-10-13

SUMMARY

This report gives a short presentation of the most important conventions, laws, standards and regulations which govern safeguards, criticality safety, transport and physical protection of spent fuel at the back-end of the fuel cycle. New international recommendations - ICRP No 26 - have been published regarding radiological protection. National standards for release limitation are issued by the National Radiation Protection Institute and these have been accepted by the Swedish Government. A joint statement has been published by the Nordic radiation protection institutes.

In this report a summary is given of recommendations and standards as well as basic principles for the system of dose limitations. New radiological units and concepts are defined. Codes and standards concerning the design of storage facilities for spent reactor fuel and highly radioactive wastes exist in the USA and West Germany.

American codes and standards and a regulatory guide related to the design of intermediate storage facilities of spent fuel and radioactive wastes are listed, and their application to Swedish conditions is discussed. In a similar way a description and a brief evaluation are given for a West German DIN-norm concerning the design of storage basins for spent fuel.

A brief survey is also given of the current work in the United States concerning codes and standards covering other activities of the later parts of the fuel cycle.

KBS TEchnical Report No 42

SAFETY IN THE HANDLING, STORAGE AND TRANSPORTATION OF SPENT NUCLEAR FUEL AND VITRIFIED HIGH-LWVEL WASTE

Ann Margret Ericsson, Kemakta, November 1977

Summary.

The safety of handling and transportation of spent fuel and vitrified high-level waste has been studied. Only the operations which are performed in Sweden are included. That is:

- Transportation of spent fuel from the reactors to an independant spent fuel storage installation (ISFSI).
- Temporary storage of spent fuel in the ISFSI.
- Transportation of the spent fuel from the ISFSI to a foreign reprocessing plant.
- Transportation of vitrified high-level waste to an interim storage facility.
- Interim storage of vitrified high-level waste.
- Handling of the vitrified high-level waste in a repository for ultimate disposal.

For each stage in the handling sequence above the following items are given:

- A brief technical description.
- A description of precautionary measures considered in the design.
- An analysis of the discharges of radioactive materials to the environment in normal operation.
- An analysis of the discharges of radioactive materials due to postulated accidents.

The dose to the public has been roughly and conservatively estimated for both normal and accident conditions. The expected rate of occurence are given for the accidents.

The results show that the above described handling sequence gives only a minor risk contribution to the public.

KBS Technical Report No 43

TRANSPORT OF RADIOACTIVE ELEMENTS IN GROUNDWATER FROM A ROCK REPOSITORY

Bertil Grundfelt, Kemakta, 1977-12-13

SUMMARY

The migration of radionuclides from a repository for vitrified high-level waste in Swedish bedrock has been studied.

The mathematical model used comprises migration with flowing groundwater, dispersion and geochemical retardation of the migrating nuclides. The model was originally developed at Batelle Pacific Laboratories, Richland Wa.

In this report a presentation of the influence of the parameters time of leach incident, leach duration and groundwater velocity on the discharge rate to a recipient is given as well as a discussion of degree of pessimism in the different assumptions made.

The results show that the discharge rates for such radiologically important nuclides as Sr-90, Cs-137 and Am-241 are less than 10^{-15} Curies per year even if the residence time for groundwater in the rock is as short as 40 years and with pessimistic assumptions concerning the time of leach incident and the leach duration.

If the recidence time for groundwater increases above 100 years the plutonium isotopes decays significantly.

KBS Technical Report No 43

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Bertil Grundfelt, Kemakta, 1977-12-13

SUMMARY

The migration of radionuclides from a repository for vitrified high-level waste in Swedish bedrock has been studied.

The mathematical model used comprises migration with flowing groundwater, dispersion and geochemical retardation of the migrating nuclides. The model was originally developed at Batelle Pacific Laboratories, Richland Wa.

In this report a presentation of the influence of the parameters time of leach incident, leach duration and groundwater velocity on the discharge rate to a recipient is given as well as a discussion of degree of pessimism in the different assumptions made.

The results show that the discharge rates for such radiologically important nuclides as Sr-90, Cs-137 and Am-241 are less than 10^{-15} Curies per year even if the residence time for groundwater in the rock is as short as 40 years and with pessimistic assumptions concerning the time of leach incident and the leach duration.

If the recidence time for groundwater increases above 100 years the plutonium isotopes decays significantly.

KBS Technical Report No 44

DURABILITY OF BOROSILICATE GLASS

Tibor Lakatos
Glasteknisk Utveckling AB, Växjö, December 1977

Summary

Borosilicate glass has a good resistance to radiation of different kinds, good mechanical properties towards compressive stress. Devitrification can occur between 600 and 800° C.

This report gives a summary of experiences from French research with high-active borosilicate glass, also English, German and American research as well as research on inactive borosilicate glasses at the Swedish Research Institute in Växjö.

KBS Technical Report No 45

CALCULATION OF TEMPERATURES IN A SINGLE-LEVEL FINAL REPOSITORY IN ROCK FOR VITRIFIED RADIOACTIVE WASTE REPORT 3

Roland Blomquist, AB Atomenergi, 1977-10-19

SUMMARY

Introductory calculations of temperatures in rock storages for final disposal of radioactive waste have earlier been performed, see KBS Technical Reports 05 och 15. In those calculations the values of important parameters such as geometrical arrangement, age of waste, thermal conductivities, etc have been varied in order to show their influence on temperatures.

Gradually the main interest have been directed to a storage in one level covering a horisontal area of 1 x 1 km. A mixture of quartz sand and bentonite has been chosen as filling material around the waste containers.

This paper reports on temperature calculations for a storage of this model.

KBS Technical Report No 46

TEMPERATURE CALCULATIONS FOR SPENT FUEL

Taivo Tarandi, VBB, June 1978

Summary

Temperature distribution in and around the final storage has been calculated for BWR-fuel.

The results are applicable also for PWR-fuel if the amount of fuel is reduced so that the effect per canister is the same.

The calculations are made with the conservative assumption of the coefficient of thermal conduction of 0,75 W/m ^{o}C in the bentonite.

The amount of BWR fuel is 1.4 ton per canister. The canister is deposed 40 years after withdrawal from the reactor.

The maximum temperature of 76^{o} C at the surface of the canister is reached 10 - 20 years after the time of deposition. The highest temperature in the rock, 60^{o} C, occurs about 60 years after the deposition.

At the same time as the temperature continues to sink, there is a levelling out of the local temperatue differences in the storage. These differences are negligible after about 1 000 years.

After 100 000 years the temperature in the storage is only a few centigrades above the initial rock temperature.

The heat from the storage reaches the ground surface about 200 years after the deposition. The maximum heat flow, 0.15 W/m², occurs about 2 000 years after deposition and is considered insignificant compared for example with solar energy flow of about 100 W/m².

KBS Technical Report No 47

INVESTIGATIONS OF GROUNDWATER FLOW IN ROCK AROUND REPOSITORIES FOR NUCLEAR WASTE

John Stokes, Roger Thunvik
Department of agricultural hydrotechnics,
Royal Institute of Technology, 1978-02-28

Summary.

I. Groundwater Flow due to Topographical and Geological Effects.

A first investigation of the principal aspects of groundwater flow when varying topography, hydraulic conductivity and geometri is presented. The groundwater table was assumed coincident with topography. The conductivity was assumed constant or exponentially decreasing with depth. The bottom was either fixed or infinite. Numerical examples containing equipotentials, streamlines, lines of equal flux and travel times are presented. Several profiles in the Forsmark area were analysed to show the effects of regional flow. This model is based on the analytical solution of the equations

$$\varphi_{xx} + \varphi_{zz} + \frac{K'(z)}{K(z)}\varphi_z = 0$$

and

$$\varphi_{rr} + \varphi_{zz} + \frac{1}{r}\varphi_r + \frac{K'(z)}{K(z)}\varphi_z = 0$$

where $K(z) = c \cdot e^{2\mu z}$ is the hydraulic conductivity. Here c and μ are empirical constants.

II. Local Groundwater Depression around a Repository.

A two-dimensional flow analysis was made to study the effect on the groundwater table due to drainage of the storage tunnels during the construction resp. operation period. The geometry was

chosen the same as for the principal study in section I. The net accretion to the phreatic surface was assumed evenly distributed in space and time. Numerical examples with equipotentials and consecutive positions of the phreatic surface are presented. The model is based on the numerical solution of the following equation of flow:

$$\nabla^2\varphi(x,z,t) = 0$$

The phreatic surface is defined by the following boundary contion:

$$\frac{\phi}{K}\frac{\partial \eta(x,t)}{\partial t} - \frac{\partial \varphi}{\partial x}\frac{\partial \eta}{\partial x} + \frac{\partial \varphi}{\partial z} + \frac{\varepsilon}{K} = 0$$

where ϕ is porosity, η is the elevation of the phreatic surface and ε is the net accretion.

III. Three-Dimensional Model for Groundwater Flow due to Topographical and Geological Effects.

As a complement to the two-dimensional computations for the Forsmark area performed in section I, a three-dimensional flow model was developed. In this model, which is a finite-element model, the conductivity is defined separately for each element. In this way fracture zones have been included as well as a conductivity varying in space. In the equation of flow

$$\frac{\partial}{\partial x}\left(K_x \frac{\partial \varphi}{\partial x}\right) + \frac{\partial}{\partial y}\left(K_y \frac{\partial \varphi}{\partial y}\right) + \frac{\partial}{\partial z}\left(K_z \frac{\partial \varphi}{\partial z}\right) = 0$$

the K-value is subsequently a function of space.
Numerical examples for a K-value eponentially decreasing with depth, and with a given geometry of the fracturezones, are studied. The results are presented as equipotentials, travel times and maps describing conditions of in- and out flow.

KBS Technical Report No 48

THE MECHANICAL PROPERTIES OF THE ROCKS IN STRIPA, KRÅKEMÅLA, FINNSJÖN AND BLEKINGE

Graham Swan
Luleå Institute of Technology, 1977-08-29

SUMMARY

The mechanical properties of Stripa Granite are presented as determined from small (laboratory size), oven-dried specimens. The properties determined include Young´s modulus, Poisson´s ratio, unaxial compressive fracture stress and the expansion coefficient, all as a function of temperature.

In addition the Brazilian tensile fracture stress, residual shear strength as a function of a normal stress and the rock´s anisotropy ratios are presented. Finally ultrasonic determinations at 1 MHz of the rock´s dilatational wave velocity are given and the deduced Young´s modulus is compared with the static value for room temperature.

KBS Technical Report No 49

MEASUREMENTS OF ROCK STRESSES IN THE STRIPA MINE

Hans Carlsson
Luleå Institute of Technology, 1977-08-29

SUMMARY

Rock stress measurements at 340 m levels of the Stripa Mine have been carried out with the Leeman tri-axial equipment. The largest principel stress is found to be 20.0 MPa and directed parallel with the strike of the granite. The intermediate principal stress is 11.5 MPa and directed almost horizontal and perpendicular to the contact. The minor principel stress has a magnitude of 5.4 MPa. The deduced vertical stress is approximately of the same value as can be theoretically calculated.

KBS Technical Report No 50

LEACHING TRIALS WITH HIGH-LEVEL FRENCH GLASS AT STUDSVIK

Göran Blomqvist, AB Atomenergi, November 1977

Glasses with incorporated highly active radioactive waste is now leached at Studsvik with solutions approximating subsurface water from crystalline rocks. Results at ambient temperature give leach rates somewhat higher than the values obtained by the French, but the values are still incomplete. For Pu, the values obtained are about 4×10^{-7} g cm^{-2} day^{-1} which is somewhat higher than the French values of approximately 2×10^{-7} gcm^{-2} day^{-1}. The temperature factor seems to be approximately 10 for Sr and Cs and very small for Pu.

The active glasses contain approximately 20 % waste oxides compared to 9 % contracted for waste from Swedish reactors.

KBS Technical Report No 51

SEISMOTECTONIC RISK MODELLING FOR NUCLEAR WASTE DISPOSAL IN THE SWEDISH BEDROCK

F Ringdal, H Gjöystdal, E S Husebye
Royal Norwegian Council for scientific and industrial research, October 1977

Abstract

The problem studied in this report concerns geotectonic risk factors for nuclear waste canisters stored underground in the Swedish bedrock, and embedded in tunnels filled with a clay-like material. The tectonic events causing potential damage can be classed as (i) earthquakes and (ii) creep motion. The numerous existing faults observed in Sweden are in general the results of past orogenic cycles, dating several hundred million years back in time. Only a few faults have shown surface movements after the most recent glaciation (ending about 8000 years b.p.), and in most cases such neotectonic movements have occurred along already existing zones of weakness.

The most relevant tectonic forces presently acting on the Fennoscandian region are due to (i) glacial rebound, (ii) remnant stresses from past orogenic cycles and (iii) mid-oceanic ridge push forces. Existing data are too scarce to allow a reliable separation of these components. The uplift of Fennoscandia is very smooth, and this implies that the probability of differential movements (or faulting) in this connection is low. A statistical analysis of the lengths and orientations of major faults in Sweden has shown that the distribution of fault lengths is approximately lognormal with a mean of 10-15 km and a standard deviation corresponding to a factor of two. Moreover, the fault density and distribution is quite similar over all of the Swedish area. The present seismicity of Sweden and surrounding areas is low, but occasional earthquakes of magnitude around 6 have been observed in historic times. Estimated depths of observed earthquakes are mostly 10 km or more, thus the expected seismic effect at

the planned depths of canister storage will be small. Surface faulting of Swedish earthquakes is at present seldom, if ever, observed. Normal faulting is generally to be expected for Swedish earthquakes, although no fault plane solutions have yet been found in this area. A model has been developed to compute the probability of a storage area being intersected by a known or unknown fault. The model is based upon lognormal distribution of fault lengths and a random orientation of the faults with all directions having equal probability of occurrence. The model is used in conjunction with the assumption that the rate of creation of new faults has remained constant over the past 1500 million years, and will remain so in the future. The probability of a new fault created during the next 10,000 years intersecting a given circular storage area of 1 km radius is then of the order $5 \cdot 10^{-6}$.

KBS Technical Report No 52

CALCULATIONS OF NUCLIDE MIGRATION IN ROCK AND POROUS MEDIA PENETRATED BY WATER

H Häggblom, AB Atomenergi, 1977-09-14

SUMMARY

Some physical and matehmatical models are given for migration of nuclides in rock and porous media penetrated by water. The cases considered are thermal convection due to the decay heat from radioactive sources and transport due to the hydraulic gradient connected with the geographic structure. The model for thermal convection is highly simplified but is conservative compared to the often made adiabatic assumption which limits convection effects to a region characterized by a high ratio between buoyant and viscous forces. The piezometric head and corresponding gradients are calculated by analytic methods. It is shown that the solutions are strongly dependent upon the variation of the permeability with depth. The special features of migration in cracks are studied. A computer program, MINUTE, was developed for numerical calculations. Times for transport of some important nuclides to the ground surface were calculated using appropriate input data for representative Swedish deposition sites. Error margins are discussed.

KBS Technical Report No 53

MEASUREMENT OF RATE OF DIFFUSION OF SILVER IN CLAY-SANDMIX

Bert Allard, Heini Kipatsi
Chalmers University of Technology, 1977-10-15

Summary

Copper has been proposed as capsule material for high active waste. The cannisters will be placed in a clay/quartz-mixture. One of the factors determining the dissolution of the cannisters will be the diffusion rate of copper, probably as Cu(I)-ions, through the claybarrier.

Because of the difficulty to keep Cu(I)-ions in aqueous solution, we have measured the diffusion of Ag(I)-ions (as the radioactive isotope ^{110m}Ag), which will have similar diffusion properties as Cu(I).

An apparatus to study Ag-diffusion in clay with radioactive tracer was constructed.

The distribution coefficient for Ag in a clay/quartz-mixture was measured to be 65×10^{-3} m^3/kg and the diffusion rate to be between 6×10^{-13} to 8×10^{-13} m^2/s.

KBS Technical Report No 54:01

GROUNDWATER MOVEMENTS AROUND A REPOSITORY

54:01 GEOLOGICAL AND GEOTECHNICAL CONDITIONS

Håkan Stille, Anthony Burgess, Ulf E Lindblom
Hagconsult AB, September 1977

SUMMARY OF RELEVANT PROPERTIES AND CONDITIONS

During each of the four time spans of a repository, namely those of pre-mining, mining (but pre-emplacement of radioactive waste), short-term conditions, and long-term conditions, the factors affecting the groundwater flow are different and must be treated separately. The basic philosophy of modelling in this project at this time is to study discrete but related phenomena using existing computer programs, with only relatively simple modifications to the programs and model development as required. There is not sufficient time for complex program development, and the sparsity of data precludes use of one comprehensive model containing all desirable features. The model used in this project is based on uncoupled programs for heat transfer, rock mechanics, and groundwater flow. Full coupling is developed between the programs for heat transfer and groundwater flow, with partial coupling between heat transfer, rock mechanics and groundwater flow. The groundwater flow is coupled to the heat transfer through thermal convection and advection, and to the rock mechanics through the permeability of the joints, which depends on the deformation of the joints.

The geotechnical parameters considered to be of importance in assessing the thermomechanical and hydrogeological behaviour are listed below. All the parameters can depend on several factors, including stress, time, temperature and the geological conditions of a specific site.

(a) Termal properties and conditions for the rock mass

- Thermal conductivity
- Specific heat capacity
- Density
- Coefficient of thermal expansion
- Convective film coefficient for heat transfer between the rock matrix and pore water

(b) Mechanical properties and conditions for the rock mass

- Young's modulus of elasticity
- Poisson's ratio
- Density
- Quasi/static compressive and tensile strengths of the intact rock mass including the failure envelope

(c) Mechanical properties and conditions for joints in the rock mass

- Cohesion and friction angle (or, the failure envelope)
- Shear stiffness
- Normal stiffness
- Creep behaviour

(d) Hydrogeological properties and conditions of the rock mass

- Joint widths and spacings
- Density and viscosity of pore water
- Permeability of the joint
- Permeability of the fractured rock mass

The properties of the jointed rock mass when treated as a continuous material can be evaluated from the properties and conditions of the intact rock and the joints. The pertubation of the groundwater flow will depend on the heat-generating characteristics of the radioactive waste, the repository geometry, and emplacement schedules, and on the thermomechanical behavior of the rock mass due to excavation and thermal loading.

The initial conditions are necessary for the calculation of the groundwater flow coupled to the thermal loading and rock mechanics. The following conditions have to be established.

(e) Initial conditions

- Initial stress field
- Initial flow field
- Initial temperature field
- Topography
- Initial permeability and porosity distributions

(f) Gross geological and environmental changes

- Changes in the stress field
- Changes in the flow field
- Changes in the temperature field

Many of the listed parameters are strongly depending on the geological and hydrogeological conditions at a specific site and of course they will also depend on the history of the rock mass.

In Table 1, the properties have been listed in conjunction with such geological considerations as

- Rock type
- Joints
- Geological and hydrogeological history
- Geological future

TABLE 1 Geotechnical parameters for various geological considerations

Geological considerations \ Geotechnical parameters	Thermal properties of the rock	Mechanical properties of the rock	Mechanical properties of the joint	Hydrogeological properties	Initial conditions
Rock type	Specific heat Thermal conductivity Thermal behaviour of rock matrix	Young´s modulus Poisson´s ratio Density Failure envelope Creep behaviour			Density
Joints:					
Filling material	Film coefficient		Failure envelope Stiffness	Hydraulic conductivity	
Width			Failure envelope Stiffness	Hydraulic conductivity	
Roughness and undulation	Thermal conductivity from rock block to rock block		Failure envelope Normal stiffness Shear stiffness Creep behaviour	Hydraulic conductivity	
Spacing				Hydraulic conductivity	
Geological and hydrogelogical history		Anistropy		Water conditions	Initial temperature field Initial stress field Initial flow field
Erosion Glaciations Weathering Tectonics		Creep behaviour	Creep behaviour		Changes in: stress field flow field temperature field

KBS Technical Report No 54:02

GROUNDWATER MOVEMENTS AROUND A REPOSITORY

54:02 THERMAL ANALYSES
PART 1 CONDUCTION HEAT TRANSFER
PART 2 ADVECTIVE HEAT TRANSFER

Joe L Ragitan
Hagconsult AB, September 1977

SUMMARY AND CONCLUSION

Conduction "baseline" thermal calculations have been completed for local near-field and global far-field temperature distributions. The effects of waste age, rock mass thermal conductivity, repository depth, repository ventilation, emplacement sequence and modeling geometry have been analyzed. These baseline results will be used in further phases of this study to analyze the effects on the rock mechanical situation and subsequent flow permeability perturbations. Additionally, the temperature fields will be utilized to assess thermally induced flow and to quantify the importance of free and forced convective heat transfer and their subsequent effects on the groundwater regime.

The temperature rises that have been observed are low in comparison to those arising in the analysis of repositories in other nations. This is mainly due to the waste age and the low GTL being investigated in Sweden. In a qualitative sense, the potential for thermally induced flow appears apparent. However, further study should quantify the magnitude of this flow.

The heat transfer due to forced convection resulting from a regional groundwater flow has been analyzed for the maximum expected flux, q of $2.(10^{-11})$ m/s and also, 10 q and 100 q. The temperature difference between assuming pure conductive heat transfer and coupled conductive and

convective heat transfer was found to be negligible for a groundwater velocity of q. Analysis of groundwater flux 10 q and 100 q results in temperatures nearly identical to those predicted by conductive heat transfer for the first 100 years. The major effect provided by the convective heat transfer with groundwater flow of 10 q and 100 q was to reduce the time required for the repository domain to return to the natural geothermal gradient. The influence of the convective heat transfer could be expected to be greater for larger values of thermal flux density or for rock masses with lower thermal conductivity than that assumed in this study.

KBS Technical Report No 54:03

GROUNDWATER MOVEMENTS AROUND A REPOSITORY

54:03 REGIONAL GROUNDWATER FLOW ANALYSES
PART 1 INITIAL CONDITIONS
PART 2 LONG TERM RESIDUAL CONDITIONS

Anthony Burgess, Hagconsult AB, October 1977

SUMMARY AND CONCLUSIONS

Based on regional models of groundwater flow, the regional hydraulic gradient at depth in equal to the regional topographic gradient. As a result, the equipotentials are near vertical.

The permeability distribution with depth influences the groundwater flow patterns. A zone of sluggish flows, the quiescent zone is developed when the permeabilit decreases with depth.This feature is accentuated when horizontal anisotropy, with the horizontal permeability higher then the vertical permeability, is included. The presence of an inactive zone till be a prerequesite for a satisfactory repository site.

The effect of an inclined discontinuity representing a singular geological feature such as a fault plane or shear zone has been modelled. The quiescent zone does not appear to be unduly disturbed by such a feature. However, meaningful quantitative predictions related to the flows in a typical singular feature cannot be made without more specific data on their hydraulic properties (permeability, anisotropy, fracture spacing).

The effect of relief of the ground surface and hence the water table was studied using a simple model. The influence of anisotropy and boundary conditions was

evaluated. With horizontal anisotropy and horizontal permeability decreasing with depth, the potential gradients due to the topographic relief decrease rapidly with depth. Perturbations to the quiescent zone under these conditions would be markedly reduced, compared to the effects for isotropic permeability.

Two dimensional analysis has been made for a site specific section of a candidate reository site at Forsmark. The lateral extent of the model was defined by major tectonic features, assumed vertical.

Potential gradients and pore velocities have been computed for a range of boundary conditions and assumed material properties. The potential gradients for the models with anisotropic permeability approach the average potential gradient between the boundaries.

The result of this study of the initial groundwater conditions will be used as input data for the analyses of the thermomechanical perturbations of the groundwater regime. In the long term, the groundwater flow will return to the initial conditions. The residual effects of the repository on the flow will be discussed in part 2 of this report.

The results of these analyses show that, for isotropic permeability distributions, the groundwater flow pattern is determined by the upper boundary potentials, i.e. topography. As a result, these flow patterns are site specific. For the anisotropic conditions, the topography affects only the upper 200 m: below that the flow is controlled by the regional gradient. The anisotropic flow patterns are therefore not as site specific, although the actual magnitudes of the fluxes and pore vlocities will depend upon the regional gradient in the area.

Compared to the impervious backfill case, pervious backfill results in a larger variation in fluxes and pore velocities in the repository area for all permeability distributions. The flow pattern for the isotropic case is similar for both backfill conditions. For the anisotropic case, however, the cross site flow becomes funnelled into the repository with a plume-like discharge downstream.

Pathways from the repository to the bounding discontinuity have been investigated and travel times determined. For the anisotropic conditions case 3 they range from 12 to 143 years with impervious backfill and from 6 to 1273 years for pervious backfill. If the permeability values at the repository level are adjusted to the value measured at Stripa, the travel times are 290 to 3421 years and 146 to 4610 years for the impervious and pervious backfill cases respectively.

The total flow into the bounding discontinuity from flow paths through the repository has been determined. However, the dilution of this flow in the discontinuity, and the pathways and travel times in the discontinuity cannot be estimated without further field data on the nature of this features at depth.

KBS Technical Report No 54:04

GROUNDWATER MOVEMENTS AROUND A REPOSITORY

54:04 ROCK MECHANICS ANALYSES

Joe Ratigan, Hagconsult AB, September 1977

SUMMARY AND CONCLUSIONS

Non-linear rock mechanics calculations have been completed for the repository storage tunnels and the global repository domain. The rock mass has been assumed to possess orthoganol joint sets or planes of weakness with finite strength characteristics.

In the local analyses of the repository storage tunnels the effects of joint orientation and repository ventilation have been examined. The local analyses indicated that storage room support requirements and regions of strength failure are highly dependent upon joint orientation. The addition of storage tunnel ventilation was noted to reduce regions of strength failure, particularly during the 30 year operational phase of the repository. Examination of the local stresses around the storage tunnels indicated the potential for perturbed hydraulic permeabilities. The permeabilities can be expected to be altered to a greater degree by the stresses resulting from excavation than from stresses which are thermally induced.

The global rock mechanics did not incorporate the excavation of the storage tunnels due to the large areal extent of the models. Subsequently, due to the relatively small thermal stresses which develop with the low gross thermal loading being considered in Sweden, the global models exhibited a reversible elastic response to the radiogenic heat disipation. The thermal loading provided by the instantaneous waste emplacement resulted in stress states and displacements quite similar to those provided by the linear waste emplacement sequence.

The stress states observed in the local and global rock mechanics models will be used in subsequent studies to quantitatively analyse the perturbations of the hydraulic permeabilities.

KBS Technical Report No 54:05

GROUNDWATER MOVEMENTS AROUND A REPOSITORY

54:05 REPOSITORY DOMAIN GROUNDWATER FLOW ANALYSES
PART 1 PERMEABILITY PERTURBATIONS
PART 2 INFLOW TO REPOSITORY
PART 3 THERMALLY INDUCED FLOW

Joe L Ratigan, Anthony Burgess, Edward L Skiba,
Robin Charlwood, Hagconsult AB, September 1977

SUMMARY AND CONCLUSIONS

The perturbations of the in situ hydraulic permeability caused by (a) stresses resulting from repository storage tunnel excavation and (b) the thermomechanical stresses resulting from the radiogenic heat have been evaluated. Changes in the permeability due to the temperature dependence of the viscosity have also been studied.

The effects of repository ventilation and waste emplacement sequence on permeability changes have been evaluated for the local and global models, respectively.

It was found that the most significant perturbation in the hydraulic permeabilities was caused by the excavation of the storage tunnels. The change in permeability caused by the temperature and thermomechanical stresses is expected to be small in comparison with expected uncertainty in in situ permeability values. The permeabilities are about 5-20% lower with storage tunnel ventilation than without ventilation.

The perturbed permeability fields will be used in subsequent analyses of groundwater flow.

Field data on inflow to excavations at depths of hundreds of meters is primarily from measurements of inflow into mines. A study of inflow to mines in the Canadian Precambrian Shield is presented by Raven & Gale (5). The inflow rates were normalised by expressing the inflow as liters per minute per kilometer of lateral development in the mine (l/min,km). They concluded that mines with pumped levels within 150 m of the ground surface showed higher inflow rates than mines where the highest pumped level was deeper than 150 m. Values for the latter condition, pertiment to the present study, have been extracted from Raven & Gale and are given in Table 1.Apart from the value of 61.5 for East Malarctic No 5 shaft, the inflows are between 0 and 3 l/min,km.

As part of the field studies for KBS, measurements of inflow rates will be performed at Stripa Mine. A rough estimate made by the author during a site visit indicates an approximate inflow of 3×10^{-2} l/min,km. Permeability test indicate that they are of the order of 5×10^{-11} m/s.

A comparison of these very linutes field data with the inflow rates from the model simulations indicates that the model values are relatively high by about an order of magnitude. The final conclusions must await the results of further field data on the permeability distributions.

Thermally induced flows have been predicted to occur using a fully coupled thermal/flow finite-element program. It is assumed that flow in the host rock in the region of the repository can be modelled by Darcy´s law with a bouyant term under fully saturated conditions.

Thermally induced flow does not cause significant advective heat transfer for expected values of rock mass permeabilities. In the future thermally induced flow analyses may be done with bouyant coupling only using temperature fields computed on the basis of conduction only if the gross thermal loading is not significantly increased. This will enchance analytical stability and output accuracy.

If regional flow is not present, natural convection cells will develop. There will be global cells which could conceivably transport contaminants to the surface. Figure 29 displays the calculated average vertical transport veloci-

ties above the repository assuming no cross-flow. The predicted travel times are in the range of 1,000 years depending on the rock mass permeability. Local cells may also develop which will have the effect of channelling the flow close to the waste cannisters.
The theoretical model indicates that in the absence of cross flow natural convection cells can be initiated with small temperature gradients. The concept of a critical gradient for flow initiation does not apply in this problem. If partially saturated conditions exist, these may inhibit thermal flows. The fully saturated case analysed here is thought to give higher velocities and hence conservative results for safety analyses.

The inflow period may last for many years, depending on the nature of flow around and into the repository. While inflow lasts, it is expected that the inflow velocities will be 2 to 3 orders of magnitude larger than the velocities caused by natural convection alone. Consequently, inflow would prevent thermally induced flows from moving upwards to the surface during the recharge period. Natural convection flows can, however, start once inflow is essentially complete.

It is expected that a regional groundwater gradient of $2 \cdot 10^{-3}$ will exist which will cause the convection flows to be swept almost horizontally, indicating that the most likely point of exit from the host rock is into a singular feature at depth and not up to the surface above the repository.

The quantity of groundwater flowing through the repository due to regional gradients is increased by the thermally induced flow effects. These flow quantities may affect leach rate predictions depending on whether it is determined by concentration in the water or by the diffusion rate in the glass matrix.The local thermally induced flows may tend to channel flow near the canisters, although this tendency may be somewhat reduced by the regional flow.

KBS Technical Report No 54:06

GROUNDWATER MOVEMENTS AROUND A REPOSITORY

54:06 FINAL REPORT

Ulf Lindblom et al, Hagconsult, October 1977

CONCLUSIONS

The overall goal of this study has been to assess the groundwater flow field in the vicinity of a conceptual high-level radioactive waste repository, situated at a depth of 500 m in the precambrian bedrock of Sweden. Finite element modelling procedures have been used employing nominal and extrapolated data for initial groundwater conditions and precedent data for material properties. The coupling of thermal, rock mechanics, and groundwater flow effects has been achieved by means of quasi-static techniques. The results of these interelated processes have been analyzed for the following identifiable periods of the repository time frame: (1) pre-construction; (2) construction, but pre-emplacement of the waste; (3) post-emplacement of the waste through the significant portion of the thermal cycle; and (4) long term.

Assessment of the results of the analysis efforts lead to the following general conclusions:

(1) For the conceptual repository design at 500 m depth with a gross thermal loading of 5.25 W/m^2, the groundwater regime will not be significantly altered by the radiogenic heat dissipation.

(2) The long term flow fields will be determined principally by the flow regime prior to construction and can therefore be reliably predicted through establishment of the existing geohydrological parameters.

From the viewpoint of site selection, the practical design of the repository, and the waste emplacement concept, the following two results are of particular interest:

(1) The degree of anisotropy of the rock permeability is of importance in selecting the repository depth and location. Horizontal anisotropy results in groundwater flow becoming increasingly quiescient with depth.

(2) For a gross thermal loading of 5.25 W/m^2, the maximum rock temperature rise in the immediate vicinity of the waste canisters will be approximately 40^oC if room ventilation is maintained for 30 years after emplacement.

On the basis of the initial groundwater conditions, assumed materials properties and the conceptual repository design specifications, the following conclusions were deduced regarding the relative importance of the principle phenomena:

(1) Heat transfer in the host rock can be satisfactorily modeled considering conduction only since the influence of advective heat transport due to groundwater flow is negligible.

(2) Thermally induced flows will cause only minor perturbations to the regional groundwater regime in the vicinity of the repository.

(3) Inflow will dominate regional and hydrothermal flows during the recharge period.

(4) The potential for the development of pathways from the repository as a consequence of rock failure due to excavation and thermally - induced stresses is negligible.

(5) The change in rock permeability due to excavation and thermally - induced stresses, and to temperature - dependent groundwater viscosity, is localized.

The travel times computed in this study were as a rule based om extrapolatious to depth of data from presently available boreholes. Preliminary results from Field test in the Stripa mine indicate permeabilities 1000 times lower and therefore these travel times may be high by a factor of 100. The reliability of the predictions of groundwater flow will be greatly improved as field data, particularly rock mass permeability and porosity measurements, become available.

KBS Technical Report No 55

SORPTION OF LONG-LIVED RADIONUCLIDES IN CLAY AND ROCK PART 1

Bert Allard, Heino Kipatsi, Jan Rydberg
Chalmers University of Technology, 1977-10-10

SUMMARY

The mechanism of sorption of water soluble species in the natural environment has been discussed.

The radiochemical and radiobiological properties of the elements in spent nuclear fuel have been briefly discussed, and 14 of the radioactive products have been selected for studies of the sorption behaviour in contact with natural rock and clay minerals. These 14 elements are Sr, Zr, Tc, I, Cs, Ce, Nd, Eu, Ra, Th, U, Np, Pu and Am.

From data available concerning composition and equilibria in natural subsoil waters two standard water compositions have been suggested for the laboratory measurements.

Suitable concentrations of the radionuclides and experimental temperatures have been proposed.

A batch technique has been used for determination of distribution coefficients for powdered solid materials.

Measurements have been performed for all of the 14 elements with granite and bentonite/quartz mixture (10:90) with variation of water composition, nuclide concentration and temperature. Moreover, the effect of variation of the particle size has been studied for granite with Sr, Cs and Am, as well as the sorption in powdered granodiorite, chlorite and silt and on fresh and old rock surfaces (granite).

The presence of organic components in bentonite has been confirmed and a preliminary complex formation study with these organics has been performed.

Using the data from the measurements on rock surfaces a recalculation of measured mass dependent to surface dependent distribution coefficients has been ackomplished.

The stoichiometric composition of the water system has been theoretically analysed especially concerning the redox properties of the water. An estimation of the valencestate of the actinides U, Np and Pu has been suggested using reasonable assumptions regarding the hydrolysis of the actinides and the presence of Fe(III)/Fe(II) in natural waters.

Using measured or calculated distribution coefficients and considering the chemical properties of the natural water system, relevant retention factors of the radionuclides have been proposed.

KBS Technical Report No 56

RADIOLYSIS OF FILLER MATERIAL

Bert Allard, Heino Kipatsi, Jan Rydberg
Chalmers University of Technology, 1977-10-15

SUMMARY

The effect of gamma irradiation on the properties of the possible buffer material bentonite (bentonite:quartz, 10:90) has been discussed, mainly in qualitative terms.

Changes in ion exchange properties and lattice structure of the clay mixture due to irradiation in an aqueous solution have been studied.

The production of gas (hydrogen) due to radiolysis has been determined as a function of the radiation dose.

The resistance to corrosion of titanium in a radiation field has been demonstrated.

KBS Technical Report No 57

RADIATION DOSES IN THE EVENT OF A FAILURE DURING THE TRANSPORT OF NUCLEAR FUEL BY SEA

Anders Appelgren, Ulla Bergström, Lennart Devell
AB Atomenergi, 1978-01-09

SUMMARY

Transport of spent fuel is performed with casks that can resist very severe accidents without leakage. Tests are performed, for specified fall heights, fire and submersion in water in accordance with the regulations of IAEA. Model- and full-scale experiments with powerful collisions in the USA have shown that the casks can resist conditions above this regulations.

However, in order to investigate the consequences of shipping accidents, a release of acitivity is assumed. This report presents the calculations of individual and collective doses from the two most severe postulated accidents which are given in a special accident analysis. One of the accidents is a ship collision together with fire on-board, the ship is floating after the collision and a certain quantity volatile fission products gives airborne activity. In the other case, it is a fire on-board, the ship will sink and cause a certain leakage to the sea.

The release of activity to the air gives, with given conditions, an acute individual dose of 0.3 rem and an acute collective dose between 0 and 210 manrem, depending on where the accident occurs.

In a parameter study there was found that

- cesium-isotopes dominate with nearly 100 % in the standard case. Even if the fraction of activity release for cesium is as low as for the more involatile substances, their contribution is 60 %

- if the release height increases from 2 to 20 m, the individual dose decreases with a factor of 10, while the collective dose is nearly unchanged

- with a wheather type, which is more normal, the individual dose can be about 30 times less and the collective dose about 8 times less than the standard case, calculated for the Pasquill F type of wheather.

Also in an accident where the activity is released to the sea, Cs134 and Cs137 are dominating. The maximum individual annual dose through fish consumption is, with given conditions, 0.5 rem and the collective dose commitment over 30 years is 5 800 manrem.

KBS Technical Report No 58

RADIATION ON HAZARDS AND MAXIMUM PERMISSIBLE RADIATION DOSES FOR HUMAN BEINGS

Gunnar Walinder, FOA, Stockholm, 1977-11-04

SUMMARY

Maximum permissible dose levels are primarely based on risks for genetic damage and cancer. The reason for this is the observation that such late effects of radiation seem to arise even after doses that are too low to give rise to acute effects.

In contrast to the tumour incidence found in irradiated human populations no genetic effects of radiation have been observed in man. This does not mean that genetic effects have not been induced but that it has been impossible to find an increase or to discern them among all the congenital defects, that can not be ascribed to the irradiation. As a consequence, the radiological risk estimation has been concentrated on the hazard of malignant diseases.

In some cases epidemiological evidence indicate a more linear relationship between tumour frequencies and radiation doses than what can be found in animal experiments. This does not imply that man is more radiosensitive than other mammals, but rather that human populations are more heterogenous with regard to genetic constitution, living conditions, etc - and thus in radiosensitivity - than inbred animals living under standardized laboratory conditions. It is accordingly necessary to base risk evaluations on epidemiological evidence.

Tumour risks are generally expressed as excess rates of incidence and

mortality per million persons per rem. These figures are, however, not obtained from direct epidemiological observations but have been calculated from such data under the assumption of a linear relationship between effect and radiation dose. This formal extrapolation of observed data involves an uncertainty which, of course, is proportionately greater for the calculated effects in the millirem range. However, although the calculated tumour risks can not be said to be founded on direct scientific evidence, there are scientific reasons to believe that the figures derived from the formal extrapolations constitute an upper limit of possible risks.

KBS Technical Report No 59

TECTONIC LINEAMENTS IN THE BLATIC FROM GÄVLE TO SIMRISHAMN

Tom Flodén, University of Stockholm, 1977-12-15

Continuous seismic reflection profiling has outlined the pattern of tectonic lineaments around Sweden. The location of the Swedish coast in the Bay of Bothnia is tectonically derived, the largest vertical displacement observed in the Nordingrå region. The NE coast of Uppland is tectonically influenced and the Uppland coast of the Åland Sea is followed by large vertical faults.

S of the Mälaren-Gulf of Finland line the bedrock dips gently away from the Swedish coast, which is only to a small extent tectonically influenced. The sea area W of Gotland forms part of the large tectonic block of SE Sweden. In the Baltic Sea, this block is boundered to the N by the Bråviken Bay fault and by faults in the Landsort Trench - N Gotland area. Towards the S it is limited by an E-W hinge line in the S Öland area. The tectonic block is divided in two parts by the NW-SE Värmland-Västervik tectonic zone that extends across the Baltic Sea from Västervik in the NW to Klaipeda in the SE.

Outside the Blekinge coast the basement dips gently towards the horst systems of the Fennoscandian Border Zone in the S.

Neotectonic structures are recorded in the Baltic Sea, but their occurrence is considered to be small. The majority of the pre-Quaternary fractures in the Baltic Sea show no evidence of Neotectonic activity.

KBS Technical Report No 60

PRELIMINARY STUDIES FOR SITE CHOICE, BEDROCK STUDIES

Sören Scherman

GROUNDWATER CONDITIONS IN THE NORTHEASTERN SECTOR OF THE FINNSJÖ DISTRICT

Carl-Erik Klockars, Ove Persson
Geological Survey of Sweden, January 1978

ABSTRACT

Bedrock investigations were performed during the years 1976 and 1977 in order to evaluate the possibilities for disposal of radioactive waste in the crystalline bedrock of Sweden.

The work comprised mainly geological and structural surface mapping and evaluation of 5000 m of drillcore. Each core extended to a depth of about 500 m, a possible storage level. Due to the importance of ground water movements, the chief emphasis in the study, both on and beneath the surface, was placed on the fracturing pattern and the mineralization on fracture surfaces.

The investigated areas are distributed along the south-east Swedish coust. These areas were chosen since they lie in the vicinty of two nuclear power stations, Forsmark and Oskarshamn, and they are composed of rock types commonly found in the crystalline bedrock of Sweden, that is granite (young and old) and gneiss, which do not have any future economic and/or mining interest.

The general map (figure 1) shows the areas investigated and the seismic activity in Sweden, measured during 1951 - 1976.
Results are shown in figures and appendices for each area - Kråkemåla, Ävrö, Finnsjön, Forsmark and Karlshamn. It is obvious that the fracture pattern evident in the plastic and ruptural

deformation of the bedrock and in the topography can be extrapolated to great depth. At Finnsjön, depressions in the bedrock have been found at great depth during the core drilling. Mapping of the cores indicate that the fracturing of the bedrock does not decrease with depth but is unchanged through most of the cores.

Investigations with gephysical well-logging methods and TV-inspection verify the results described above. Results from the water pressure testing show quite a good correlation when compared with the fracturing in the core. However, in some places a high fracture frequency does not correspond to increasing k-values. This can be explained by the presence of clay minerals which can seal individual fractures and crush zones. Errors in core measurements can also occur during water pressure testing; this may also explain the absence of a direct correlation between fracture frequency and water-flows.

An increase in water-flow is observed in the lowest part of the borehole, situated in the coastal gneisses at Karlshamn. However, the fracturing of the core is extremely low and the observed increase is explained by the preasumed vicinity to more granitic rock-types and a contact between granite and gneiss which both tend to increase the permeability of the rock.

KBS Technical Report No 61

PERMEABILITY DETERMINATIONS

Anders Hult, Gunnar Gidlund, Ulf Thoregren

GEOPHYSICAL BOREHOLE SURVEY

Kurt-Åke Magnusson, Oscar Duran
Geological Survey of Sweden, January 1978

Abstract

Water pressure testings have been carried out in boreholes at Kråkemåla (K 1, K 2, K 3), Ävrö (Ä 1, Ä 2), Finnsjön (Fi 1, Fi 2, Fi 3) and Karlshamn (Ka 1), for determination of the permeability in the bedrock.
The tests are carried out in 2 m or 3 m sections delimited by packers. Water is pumped out between the packers. The water flow and pressure (ca 0,2 MPa) is measured and the permeability is calculated according to Bank (1972) and Moye (1967).

The minimum permeability is set by the smallest amount of water which can be measured by the water flow meter (0,002 l/min).

The results are shown in diagrams no. 1 - 9. The permeability is given in a logarithmic scale. All holes except Fi 2, Fi 3, K 3 and Ka 1 are vertical. Fi 2, Fi 3 and K 3 have an intended inclination of 50^0, Ka 1 has an intended inclination of 80^0. However measurements show that the inclination in Ka 1 is close to 90^0. The lengthscale could therefore be interpreted as depth in this hole. For the other three holes scales of both length and vertical depth are shown.

The fractures are not evenly distributed in the bedrock. Hence

one joint in one section can give the same permeability as several fractures in another section. The diagrams show that a considerable difference in permeability can be forthcoming between adjacent sections.

As can be seen in the diagrams the results can differ somewhat between different areas and within the areas.
Calculated k-values are often less than ca 2×10^{-9} m/s, which is the lower limit for the measurements (4×10^{-10} m/s in some holes). Most k-values are less than 1×10^{-6} m/s but higher values are found. Measurements of Ka 1 as one section from 23 down to the bottom did not show any water seepage which gives a k-value $\leq 2 \times 10^{-12}$ m/s. In Ka 1 the small flows measured in 2 m sections in diagram 9 are probably due to small local fractures that conduct the water from the section back to the borehole.

In several holes there is a greater amount of open fractures in the upper part of the bedrock than in the deeper. In the deeper parts, sections with water seepage are confined to zones where the fracture frequency is more intense to a certain degree. Between these zones there are considerable parts of the holes where there are no or just a few sections with water seepage.

KBS Technical Report No 62

ANALYSES AND AGE DETERMINATIONS OF GROUNDWATER AT GREAT DEPTHS

Gunnar Gidlund, Geological Survey of Sweden, 1978-02-14

SUMMARY

One part of the SGU-programme for the KBS is chemical analyses and dating of groundwater from depths of 500 to 1000 m.

The lack of a proper instrumentation has forced the SGU to an intensive development work during the spring and early summer 1977, which, among other things, resulted in a groundwaterpump, capabel of drawing up water from about 1000 m depth.

About 150 l of water has been collected for the ^{14}C-analyses. The carbonates have been precipitated with bariumchloride.These samples have also been analysed in respect to hydrochemistry and tritiumcontent.

Some of the samples have been collected in existing facilities (the Juktan-tunnel, the Håksberg mine and the Stripa mine), others in bore-holes drilled by the SGU (Kråkemåla, Forsmark and Finnsjön).

The dating of groundwater from the bore-holes indicates ages ranging from 2000 years to 11 000 years. However, the tritium- and hydrochemical analyses have given some evidence of contamination by the cooling water used in the drilling process.

KBS Technical Report No 63

GEOLOGICAL AND HYDROGEOLOGICAL DOCUMENTATION AT THE STRIPA RESEARCH STATION

Andrei Olkiewicz, Kenth Hansson,
Karl-Erik Almén, Gunnar Gidlund,
Geological Survey of Sweden, February 1978

ABSTRACT

At the request of the KBS (Nuclear Fuel Safety), the Swedish Geological Survey has carried out a geological documentation of the Stripa Test Station.
This report comprises: a summary of previous work; geological and joint mapping of the granite on the surface as well as in tunnels at the level of 330 and 360 m; mapping of 3 drillcores and TV-inspection of one drillhole regarding change in fissure frequency caused by blasting of a tunnel in the close neighbourhood.

The results of mapping of the granite in the investigated areas show that they are dominated by reddish, medium-grained, massive monzogranite, but that they show different degrees of tectonisation.

The granite on the surface, in the ventilation tunnel and in the northern part of the lower tunnel (360 m level) is, in contrast to the upper tunnel (330 m level), partly strongly crushed and sometimes also brecciated.
The orientation of the joints is similar in all investigated areas.

There is an absolute dominance of north- and almost vertical-dipping fractures. However the northern part of the upper tun-

nel is an exception as south-dipping fractures are more numerous than the north-dipping ones. Two strike directions dominate: one with a rather even distribution from N to E and the second concentrated in a smaller sector from NW to W.
Most of the joints are closed. They are mostly lined with clorite, occasionally with calcite. Water leakage is very low - only drop or moisture surfaces.

The mapping of the 3 drillcores shows that they are dominated by reddish, medium-grained, massive monzogranite. The frequency of joints and joint sets is relatively high and unevenly distributed.
The joints are usually lined with clorite, occasionally with calcite. TV-inspection of the drillhole did not yield satisfactory results due to technical reasons.

The hydrogeological investigations have included the following:

- Water injection tests in Dbh 2 before and after driving a drift parallell to the borehole
- Hydrostatic pressure tests in Dbh 2
- Determination of permeability in DbhV 1
- Water chemistry and water datings in Dbh 2 and DbhV 1

The results of the water injection tests in Dbh 2 before and after driving the drift proved to be difficult to evaluate. The borehole had an inclination upwards, which made it impossible to evacuate the air during water injection. The effects of fractures,that may be induced after driving the drift will be completely covered by the effects of air compression in the hole.

The hydrostatic pressure tests in Dbh 2 were made in 3,71 m long sections in that part of the borehole, which is situated beyond the drift(45 - 97 m).
Most of the pressure curves were difficult to interpret. The hydrostatic pressure in the section 89 - 97 m (end of the hole) was calculated to 1,67 MPa. Near the end of the drift (46,00 - 49,71 m) the pressure was 0,22 MPa.

The borehole DbhV 1 is drilled vertically from the 410 m level, far below the groundwater level. Therefore there is a hydrostatic overpressure around the hole, and water is continously flowing out from it.

The permeability tests were performed by measuring the waterflow and hydrostatic pressure in different levels. Sections of 6,68 m were isolated by two packers, and a plastic tube conducted the water from each section up to the 410 m level. A pressure transducer with a monitor and recorder registrated the hydrostatic pressures after the plugging of the tube.
The calculated average permeability was $6{,}5 \times 10^{-10}$ m/s.

Professor Peter Fritz, who is connected with the LBL-program at Stripa, has kindly made available the results of some hydrochemical analyses for our use in this report. The water samples have been taken from surface waters, shallow groundwater and deep groundwater. Specific intervals have been packed off giving a major part of the samples a well defined sampling depth.

The number of water samples is limited. There has been no time available for making a study of the change of chemical composition of the groundwater through time. Therefore the results must be regarded as preliminary.

pH, Na and Cl increase with depth.

Extremely high concentrations of Rn have been measured in the deep groundwater. Contents in the order of 2 u Ci/l have been noticed.

The tritium concentrations show a significant decrease with depth. A content of less than 1 TU has been measured in samples of deep groundwater. This might indicate that the groundwater is not contaminated by cooling water from the drilling process.

The ^{14}C-analyses of the deep groundwater indicate an age of 30 600 years at a depth of 360 m and 23 300 years at a depth of

410 m. After "correction" for the ^{13}C-content, these ages are reduced to 27 000 years and 23 300 years respectively.

The values of the quotient $^{13}C/\,^{14}C$ are such that the groundwater seems to have been infiltrated during an age with an activ vegetation cover.

The ^{18}O-content of the deep groundwater can be interpreted in two ways. One alternative is that the recharge of the water occurred 600 m higher than the elevation of Stripa. Another, and perhaps better explanation, is that the precipitation infiltrated locally during a climatic period with an average airtemperature 1 or 2 ^{o}C below the present one.
The different dating methods of the deep groundwater may, taken together, indicate, that the water infiltrated during an interstadial period of the latest ice age, the so-called Yngre Dösebacka-period between Weichsel II and Weichsel III.

KBS Technical Report No 64

STRESS MEASUREMENTS IN SCANDINAVIAN BEDROCK - PREMISES, RESULTS AND INTERPRETATION

Sten G A Bergman, Stockholm, November 1977

SUMMARY

In situ stress measurements in Scandinavian bedrock are reviewed and discussed. The conditions precedent in order to obtain reasonably reliable three - dimensional stresses in the rock mass space mosaic structure are analyzed. It is concluded that measurements with the LEEMAN three - dimensional or door - stopper method and with the modified LEEMAN - HILTSCHER method give good results due to the fact that the deformations are measured within a very small volume (less than a few hundreds of cm^3). The effects from usually fairly high perturbation block - bound stresses can be equalized by using the average value for more than 8 - 10 measuring "points" along a borehole. Experience shows that such average values have a fair consistence.

The method used by HAST combines one-dimensional measurements from various points. The block-bound perturbation stresses are not equalized. It is therefore probable that this method tends to overestimate the stresses. This conclusion is supported by field tests comparing the Hast method with the afore-mentioned methods and also by the fact that, as shown in Fig.4, the HAST in situ stresses generally are higher than those measured by other methods.

Generally, however, it can be stated that the stress measurements indicate higher horizontal compression stresses and shear stresses than should be expected from elastic theory. Most of the stress values obtained with the measuring methods judged to be reliable

can be interpreted as residual stresses caused by creep phenomena in the rock mass during late glaciations.

It is concluded that it will probably be fairly easy to find Swedish bedrock, where qualified rock tunnels (chambers) with moderate span can be cut out at 100-1000 m depth without any manifest stability problems, excavation difficulties or heavy reinforcements to be anticipated from rock stress conditions.

KBS Technical Report No 65

SAFETY ANALYSIS OF ENCAPSULATION PROCESSES

Göran Carleson, AB Atomenergi, 1978-01-27

Summary

The radiological safety in an industrial factory with a capacity of 2 tons of uranium per day was thoroughly investigated and analysed in regard to interior as well as exterior environmental effects. Two recently developed Swedish proposals were studied. The fuel elements are either directly encapsulated in a copper container, or by isostatic high pressure in a sintered aluminium oxide container. The analysis was performed for all the process steps, which include reception station with submerged storage racks, dismantling of the fuel elements in a water pool, encapsulation in dry concrete cells, storage of the product, and delivery station.

The activity discharge to the ventilation system as well as the dose loads to the plant staff and to the external environment was calculated for yearly normal discharges and conceivable accidents. The calculations were based on probability estimates and those safety precautions which were judged as necessary to obtain acceptable radiation doses.

The global dose commitment to the external environment amounts to 10^{-3} manrem/year. Extremely rare activity discharges can cause individual radiation doses of 40 µrem for people living close by. The individual dose load to the working staff is on average 200 mrem per year. The difference in dose commitments between the two encapsulation processes can be regarded as negligable.

KBS Technical Report No 66

VIEWPOINTS ON THE MECHANICAL RELIABILITY OF A CANISTER FOR NUCLEAR WASTE

Fred Nilsson
Royal Institute of Technology, Stockholm, February 1978

Summary

Three different concepts for encapsulation of used nuclear fuel are discussed with respect to their mechanical safety.

In the first concept the burnt out fuel elements are encapsulated into a copper capsule. The material properties of copper are discussed especially with reference to toughness and creep. A simple fracture mechanical analysis shows that the risk for direct fracture is negligible at the actual stress levels. The loads on the capsule are studied and are found to be normally less than 40 MPa (residual stresses). Transient loads that might arise in the handling of the capsule might however be dangerous to its integrity.

The next concept is encapsulation of the fuel elements into a sintered aluminium oxide capsule. A fracture probability analysis based on Weibull's statistical fracture theory gives fracture probabilities that are acceptable. Extended studies of this concept, especially of the risk for delayed fracture, is recommended.

The last concept is a Pb-Ti capsule for glassed refined fuel. An analysis of the relaxation of internal stresses is performed. The critical point of these capsules appears to be the welds on the titanium shell where the risk for a direct fracture is not negligible.

KBS Technical Report No 67

MEASUREMENT OF GALVANIC CORROSION BETWEEN TITANIUM AND LEAD AND MEASUREMENT OF CORROSION POTENTIAL OF TITANIUM UNDER GAMMA RADIATION
3 TECHNICAL MEMORANDUMS

Sture Henriksson, Stefan Poturaj,
Mats Åsberg, Derek Lewis,
AB Atomenergi, January-February 1978

Summary

Measurements of the redox potential on electrodes of platinum and the corrosion potential on titanium in a simulated γ-irradiated environment for disposal have verified that titanium remains passive in the pH interval established in a mixture of bentonite, quarz sand and water. The potential of titanium is not influenced by γ-rays, whereas the platinum potential is lowered 100 mV at radiation.

Redox potentials have been calculated for water containing low concentrations of hydrogen peroxide, which can be formed by radiolysis.

The galvanic corrosion of lead in contact with titanium has been measured in water saturated with air at 80° C, with and without buffer material. The objective is to study the galvanic corrosion of the lead canister if lead is uncovered by local corrosion of the titanium canister.

The investigation shows several factors of practical interest to reduce corrosion of lead:

- the establishment of a protective surface layer in ground water containing carbonates;

- a certain corrosion reduction effect of buffer materials;

- effective cathodic polarisation of titanium in the direction to lead;

- reducing corrosion as a larger surface area of lead is uncovered.

The oxygen content during the experiments was 8 ppm O_2 which is about 1 000 times higher than the oxygen content in the final storage. An extrapolation to this low oxygen content shows that the rate for local corrosion of lead corresponds to 5 μm/year in the final storage. The thickness of lead is 10 cm.

KBS Technical Report No 68

DEGRADATION MECHANISMS IN CONNECTION WITH POOL STORAGE AND HANDLING OF SPENT NUCLEAR REACTOR FUEL

Gunnar Vesterlund, Torsten Olsson
ASEA-ATOM, 1978-01-18

SUMMARY

This report deals with potential mechanisms for the degradation of light water reactor fuel in water pool storage. The assessment is made that neither general corrosion, local corrosion, stress corrosion nor hydrogen embrittlement will cause any significant degradation of the fuel and the fuel cladding within 50 years of storage. It is also concluded that no hazard is involved in storing defective fuel in the same manner as non-defective fuel as the degradation will not continue at low temperatures and the water leaching of fission products in the fuel is slow.

No problems are anticipated in the handling of high burn-up fuel. Large scale experience and well established routines exist for such handling. By experience the dose burden of personnel involved in handling of even newly discharged reactor fuel is low.

It is also shown in the report how other activated materials in the fuel assemblies safely can be taken care of.

KBS Technical Report No 69

A THREE-DIMENSIONAL METHOD FOR CALCULATING THE HYDRAULIC GRADIENT IN POROUS AND CRACKED MEDIA

Hans Häggblom, AB Atomenergi, 1978-01-26

Introduction

When the water flow due to hydraulic gradients is calculated in two dimensions it is found that outlets into the ground surface are always obtained close to the bottom of a valley [1, 2]. This is not a real physical situation, because the valley would then be replaced by a lake. The drainage of the valley is of course due to a gradient in the perpendicular direction. This is an important reason for making three-dimensional calculations. Another reason is the ability to make realistic calculations of the dilution of migrating nuclides in a three-dimensional model. The obstacle for such a model is its complexity which may lead to long running times for computer programs with purely numerical solution methods. Much benifit can, however, be obtained from existing analytical two-dimensional solutions [2]. In this work it will be shown how such solutions can be used for synthezising three-dimensional solutions. It is expected that the method can be a base for a computer program with relatively short running times. The programming technique is also outlined.

KBS Technical Report No 70

LEACHING OF IRRADIATED UO_2 FUEL

Ulla-Britt Eklund, Ronald Forsyth
AB Atomenergi, 1978-02-24

Summary

Leaching tests over a period of 105 days have been performed on 20 mm long sections of an irradiated fuel rod. The sections - fuel with clad - were selected from axial locations along the rod corresponding to average linear heat ratings of 11.2 and 23.5 kW/m. For each fuel type, two leachant solutions were used: a synthetic Swedish groundwater and distilled water as reference. Leaching was performed at $60^{o}C$ without mechanical stirring.

All the leached species which were measured (U, Sr-90, Cs-137 and total alpha activity) showed large initial leaching rates - expressed as fractions of the total inventory leached per day - but by the end of the leaching period, values were obtained of about 10^{-6}/d for U, Sr-90 and Cs-137 and about 10^{-7}/d for total alpha activity.

Comparison of the cumulative amounts leached out during 105 days between the highly-rated and low-rated specimens showed the greatest difference in the case of Cs-137 where values of about 0,7 % and 0,03 % of the total Cs-137 inventory were obtained. These values demonstrate the sensitivity to heat rating of the movement of Cs-137 during reactor operation to the pellet-clad gap and pheripheral cracks.

KBS Technical Report No 71

ROCK FISSURE SEALING WITH BENTONITE

Roland Pusch, Luleå Institute of Technology, 1977-11-16

If the rock which surrounds the deposition holes with radioactive canisters can be effectively sealed, they will be situated in a practically impervious medium with very favourable diffusion properties.

This report presents a new technique for sealing narrow rock joints by applying electro-phoresis. The idea is that 4-8 holes for electrodes are bored around the central deposition hole. A bentonite slurry with a water content of 500-2000% is applied in the central hole and is injected into the joints by applying an under-pressure in the surrounding holes (an over-pressure in the central hole can be applied as well). Then, anodes are inserted in the outer holes and a cathode in the central hole and a direct current of 200-2000 V (0.5-5 V/cm) is applied for one or two weeks. The deposition hole is cleaned after which the application of highly compacted bentonite bodies and canister can begin.

The report describes a series of preliminary tests with parallel glass plates simulating rock joints with a width of 0.01-0.2 mm. It was found that the "electro-kinetic" treatment had a considerable sealing effect. Thus the percolation rate value was reduced to 1/1000 or less of the value of the open passage.

The report also describes the preliminary results of an investigation where fairly dry, highly compacted bentonite confined in a cylindrical space was exposed to water through narrow slots. The slots (0.025-0.5 mm wide) simulated joints opened by stress changes in the rock where the deposition holes with highly compacted bentonite and canisters are situated.

The idea was to find out if the water uptake in the bentonite would cause a rapid swelling and extrusion through the slots. After 10 days the device was opened and it was found that bentonite had been extruded only through the 0.5, 0.3 and 0.15 mm wide slots. In the first-mentioned slot the extrusion depth was about 8 mm. The rate of extrusion was found to decrease rapidly.

KBS Technical Report No 72

THERMAL CONDUCTIVITY TESTS ON BUFFER MATERIAL OF COMPACTED BENTONITE

Sven Knutsson, Luleå Institute of Technology, 1977-11-18

The report concerns the thermal conductivity of highly compacted bentonite which has been suggested as embedding substance for radioactive canisters.

The first part of the report gives a short summary of the theoretical relationships associated with the different heat transfer mechanisms in moist granular materials. The method of calculating the thermal conductivity of soils given by Ø Johansen is referred to as well.

Chapter 3 describes the experimental determination of the conductivity of a cylindrical body of compacted bentonite (diameter 105 mm, height 313 mm). The density of the bentonite was 2.02 t/m^3 which corresponds to a degree of water saturation of 66% since the water content was 11%.

Three tests were run at constant water content but at different mean temperature. No time dependence was found for the thermal conductivity, which implies that no water transportation was caused by the temperature gradient. A slight increase of the conductivity with increasing temperature was found, the maximum value being 0.78 W/m, K for

$73^{o}C$. The stable water distribution is due to the strong bonds between water molecules and mineral surfaces when the water content is as low as 11%.

Complete water saturation corresponds to a water content of 16.5% which still is a very low value indicating that the degree of water fixation is high also in this state. This means that practically no water movement will take place when the mass is saturated. Consequently the mass will have a thermal conductivity which can be estimated at 0.9-1.3 W/m, K.

KBS Technical Report No 73

SELF-INJECTION OF HIGHLY COMPACTED BENTONITE INTO ROCK JOINTS

Roland Pusch, Luleå Institute of Technology, 1978-02-25

Although much work remains to be done in order to obtain relationships of general validity, a few statements can be made here:

- In the narrow joints (width smaller than 1 mm) that can possibly be opened by various processes, the rate of bentonite extrusion will be very slow except for the first few centimeter move which may take place in a few months.

- Even after several thousand years the extrusion of fairly dense bentonite will probably not exceed one or a few decimeters in such narrow joints.

- In the outer part of the bentonite zone there will be a successive transition to a very soft, diluted bentonite suspension. The suspension will probably be a gel. However even if there will be a sol transition it will not consist of individual 10 Å montmorillonite sheets but of fairly large particle aggregates which will be stuck where the joint width decreases. Here, such aggregates will collect, thus forming a dilute, but fairly impervious and very surface-active fill.

- The loss of bentonite extruded through such narrow joints is negligible.

- The reduction of swelling pressure in the highly compacted bentonite which is still in the deposition hole will be insignificant.

- The swelling pressure of the extruded bentonite will decrease rapidly with the distance from the deposition hole. Already at a distance of one or two decimeters it will be less than 3 MPa while the pressure is still 10 MPa in the deposition hole.

KBS Technical Report No 74

HIGHLY COMPACTED NA BENTONITE AS BUFFER SUBSTANCE

Roland Pusch, Luleå Institute of Technology, 1978-02-25

Excellent barrier functions of repositories for storing highly radioactive waste products can be expected by using highly compacted Na bentonite. The report describes the result of a number of investigations of the mechanical and physical properties of such montmorillonite-rich clay material, compacted to a bulk density of about 2-2.3 t/m^3.

The main basis for understanding these properties is that montmorillonite clay has a tremendous affinity to water, the uptake of which yields strong swelling. If expansion of very dense clay of this type is prevented, as will essentially be the case in deposition holes with radioactive canisters, the water uptake from the confining rock will produce a swelling pressure which is strongly dependent on the density. The swelling ability means that a perfect, close contact is established between bentonite and rock and bentonite and radioactive canisters, respectively. It also means that joints, which extend from the deposition holes and which may be opened because of changed stress or temperature conditions, will tend to be filled with extruded bentonite. This beneficial property of highly compacted bentonite, as well as its extremely low permeability and low diffusivity, provides an excellent isolating power.

KBS Technical Report No 75

SMALL-SCALE BENTONITE INJECTION TEST ON ROCK

Roland Pusch, Luleå Institute of Technology, 1978-03-02

Conclusions

The main conclusions from this investigation are:

* Pressure-injection of dilute bentonite suspensions ($w \sim 1000$) has a sealing effect but the depth of extrusion of the suspension into rock joints is probably not large because of gelation of the strongly thixotropic substance.

* Electro-kinetic injection of montmorillonite particles from a Na bentonite gel is a promising technique for rock tightening. It was found to be more effective than the pressure-injection. This was probably due to the electro-phoretic particle transport far out into the joints, directly from the central hole through the pressure-injected bentonite, and from the outer part of this bentonite.

* Since the clay particles have a net negative electrical charge they are transported towards the anodes, which should therefore be arranged around the central hole with the cathode. In practice, the joint system of the rock mass is not known in detail

when the location of the bore holes with anodes is to be decided. This is of no importance since the electrical field is continuous and transports clay particles towards the anodes in any kind of passages, provided that they begin at the deposition hole or are connected with joints which start there. A further testing of the technique on a larger scale should now be made.

KBS Technical Report No 76

EXPERIMENTAL DETERMINATION OF THE STRESS/STRAIN SITUATION IN A SHEARED TUNNEL MODEL WITH CANISTER

Roland Pusch, Luleå Institute of Technology, 1978-03-02

The effect of a differential movement triggered by a critical deviatoric stress condition has been investigated theoretically in a previous study, which showed that such movement can be avoided if no canisters are placed across potential slip planes in the rock. Yet, the study of the "incredible case" of such movement has shown that the canister stresses can be at least roughly estimated by applying Meyerhof's and Berezantzev's theories for deeply buried foundations. The discrepancies between calculations using these two theories was considerable, however, which called for an experimental study, the result of which is presented in this report.

A model canister (length 96 mm, diameter 16 mm) was equipped with strain gauges and embedded in a 10% bentonite/90% Pite silt buffer mass compacted to a dry density of about 1.4 t/m^3 in a shear box. Before shearing, artificial ground water was absorbed for 2.5 weeks which yielded about 50% water saturation. Ten stepwise displacements (0.5% of the 70 mm shear box diameter) were followed by ten 1% displacements under constant volume conditions. Each displacement was kept constant for 1-2 minutes. The measured canister strains and stresses together with the deformation pattern observed in previous tests, suggest that Meyerhof's theory can be used for practical application. It is concluded, however, that canister stresses can only be very roughly estimated in this way. More exact information requires finite element analysis based on a formulation of a reasonable stress/strain buffer mass model. Also, half-scale shear tests are recommended.

KBS Technical Report No 77

NUCLIDE MIGRATION FROM A ROCK REPOSITORY FOR SPENT FUEL

Bertil Grundfelt, Kemakta konsult AB, Stockholm
1978-08-31

SUMMARY

A study of the migration of radionuclides from a repository for spent, unprocessed fuel is presented. The study makes use of a unidimensional dispersion model developed at BNWL.

The results show that a number of nuclides decay significantly during the migration. The doses to future man was calculated in separate study performed at Studsvik. The dose calculations are based on the activity in-flows, presented in this report, and show that the predominant dose contribution comes from the nuclide radium-226. This nuclide is formed mainly by the decay of uranium-238 which means that the main part of the dose would arise even from a repository for non-irradiated fuel.

KBS Technical Report No 78

EVALUATION OF RADIOLYSIS IN GROUNDWATER

Hilbert Christensen, AB Atomenergi, 1978-02-17

Summary

Computer calculations of the radiolysis of ground water have been carried out with regard to the final disposal of high-active waste.

Radiolysis of water outside the container of waste encapsulated in glass results in an equilibrium situation after less than 50 hour. The equilibrium concentrations of oxygen, hydrogen, hydrogen peroxide are low, in the order of μM, correpsonding to 20, 10 and 20 ppb, respectively. Radical concentrations are much lower, $C(OH) \leq 2 \cdot 10^{-12}$ M and $e^-_{aq} \leq 5 \cdot 10^{-15}$ M.

For fuel elements directly deposited in a copper container the radiolysis of water inside the container has been studied by calculations. The purpose has been to determine whether hydrogen could be formed in concentrations exceeding the solubility limit and also to calculate the amount of copper corrosion. Assuming that oxygen reacts diffusion controlled with the copper surface it was found that hydrogen was formed in concentrations far below the solubility limit even after several million years. Zirconium may possibly react with water below $100^{\circ}C$ resulting in the formation of hydrogen at a rate of $1 \cdot 10^{-4}$ mole/year. The corrosion of copper is less than 50 mg per liter of water present.

KBS Technical Report No 79

TRANSPORT OF OXIDANTS AND RADIONUCLIDES THROUGH A CLAY BARRIER

Ivars Neretnieks
Royal Institute of Technology, Stockholm, 1978-02-20

Summary

The masstransfer rate for oxidants to, and radionuclides from a capsule in a repository has been computed. The capsule which is 0.75 m in diameter is surrounded by Montmorillonite clay. The hole is 1.5 m in diameter. For one capsule about 1220g copper will corrode due to oxygen corrosion in 10 000 years. If the fissures in the rock nearest the hole are filled with clay, the corrosion will decrease significantly. This is valid for a case where the groundwater is in equilibrium with oxygen of 0.2 bar pressure (normal air pressure). Measurements of the oxygen content in groundwater at large depths show a more than 1 000 times smaller values. The transport rate will then be correspondingly smaller. Corrosion due to sulphate/sulphide corrosion may reach some 590 g in the same time if there is 10 mg/l of the least abundant component.

The radionuclides Sr^{90}, Cs^{137}, Am^{241} and Am^{243} will decay totally in the clay barriers. Pu^{240} will be seriously hindered.

The total dissolution of the uranium oxide in a capsule takes at least 1.8 million years.

Nuclides with high solubilities decrease in about 2 000 years to half their original concentration.

The sodium in the Montmorillonite clay in the fissures is exchanged for calcium in about 20 000 years. The exchange of the sodium in the clay in the hole takes millions of years.

KBS Technical Report No 80

DIFFUSION OF POORLY SOLUBLE NUCLIDES FROM A CANISTER FOLLOWING CANISTER PENETRATION

Karin Andersson, Ivars Neretnieks
Royal Institute of Technology, Stockholm, 1978-03-07

Summary

Diffusion of nuclides of low solubility from a copper capsule after its penetration has been computed. It is assumed that by some mechanism the copper has corroded, leaving a hole equal to the inner diameter of the capsule. The uranium oxide is slowly dissolved and the nuclides diffuse out through the hole. The diffusivity of the uranium ions is taken to be 10^{-10} m^2/s in the copper corrosion products as well as in the porous mass in the capsule (UO_2, zircaloy corrosion products, lead corrosion products).

The dissolution of the uraniumoxide matrix is governed by the solubility of UO_2 and the distance it has to diffuse. The solubility is assumed to be very high - 1070 g/m^3 - which is the value obtained under oxidizing conditions and in a water with very high carbonate content (550 mg/l). Initially the transport rate of uranium is 2 g/year. It decreases rapidly as the uraniumoxide is dissolved and as the diffusion distance increases. It will take 2.8 million years to dissolve and transport away a mass of uraniumoxide in the two meters nearest the assumed hole in the capsule. If this hole has only 10 % of the inner area of the copper capsule the time increases to 7.1 million years.

When 0.2 m of the uraniumoxide nearest the hole have been

dissolved, the dissolution rate is smaller than the transport from the repository. The rate of diffusion from the capsule may than become the rate determining step.

KBS Technical Report No 81

FABRICATION OF COPPER CANISTERS FOR FINAL STORAGE OF SPENT NUCLEAR FUEL

Jan Bergström, Lennart Gillander, Kåre Hannerz
Liberth Karlsson, Bengt Lönnerberg, Gunnar Nilsson
Sven Olsson, Stefan Sehlstedt

ASEA, ASEA-ATOM, June 1978

SUMMARY

This report describes the manufacturing method for the copper capsule, which is proposed by KBS for final disposal of spent nuclear fuel.

The design and the dimensioning with respect to stress calculation is shortly explained.

Alternative casting and forging methods are discussed and compared. The preferred method is conventional static casting and upsetting and strech forging of the block in order to obtain a fine granular structure. Available equipment in the steel industry are considered useful for those operations.

The machining of the capsule consists of turning and long hole drilling, which can be performed by available equipment in Swedish workshop industry.

Non-destructive tests can be made with ultrasonic technics as the capsule has a fine granular structure.

Inserting of fuel and filling with melted lead into the capsule are non-conventional processes, which are based on the experience from fuel handling in nuclear plants and on lead pouring in other industrial applications. Special lead pouring tests have also been made for this particular purpose.

The welding of lids is made by electron beam welding. Welding tests has been made with good result. The weld depth is on the limit of what is possible with the present Swedish equipment,

but this field is being developed very quickly and welding equipment with higher capacity is already available which will simplify this operation.

The ultrasonic test of the welds has been experimentally checked and does not involve any particular problems.

KBS Technical Report No 82

HANDLING AND FINAL STORAGE OF RADIOACTIVE METAL COMPONENTS

Bengt Lönnerberg, Alf Engelbrektson, Ivars Neretnieks
ASEA-ATOM, VBB (The Swedish Hydraulic Engineering Co, Ltd), Royal Institute of Technology, June 1978

Summary

As the first operation in accordance with the programme for final storing of non reprocessed nuclear fuel, the fuel elements will be dismounted and the fuel rods will be taken care of in the main process of the final storage plant.

After the dismounting of the fuel elements, the next stage is to undertake the final storing of the metal components, which have kept the fuel rods together. These components have been activated by the neutron flow within the fuel case and for this reason they emit certain amount of radioactive radiation.

The components are transmitted to a pool where they are cut into pieces, compacted and placed in wire baskets. These are transferred in a water channel to a cell, where the metal components are embedded into concrete blocks. Thus the baskets are placed in prefabricated concrete containers, after which the metal parts are embedded into cement grout, injected from the bottom of the containers.

The blocks are finally stored in rock tunnels constituting a storage similar to the repositories for vitrified waste and spent fuel, although somewhat simplified, taking advantage of the much lower amount of radioactive material in the case of metal components. Thus a depositioning depth of 300 m in rock is very much on the safe side and it is appropriate in this case to fill the tunnels with concrete, ensuring by its alcalinity a sufficiently low rate of dissolution of the metal and migration of radioactive substances.

KBS Technical Report No 83

HANDLING OF CANISTERS FOR SPENT FUEL IN THE FINAL REPOSITORY

Alf Engelbrektson, VBB (The Swedish Hydraulic Engineering Co, Ltd), April 1978

In accordance with the project for the final storage of non-reprocessed spent nuclear fuel, the waste will be encapsulated into copper canisters (see Figur 2), which will be deposited in a final repository located in rock 500 m below ground level. The repository consists of a system of horizontal storage and access tunnels, connected to vertical shafts from ground level (see Figur 1). The canisters will be placed in vertical holes in the bottom of the tunnels, where the copper cylinders will be surrounded by blocks of highly compacted bentonite (see Figur 5).

The purpose of this report is to describe the transport of the fuel canisters from the encapsulation plant above ground level and the handling of the canisters until they are finally placed in their depositioning holes. The transport scheme is illustrated in Figur 3.

The repository is designed for a total of 6 400 canisters, each containing about 1.4 tons of spent fuel, and the transport system for a capacity of 1 - 2 canister transports per day.

After the encapsulation operation, a canister will be transferred by means of a transfer cart pulled by an electric tractor to a lift shaft for canister transports. Standing on the transfer cart, the copper cylinder is lowered to a level just below the repository, where the cart is pulled by remotely operated equipment into a reception room, covered by a concrete slab for radiation protection.

The transport to the depositioning hole is performed by means of a special transport truck (see Figur 6), provided with a radiation shielding tube (see Figur 7). The truck is placed on the floor above the reception room, the tube is raised to an upright position and the canister is lifted into the tube. After the tube has been lowered to a horizontal position, the truck is driven to the actual storage tunnel, the tube is raised above the depositioning hole and the canister is lowered into the pit, which has been previously lined with bentonite blocks.

The design of the transport system is based upon the basic principles that a complete radiation protection shall be provided during normal operations in the tunnels and that no radioactive emission or radiation protection problems should arise in consequence of transport accidents etc. Possible accident situations have been analysed. As the copper canister with its core of lead poured around the fuel rods is extremely robust, the only accident, which could seriously damage the canister, would be a fall of the transport lift from a higher level. The risk for such a downfall will be very slight, due to the conservative design of equipment and the installation of a mechanical lift catching system. Arrangements are foreseen, however, to catch a falling canister in a well below the lift shaft (see Figur 8). The well is filled with water and a falling canister is retarded by the flow force developed when water is pressed past the canister in the shaft section decreasing downwards. Finally the canister is stopped by a sand bed or a mechanical shock absorber at the bottom of the shaft. Another device for retarding a falling lift cage using the force of enclosed air in the shaft below the cage has also been studied and found feasible.

KBS TEchnical Report No 84

FABRICATION AND HANDLING OF BENTONITE BLOCKS

Alf Engelbrektson et al
VBB, ASEA, ASEA-ATOM, Gränges Mineralprocesser, June 1978

Summary

In accordance with the project for the final storage of spent nuclear fuel, the waste will be encapsulated into copper canisters, which will be deposited in a final repository located in rock 500 m below ground level. The repository consists of a system of horizontal storage and access tunnels, connected to vertical shafts from ground level. The canisters will be placed in vertical holes in the bottoms of the tunnels, where the copper cylinders will be surrounded by blocks of highly compacted bentonite. When the blocks are saturated with water and expansion is essentially retained as in the actual case, a very high swelling pressure will arise. The bentonite will be extremely impermeable and thus it will form a barrier against transport of corrosive matters to the canister.

The blocks are fabricated by means of cold isostatic pressing of bentonite powder, using high pressure equipment of a type, which is commonly used for the fabrication of hard metal, insulators, graphite blocks, refractory materials and other ceramics, etc. The base material in the form of powder is enclosed in flexible forms, which are introduced into pressure vessels where the forms are surrounded by oil or water. By means of a pumping system the liquid is subjected to very high pressures acting uniformly over the forms. Thus the powder is compacted into rigid bodies with a bulk density of about 2.2 t/m^3 for "air dry" bentonite, which might be compared with a specific density of about 2.7 t/m^3.

The plant for the fabrication of bentonite blocks should preferably be located at the site for the incapsulation plant above the final repository. The blocks will be transported by a lift from ground level to the repository, where they will be loaded on a special truck equipped with a crane. The truck can take a complete set of blocks to one depositioning hole, where

the placing is performed using the crane. The placing of a canister is preceded by piling up bentonite blocks to a level just below the canister lid position, after which the slot around the blocks is filled with bentonite powder. The rest of the blocks are mounted after filling bentonite powder into the inner slot around the canister as well.

Finally the storage tunnels will be sealed by filling them with a mixture of sand and bentonite, as described in ref. [2] and [3]. Preferably, the sealing of a tunnel should be carried out a short time after its depositioning holes have been filled up. Until the sealing of the tunnel, ground water is drained through a system of bore holes, for which reason there is only a slight risk of early water uptake and swelling of the bentonite. During a possible delay, however, the vertical displacements of the bentonite blocks should be measured. If any significant swelling should be observed, additional drainage holes can be drilled around the actual depositioning hole or the fill can be supported by temporary struts or permanent columns of piled granite blocks against the roof of the tunnel.

When, following the sealing of the storage, the bentonite is gradually saturated with ground water, swelling takes place until the swelling pressure of the bentonite fill in the holes is balanced by the reaction from the compressed sand-bentonite fill in the tunnel and by the friction forces at the rock surfaces. According to estimates, taking into account the initial voids in slots and joints between blocks as well as the time-dependent deformation of the tunnel fill above the hole, the final density of the bentonite blocks will amount to about 2.1 t/m^3. This means that the bentonite bodies below the canisters will maintain sufficient bearing capacity and that the permeability of the material around the canisters will remain extremely low [1].

KBS Technical Report No 85

CALCULATION OF THE CREEP RATE OF A LEAD JACKET CONTAINING A GLASS BODY UNDER THE INFLUENCE OF GRAVITY

Anders Samuelsson

ALTERATION OF CREEP PROPERTIES OF A LEAD JACKET AS A RESULT OF MECHANICAL DAMAGE

Göran Eklund
Institute of Metals Research, September 1977- April 1978

INTRODUCTION

Reprocessed, vitrified waste is proposed to be encapsulated in lead which then is encased in a titanium container with a tight fit. As the density of the glass is lower than the density of the lead, the glass will move upwards (float). The lead is then subject to creep deformation and the dominant mechanisms of deformation is diffusion creep and Harper-Dorn creep.

When the velocity of the creep is calculated it is found that the speed is strongly influenced by the grain size of the lead. At a grain size of 10^{-5} m it will take 1.8×10^2 years for the glass to move upwards 50 mm. With a grain size of 10^{-4} m the corresponding time is 1.8×10^5 years and at 10^{-3} m, 1.7×10^8 years.

Now the intention is to heat treat the lead before the final disposal of the waste so that the lead will have a grain size big enough to ensure that the glass will not rise to the surface within a very long period of time.

However, at the handling of the canister mechanical impact could damage the canister and expose the lead to mechanical deformation. This will cause the lead to recrystallize at which an area with smaller grain size is formed. It has then been suggested that the diffusion velocity for the lead atoms will increase locally and that there is a risk that the location of the glass in the lead encasement will be changed or that lead could flow out through a hole in the titanium shell.

For the reason some tests have been made in order to determine:

1. How small will the lead crystals become at a shock-like mechanical impact and at a slower mechanical deformation, as at a Brinell-impression ?

2. How rapidly will the grains grow at the deformed steel at the temperature which the canister will have in the Final Repository ?

EXPERIMENTS

The tests were made on pellets of "Boliden lead" with the dimensions, diameter 30 mm, height 20 mm. One pellet was exposed to a heavy blow with a rivet hammer and on another pellet a Brinellpimpresseion was made with a 6 mm ball. The pellets were cut in two halves after which the structure was exposed by etching. After documentation of the structure the pellets were heat treated during 70 hours at 80^{o}C after which the structure was again exposed by etching.

RESULTS

Fig 1 shows the initial grain size before the deformation with a rivet hammer, the grain size is here about 5 x 10^{-4}m After deformation the smallest grain size is 5 x 10^{-5}m, fig 2. A heat treatment at 80^{o}C during 70 hours will give an increase of the grain size and the smallest grain which then may be observed is about 10^{-4} m, fig 3.

Fig 4 and 5 show the structure with Brinell-impression before and after heat treatment.The fact that a big increase of the grain size has taken place during the relatively short time of heat treatment can here be observed.

CONCLUSIONS

A big deformation with high deformation velocity will not give such a decrease of the grain size that the creep velocity will become abnormally high. Furthermore a large increase of the grain size will occur at the temperature which the canister will have when placed in the Final Repository. The increase of grain size is considerable already after 70 hours. It is therefore possible to conclude that a mechanical deformation of the lead encasement will not give any deterioration of the creep properties and that apprehensions expressed in this respect have no basis in reality.

KBS Technical Report No 86

DIFFUSIVITY MEASUREMENTS OF METHANE AND HYDROGEN IN WET CLAY

Ivars Neretnieks, Christina Skagius
Royal Institute of Technology, Stockholm, 1978-01-09

Summary

One possible way to store the waste from a nuclear power plant is to encapsulate the radioactive waste in copper and surround the capsules with a barrier of compacted clay in a repository far below the ground surface. Oxygen and other oxidizing agents must be hindered from reaching the copper capsules. One major barrier for this transport is the diffusion resistance in the compacted clay surrounding the capsules.

An experimental determination of the diffusivities of methane and hydrogen through a barrier of compacted clay at a temperature of 50°C has been made. The methane molecule is somewhat larger than oxygen, while hydrogen is smaller. Methane and hydrogen were used instead of oxygen in the experiments to circumvent the experimental difficulties in measuring oxygen.

The measured diffusivity of methane in compacted clay was $3.9 \cdot 10^{-11}$ m^2/s and the diffusivity of hydrogen in compacted clay $1.8 \cdot 10^{-11}$ m^2/s. The ratio between the diffusivity of methane in water and in compacted clay was about 100 and the corresponding ratio for

hydrogen was about 500. The clay was compacted to a density of 2.1 g/cm^3 on an air dry basis.

The expected diffusivity of a molecule in a porous body of the same low porosity as that of the clay, is more than 50 times lower than the diffusivity in free water.

The diffusivity of oxygen in the compacted clay is expected to be of the same magnitude as that for methane.

KBS Technical Report No 87

DIFFUSIVITY MEASUREMENTS IN WET CLAY, NA-LIGNOSULPHONATE Sr^{2+}, Cs^{+}

Ivars Neretnieks, Christina Skagius
Royal Institute of Technology, Stockholm, 1978-03-16

Summary

Diffusivity measurement of cesium, strontium and sodiumlignosulfonate were made in a wet compacted clay. The density of the clay was 2.1 g/cm^3 on an air dry basis. The temperature was $50^{o}C$.

Strontium and cesium are ion exchanged into the clay and will thus be retarded. The measured diffusivities were $5.4 \cdot 10^{-11}$ m^2/s for cesium and $3.3 \cdot 10^{-11}$ m^2/s for strontium. These diffusivity values have been recomputed to describe a case with no ion exchange. In this way they are comparable with diffusivities of other small molecules. These are quite expected values and compare well with previous measurements with methane and hydrogen.

Sodiumlignosulfonate with a mean molecular weight of 24000 could not be found on the recieving end of the diffusion cell after more than 800 hours. The diffusivity then must be less than $3 \cdot 10^{-14}$ m^2/s.

KBS Technical Report No 88

GROUNDWATER CHEMISTRY AT DEPTH IN GRANITES AND GNEISSES

Gunnar Jacks
Royal Institute of Technology, Stockholm, April 1978

SUMMARY

Available data make it possible to asses the composition of the ground water at depth what concerns most of the components. The place considered for a storage is beneath a local water divide along the eastcoast of southern and central Sweden.

pH is fixed by the carbonate system and the ground waters can be expected to be saturated with respect to calcite. pH may vary from about 7,2 to about 8,5 with the most probable value around 8. The content of bicarbonate should be reversly proportional to the pH, and 3 meq./l or 180 mg/l can be expected as a mean value. Ca^{++} may vary from 10 to 100 mg/l with the most probable value at 40 mg/l. The content of Cl^- is difficult to asses as it is associated with the presence of relict sea water. Relict sea waters are linked to the postglacial clays and low sections of the terrain. Near local water divides the content of Cl^- is low approaching the atmospheric contribution or below 10 mg/l. At depth fluid inclusions in minerals may contribute some Cl^- when the residence time of the ground waters in the rock becomes drastically extended. The content of F^- is is restricted by the solubility of fluorspar and may be about 7 mg/l as a maximum, but normally 3,5 mg/l or lower. SO_4^{--} seems to be of the same order as the atmospheric contribution or about 15 mg/l. In the deep boreholes sampled so far the contents have been lower. This may be due to a lower atmospheric transport in the past or a microbial sulphate reduction. The latter

is not unlikely. The deep ground waters ought to be examined for sulphate reducing bacteria. The partial pressure of oxygen in the ground water should be very low. An increasing iron-content towards depth is likely. Organic substance in the order of a few tenths of mg/l may be present. It is in the form of fulvic acids with a complex-binding capacity of 10-15 meq./g. Inorganic colloids as Fe- and Al-Si-precipitates are quantitatively of minor importance. Part of the iron may be attached to the organic substance.

The local heating of the water close to the storage may bring about precipitation of calcite while the subsequent cooling when the water leaves the storage may result in precipitation of aluminiumsilicates. This may have a sealing effect on the rock.

KBS TEchnical Report No 89

INFLUENCE OF GLACIATION ON A WASTE REPOSITORY SITUATED IN PRIMARY BEDROCK 500 M BELOW THE SURFACE OF THE GROUND

Roland Pusch, Luleå Institute of Technology, 1978-03-16

Advancing glaciers are known to affect the bedrock to a considerable extent. Large rock volumes are known to have been displaced and disintegrated by such action. Two main effects are known: the shear stresses in the glacier/rock interface, and increased stresses in the interior of the bedrock.

The most obvious effect of the shear stresses are probably at or close to the glacier front and this is a condition which can be treated by applying the theory of plasticity. Using the Mohr/Coulomb criterion and Brinch Hansen's failure model consisting of combined Prandts and Rankine zones it is found that failure cannot be produced to more than a few meters depths even when the advancing glacier has a very steep and high front. It can easily be shown, however, that this is only valid for plane, horizontal ground surfaces. Hills and other rock obstacles are easily displaced along joints and weak zones. It is concluded that a repository at 500 m depth cannot be opened and exposed by glaciers if proper topographic conditions are chosen.

A thick ice cover, 3 km being a possible height, is shown to affect the existing stress situation at 500 m considerably. It could produce an increase of the vertical stresses to about 13 to 43 MPa and increase the horizontal stresses from 6-20 MPa to 20-35 MPa. This is equivalent to the present situation at 1500 m depth in Swedish bedrock and it means that critical tangential stresses may be produced at the tunnel and bore hole peripheries. If highly compacted bentonite is used in the deposition holes its high density and stiffness, as well as its swelling pressure (8-10 MPa), will partly balance the increased rock pressure and the rock will be less affected than around the tunnels. No major displacements or severe joint formation will probably take place in any part of the repository, however.

The most important effect of a glaciation will probably be caused by the varying glacier thickness which could initiate some differential movements along steep, continuous weak zones in the bedrock. The repository should therefore be situated in rock volumes separated by potential displacement zones.

KBS Technical Report No 90

COPPER AS AN ENCAPSULATION MATERIAL FOR UNREPROCESSED NUCLEAR WASTE - EVALUATION FROM THE VIEWPOINT OF CORROSION
FINAL REPORT 1978-03-31

The Swedish Corrosion Research Institute and its reference group

Summary

The Nuclear Fuel Safety Project (KBS) has proposed, to fulfil the requirements of the so-called "Conditions Act", that spent unreprocessed nuclear fuel shall be disposed of by encapsulation in copper canisters with 200 mm thick walls. The canisters are to be placed in vertical boreholes in rock, 500 m below the surface, and embedded in a buffer of compacted bentonite.

The Swedish Corrosion Institute has been assigned the task of evaluating this proposal from the viewpoint of corrosion and of estimating the life of the canisters under the given conditions. To do this work, the Corrosion Institute has appointed an expert group of 10 Swedish specialists, mainly from the fields of corrosion and materials technology.

The thermodynamic possibilities of various corrosion reactions on copper under the prevailing conditions were studied, whereby bacterial action was also taken into account. Oxygen entrapped in the buffer material at the time of the closing of the repository was found to be an important oxidant for corrosion. Sulphide in the groundwater was found to be another important reactant. The supply of oxygen and sulphide, mainly by diffusion, was calculated, and from this the maximum possible corrosion of copper. The morphology of the attack was also examined, the finding being that an attack which starts as pitting will penetrate into the metal at a decreasing rate,

after which the continued attack will proceed mainly in the form of a widening of the pits or an initiation of new pits. So the corrosion of a very thick-walled canister will, in the long run, have the character of a more or less uneven attack over the entire surface, with heterogenities not exceeding a pitting factor of 25. The expert group arrived at the conclusion that, under the given conditions, the canisters will last for hundreds of thousands of years.

The expert group was unanimous in its opinion, with the exception of Professor Gösta Wranglén, who has submitted a special statement of his own.

KBS Technical Report No 91

SHORT-TERM VARIATIONS IN THE PRESSURE LEVEL OF THE GROUNDWATER

Lars Y Nilsson
Royal Institute of Technology, Stockholm, September 1977

Summary

Investigations have demonstrated that the ground water level of aquifers in the Swedish bedrock shows short-time variations without changing their water content. The ground water level is among other things affected by

- regular tidal movements occuring in the "solid" crust of the earth
- variations in the atmospheric pressure
- strong earthquakes occuring in different parts of the world

These effects proves that

- the system of fissures in the bedrock are not stable
- the ground water flow is influenced by both water- and airfilled fissures

KBS Technical Report No 92

THERMAL EXPANSION OF GRANITOID ROCKS

Ove Stephansson, Luleå Institute of Technology, April 1978

ABSTRACT

The thermal expansion of rocks is strongly controlled by the thermal expansion of the minerals. The theoretical thermal expansion of the Stripa Granite is found to be $21 \cdot 10^{-6}$ $[^{0}C]^{-1}$ at 25 ^{0}C and $38 \cdot 10^{-6}$ $[^{0}C]^{-1}$ at 400 ^{0}C.

The difference in expansion for the rock forming minerals causes micro cracking at heating. The expansion due to micro cracks is found to be of the same order as the mineral expansion. Most of the micro cracks will close at pressures of the order of 10 - 20 MPa.

The thermal expansion of a rock mass including the effect of joints is determined in the pilot heater test in the Stripa Mine.

KBS Technical Report No 93

PRELIMINARY CORROSION STUDIES OF GLASS CERAMIC CODE 9617 AND A SEALING FRIT FOR NUCLEAR WASTE CANISTERS

I D Sundquist, Corning Glass Works, 1978-03-14

ABSTRACT

At the request of ASEA-ATOM, Sweden, a study was initiated to evaluate glass-ceramic Code 9617 and a sealing frit (glass Code 186AYU) to determine the suitability of these materials for use in the construction of a cannister to contain and permanently isolate spent nuclear fuel rods. The study was aborted after one quarter of the work was completed. Nevertheless results obtained define test conditions for future work and provide preliminary estimates of corrosion rates.

The granite repository storage conditions included ground water in contact with bentonite clay plus sand with temperature cooling from 100 to 40°C in 500 years. The water exchange rate is 1 ml per cm² of cannister surface per year. Tests were conducted at 60, 90, 130, and 180°C. Most of this preliminary work was done with distilled water with a few tests in water containing bentonite and sand or salt.

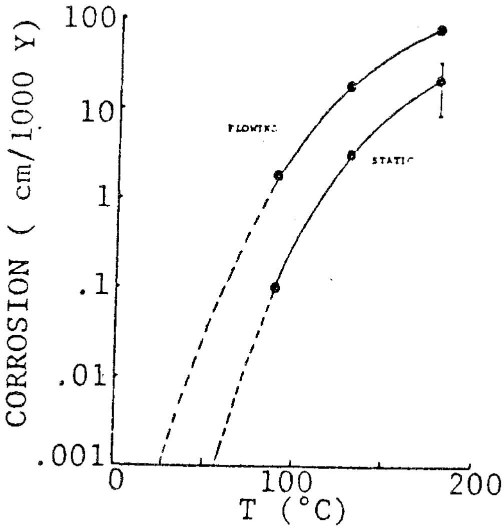
It is recommended that future tests be conducted under conditions which embrace the actual storage temperatures, thus allowing prediction of long term rates from an extrapolation of time rather than from the accelerated effect of increased temperature which our results indicate may alter the reaction mechanism. Other test conditions should also simulate real conditions as nearly as possible. Of particular importance is a consideration of the changing chemistry of the test solutions when "static" conditions are used; i.e., when the test solution remains in contact with the test specimen for long periods of time.

Preliminary data suggest that the glass-ceramic would lose about 0.1 cm of thickness in the first 1000 years of storage. The figure shown below

presents a comparison of estimated corrosion rates as a function of temperature for a flowing and a static water system.

There is little evidence to suggest that the corrosion rate in ground water/bentonite/sand mix or in salt water would differ by more than an order of magnitude.

It appears that the spodumene fraction of the glass-ceramic is very corrosion resistant, even at 180°C. This suggests that modifications in bulk composition or processing could dramatically decrease overall corrosion rates.



KBS Technical Report No 94

WASTE FLOWS IN REPROCESSING

Birgitta Andersson, Ann-Margret Ericsson
Kemakta, March 1978

The three main products from reprocessing operations are uranium, plutonium and vitrified high-level-waste. The purpose of this report is to identify and quantify additional waste streams containing radioactive isotops. Special emphasis is laid on Sr, Cs and the actinides.

How the fission-products (FP) U, Pu and other actinides (Am, Cm, Np) are divided between these three product streams is illustrated in the following table.

	FP	U	Pu	Other act.
HLW	99 %	0.1 %	0.5 %	> 99 %
Pu	< 0.01 %	< 1 %	99 %	< 0.01 %
U	< 0.01 %	99 %	0.01 %	< 0.01 %

The main part, more than 99 % of both the fission-products and the transuranic elements are contained in the HLW-stream. Small quantities sometimes contaminate the U- and Pu-streams and the rest is found in the medium-level-waste according to the table below.

	FP	U	Pu	Other act.
Hulls and hardware in bitumen	0.05 %	0.05 %	0.05 %	< 0.01 %
Saltcakes in concrete or bitumen	0.01 %	0.2 %	0.2 %	0.01 %
Filters etc. in concrete or bitumen	<1 %	0.5 %	0.5 %	< 0.01 %

KBS Technical Report No 95

SEPARATION OF C-14 IN REPROCESSING

Sven Brandberg, Ann-Margret Ericsson
Kemakta, March 1978

Carbon-14 is formed during the operation of nuclear reactors both in the fuel, in the cooling water and in the core hardware. The part from the cooling water is released to the off-gas-system of the reactor and amounts to about 5 Ci/GWe - yr. The quantity formed in the fuel is related to the presence of nitrogen. At 10 ppm N 12 Ci/GWe - yr is formed. It is released as CO_2 during dissolution of the spent fuel at the reprocessing plant.

The contribution of C-14 to the biosphere due to nuclear power is still small. However, on a long-term it will become of the same order as the natural formation if no counter-measures are taken.

Presently C-14 is not removed from the off-gases either at the reactors or the reprocessing plants.

Promising development is however going on. A pilot plant is in operation based upon absorbtion of CO_2 in a fluorocarbon solvent. A CO_2-decontamination of 99.99 % is obtained and Kr-85 and iodine are absorbed at the same time.

Other methods as caustic scrubbing and separation by molecular sieves are also considered. These methods have not been tested for this special purpose but are well-known in the chemical industry.

The quantity of C-14 released during reprocessing is small. If it is recovered as $CaCO_3$ and the dilution with C-12 is 20-fold the whole Swedish nuclear program of about 10 GWe will produce only 4 kg/yr. This quantity will contain 120 Ci as β-emitter with a half life of 5 730 years.

KBS Technical Report No 96

CORROSION TESTING OF UNALLOYED TITANIUM IN SIMULATED DEPOSITION ENVIRONMENTS FOR REPROCESSED NUCLEAR FUEL WASTE

Sture Henriksson, Marian de Pourbaix
AB Atomenergi, 1978-04-24

SUMMARY

On the commission of the Nuclear Safety Project (KBS) corrosion tests have been carried out on unalloyed titanium, which is planned to be used as an outer corrosion resistant canister for reprocessed nuclear waste. The corrosive medium was modified Baltic water at 100 and 130 °C with either a high (8 ppm) or a low (<10 ppb) oxygen content. The total duration of the tests was 300 days.

Very low oxidation rates (<0.1 μm/year) were obtained, corresponding to a life time of ten thousands of years for a 6 mm thick titanium canister. In spite of the considerably accelerated corrosive conditions, compared with those estimated for the final disposal, no signs of localized corrosion were found, nor could any hydrogen pick-up be detected.

The report expired is replaced by KBS technical report 79-14, in which a larger-scale test series is reported.

KBS Technical Report No 97

COLLOID CHEMICAL ASPECTS OF THE "CONFINED BENTONITE CONCEPT"

Jean C Le Bell, Institute of Surface Chemistry, 1978-03-07

Summary

A short review of concepts in colloid chemistry of relevance to the investigation is given. The basic principles of coagulation and swelling by electrolytes and swelling pressure are discussed. A survey is given of literature of relevance to the colloid chemical properties of the bentonite to be used as buffer mass for waste cannisters in the final respository.

Measurements of the amount of particles released from a bentonite gel by light scattering and visual inspection show that while particles are released in distilled water, the gel will be coagulated if in contact with ground water and consequently the release of particles is negligibly small.

Studies of sedimentation volumes by ultracentrifugation also clearly indicate that the bentonite in contact with ground water under the repository pressure will form a completely stable coagulated gel.

The swelling of confined bentonite was studied in an "artificial crack" of width 0.5 mm. The bentonite flowed readily into this crack and into the much narrower crack formed when the cell was broken. The swelling properties of the bentonite at the repository depth are discussed. It is argued that the gel, if

sufficient volume is available, will swell spontaneously to a volume that is ≈30% larger than the initial one and then form a stable, coagulated gel containing 30-35% water in equilibrium with the ground water.

Investigations of the diffusion of colloidal matter (sodium lignosulphonate molecules of mean diameter 6 nm) and calcium ions into a dilute bentonite gel show that colloidal matter very probably will have a negligible rate of diffusion while the calcium ions diffuse rapidly. This implies that the initial bentonite gel which is partially in its sodium form will be completely exchanged to its calcium form when brought into contact with ground water which ensures that it will remain coagulated even in its swollen state.

KBS Technical Report No 98

ABSORPTION OF LONG-LIVED RADIO NUCLIDES IN CLAY AND ROCK PART 2

Bert Allard, Heino Kipatsi, Börje Torstenfelt
Chalmers University of Technology, 1978-04-20

SUMMARY

Chemical equilibria of importance for the retention of radionuclides in the ground have been discussed and equilibrium constants have been collected for reactions in the subsoil water.

The sorption of the 14 elements previously investigated[1] Sr, Zr, Tc, I, Cs, Ce, Nd, Eu, Ra, Th, U, Np, Pu and Am has been studied at long contact times with granite and bentonite/quartz mixture.

The sorption of Cs, Sr, Am and Ag has been studied in pure bentonite as well as the sorption of Cs, Sr and Am on fresh and old rock surfaces (granite). Moreover, the sorption of Ni which is formed as an activation product has been studied on granite.

Possible sorption mechanisms have been suggested.

The influence of the redox potential of the water on the formation of different species and changes in sorption due to changes in valence states have been discussed and demonstrated for U and Tc.

The contents of organic complexing agents in subsoil water and their role in the nuclide migration have been discussed.

Aspects have been given on the mechanisms of dissolution of UO_2 in burnt-out nuclear fuel.

Using measured and calculated distribution coefficients for powdered rock and rock surfaces after long contact time and assuming reducing conditions the retention in the ground is discussed.

KBS Technical Report No 99

LEACHING OF HIGH-LEVEL RADIOACTIVE FRENCH GLASS STATUS REPORT 1978-06-01

Göran Blomqvist, AB Atomenergi, 1978-06-19

SUMMARY

The leaching of highly active French glasses at Studsvik, which have been described earlier, have continued. This report gives analytical data up to 1978-04-03. The leaching history covers several periods:

1.	1977-07-18 to 1977-08-15	Temp 25°C Dynamic leaching
2.	1977-08-15 to 1977-09-05	Temp 25°C Static leaching
3.	1977-09-05 to 1977-12-25	Temp 60°C Static leaching
4.	1977-12-23 to 1978-01-09	Temp 25°C Storage period
5.	1978-01-09--	Temp 25°C Dynamic leaching

LWR glasses containing Cs and Sr show considerable differences in the beginning which tend to diminish during stage 5. Observed leach rates are 1-2 x 10^{-6} g cm^{-2} day^{-1} at the end of observations. They have a tendency to decrease further.

Increase in temperature from 25°C to 60°C gave a leach rate increase with a factor of about 10. In all cases but one the leach rate at 60°C decreased considerably with time.

For Pu, the leach rate remained approximately constant at 5 x 10^{-7} g cm^{-2} day^{-1} during stages 1 - 3 and beginning of stage 5. After a few weeks of that stage there was a temporary strong increased of Pu leaching, that decayed back to about 5 x 10^{-6} g cm^{-2} day after about a month. This effect is still unexplainded but may be due to inhomogenity.

KBS Technical Report No 100

DOSE AND DOSE COMMITTMENT FROM GROUNDWATER-BORNE RADIOACTIVE ELEMENTS IN THE FINAL STORAGE OF SPENT NUCLEAR FUEL

Ronny Bergman, Ulla Bergström, Sverker Evans
AB Atomenergi, 1978-10-06

SUMMARY

The turnover of radioactive matter entering the biosphere with the groundwater has been studied with regard to exposure and doses for critical groups and populations.

The main alternatives considered for the outflow of radioactive effluents to the biosphere are:

- outflow in a valley containing wells and to a nearby lake
- outflow to a freshwater lake and to the downstream lake system
- outflow in a coastal region of the Baltic Sea

Mathematical models of a set of coupled ecosystems on, regional, intermediate and global levels have been used for calculations of doses. The intermediate system refers to the Baltic Sea.

The mathematical analysis, based on first order kinetics for the exchange of matter in a system according to compartment pricinples, also includes

products in decay chains, i.e. daughter nuclides generated by decay of nuclides during ecological cycling.

The time-dependent exposures have been studied for certain long-lived nuclides of radiological interest in waste from disposed fuel. Dose and dose commitment have been calculated for different episodes for outflow to the biosphere. The source strength is equal to the outflow at the boundary between geosphere and biosphere calculated by use of a geospheric model.

KBS Technical Report No 101

LEAKAGE OF NI-59 FROM A ROCK REPOSITORY

Ivars Neretnieks, Karin Andersson, Lennart Henstam
Royal Institute of Technology, Stockholm, 1978-04-24

Summary

Radioactive metal parts are compacted and placed in concrete boxes. The boxes are placed in a tunnel in good rock at 500 m depth. They are surrounded by a buffer mass consisting of a clay/quartz mixture with very low water permeability. The metal parts consist of zircaloy, stainless steel and inconel. The total amount of metal is 720 tons. 78 tons of this is nickel. The most important radionuclide in the metals is nickel-59 with a halflife of nearly 80 000 years.

It is not quite clear if the nickel in the alloys will corrode or not. In this study it has been assumed that it will be oxidized to Ni(II). The solubility of nickelhydroxide is very low < 1 mg/l in water with pH between 10 and 13. This pH is maintained by dissolution of the hydroxides from the concrete. It is furthermore assumed that organic complexing agents in the groundwater may increase the solubility of nickel to 30 mg/l.

If the concrete and the buffer mass are not destroyed, the leachrate of nickel is very low. It will take about 20 million years for all the nickel to escape under these circumstances.

The barriers may be degraded by several mechanisms. The clay might be destroyed by the hydroxyl ions from the concrete and

the concrete will be degraded as its hydroxyl ions escape. The clay could be destroyed in 100 000 years if very high and unlikely reaction rates are assumed.

Under the very unlikely circumstances that the concrete and the clay/quartz mixture becomes very permeable to water flow due to degradation , the maximum leach rate for nickel will be $2 \cdot 10^{-6}$ parts per year.

KBS Technical Report No 102

METHOD FOR BENDING IRRADIATED FUEL RODS

Torsten Olsson, ASEA-ATOM, 1978-03-29

S U M M A R Y

ASEA-ATOM has performed a study including experiments which verifies that it is possible to bend, or rather to wind up irradiated fuel rods to a plane spiral with an outer diameter of about 300 mm.

The fuel rods are inserted into stainless steel containers, each with room for two rods. In the container there is also a band of spring material along the side that will be subjected to tensions in the winding-up process. The container is sealed by welding and then winded up to form a plane spiral.

Chapter 2-4 describe how the fuel rods can be packed and winded up in an industrial scale. Chapters 5-7 describe performed experiments which verify the proposed method.

KBS Technical Report No 103

SOME ASPECTS ON COLLOIDS AS A MEANS FOR TRANSPORTING NUCLIDES

Ivars Neretnieks
Royal Institute of Technology, Stockholm, 1978-08-08

Summary

The diffusivity of a macromolecule through a compacted clay layer was measured. The results indicate that molecules and thus colloids of this size - M = 24 000 - will not diffuse through the clay barrier in any appreciable amount.

Another set of experiments indicate that the clay will not be a source of colloids as the montmorillonite particles form a stable gel in groundwater with its rather high Ca^{2+} content.

Transport of radionuclides from the repository by adsorption on colloids coming from the clay will thus be small.

The colloid content of groundwater from Finnsjön was measured. This was less than 1 mg/l, so low a colloid content is of little importance.

Experiments designed to observe the adsorption of colloidal particles from the clay on rock surfaces were inconclusive.

KBS Technical Report No 104

FINITE ELEMENT ANALYSIS OF BENTONITE-FILLED ROCK REPOSITORY

Ove Stephansson, Kenneth Mäki, Tommy Groth,
Per Johansson, Luleå Institute of Technology, Juli 1978

SUMMARY

This study presents the results of a finite element analysis of the stability of a storage hole filled with compacted bentonite and a storage tunnel filled with a bentonite-sand mixture for a final repository of radioactive waste. The bentonite has swelling properties and the swelling pressure causes additional loading on the wall of the storage hole and tunnel. In this analysis we assume a swelling pressure of 10 MPa for the storage hole with the diameter of 1.5 m and 0.5 MPa for the storage tunnel with the width of 3.7 m and the highth of 4 m.

A tunnel with a circular hole in the bottom is a composite structure where a three-dimensional finite element analysis is needed in order to solve the stress and strain distribution in a proper way. However, today we lack three-dimensional programmes to simulate the properties of a jointed rock mass. Therefor we chose to study the problem of a repository for the two cases of a plane strain analysis and an axisymmetrical analysis and with the assumption that the rock has linear elastic properties. The analysis for plane strain reproduce the situation around the tunnel and the axisymmetrical analysis reproduce the situation for the storage hole. This part of the study was performed with the programme ADINA at the Division of Rock Mechanics at the University of Luleå. This way of analyzing the problem helped us to evaluate the results from the two-dimensional finite element analyses (in plane strain) of

a repository in a jointed rock mass.

Results of the FEM-analysis at plane strain show stress concentrations in the roof of the tunnel and below the bottom of the storage hole. While the axisymmetrical analysis indicated a more even and homogeneous stress distribution with minor stress concentrations at the bottom of the hole. The FEM analysis were performed for three different loading conditions, σ_h/σ_v = 0.25, 1.0 and 1.5 where σ_h is the horizontal load and σ_v the vertical load and for the two cases with and without bentonite fill. Results indicate that the most favourable stress distribution is obtained for the case of large horizontal stresses which work against the swelling pressure from the bentonite. Displacements caused by the bentonite swelling is of the order of a couple of millimeters for plane strain analysis and one order of magnitude less for the case of axisymmetrical analysis.

In the FEM-analysis of a repository in a jointed rock mass we studied the effect of swelling of bentonite and the intrusion of bentonite into the joints for the case of a rock mass with induced sets of joints. The induced joints are supposed to have been formed due to blasting and occurs in the vicinity of the tunnel and the hole. The joints reach only about a meter into the rock mass and the horisontal load is one fourth of the vertical load. The loading of the models is performed in seven steps: 1, initial loading; 2, excavation of the tunnel; 3-4, excavation of the hole; 5, filling of bentonite and bentonite-sand mixture; 6, bentonite filling and bentonite intrusion 0.2 m into the joints around the storage hole; 7, bentonite filling and bentonite intrusion 0.5 m into the joints around the storage hole. For each load step we analyzed the stresses and displacements in the solid rock blocks and the joints and the opening up of the joints. The analysis were done with the programme BEFEM at the Department of Mining, Royal Institute of Technology, Stockholm.

The blocky rock mass close to the openings cause a destressing. The swelling of the bentonite and the intrusion of bentonite into joints have a local effect on stresses and displacements.

Blocks situated at the mouth of the storage hole show the largest displacement, approximately 70 mm. This is valid when the hole is filled with swelling bentonite and at 0.5 m intrusion of bentonite into the joints counted from the periphery of the hole. Results of analysis in plane strain show that the joints in the vicinity of the storage hole will open up a maximum of 3 mm when the hole is excavated. This is further increased to 13 mm around the top of the hole when the hole is filled with bentonite. Around the bottom of the hole the deformations are calculated to be 5 mm. However, in the real case the displacements will be less due to the confined conditions around the hole. From this study we conclude that the opening of the joints will not exceed some millimeter in the rock mass around the storage hole and the tunnel.

KBS Technical Report No 105

NEUTRON-INDUCED RADIOACTIVITY IN FUEL ASSEMBLY COMPONENTS

Nils A Kjellbert, AB Atomenergi, 1978-03-30

ABSTRACT

A thorough investigation of the importance of various nuclides in neutron-induced radioactivity from fuel element construction materials has been carried out for both BWR and PWR fuel assemblies. The calculations were performed with the ORIGEN computer code. The investigation was directed towards the final storage of the assembly components and special emphasis was put to the examination of the sources of carbon-14, cobalt-60, nickel-59, nickel-63 and zirconium-93/niobium-93m.

It is demonstrated that the nuclides nickel-59, in Inconel and stainless steel, and zirconium-93/niobium-93m, in Zircaloy, are the ones which constitute the very long term radiotoxic hazard of the irradiated materials.

KBS Technical Report No 106

RADIATION LEVEL AND RADIANT ENERGY IMPARTED TO WATER OUTSIDE OF CANISTERS IN THE FINAL REPOSITORY

Klas Lundgren, ASEA-ATOM, 1978-05-29

Summary

The radiation field outside a capsule of copper containing approximately 500 spent BWR fuel rods has been calculated. An average burnup of 30 000 MWd/tU and a cooling time of 40 years have been assumed. Neutrons from transuranic elements and photons from fission products contribute to the dose-rate. On the surface of the capsule the gamma dose-rate will be of the order of 17 mrem/h and the neutron dose-rate 40 - 95 mrem/h.

Interacting with water, ionizing radiation produces free radicals and oxygen. Energy deposition in water outside the above capsule of copper has been calculated to be $1 \cdot 10^{18}$ and $3 \cdot 10^{19}$ MeV, summed up to 10^4 and 10^6 years, respectively. The maximum theoretical amount of copper that could corrode is 17 g up to 10^4 years. In reality, the amount will certainly be much less. The neutron and gamma energy deposition in water in the event of any water getting inside the capsule, has also been calculated. Deposition rate after a cooling time of 40 years will be approximately $1.5 \cdot 10^{14}$ eV/g H_2O,s and energy deposition, summed up to 10^4 and 10^6 years, will be $2 \cdot 10^{23}$ and $9 \cdot 10^{23}$ eV/g H_2O, respectively. This is true when the water is

not in direct contact with the fuel material. If the reverse is true, there are also contributions from α - and β-radiation.

Energy deposition in water outside a capsule of lead containing waste from reprocessing has also been calculated and the result is $1\ 10^{18}$ and $3 \cdot 10^{19}$ MeV, summed up to 10^4 and 10^6 years, respectively (assumed a cooling time of 40 years before storage). The capsule is assumed to have a cladding of titanium, so that the lead cannot corrode unless the cladding is penetrated. In that case, the maximum theoretical amount of lead that could corrode will be 28 g up to 10^4 years. In reality, the amount will certainly be much less.

KBS Technical Report No 107

LEAD-LINED TITANIUM CANISTER FOR REPROCESSED AND VITRIFIED NUCLEAR FUEL WASTE - EVALUATION FROM THE VIEWPOINT OF CORROSION

The Swedish Corrosion Institute and its reference group
Final report, 1978-05-25

Summary

The Nuclear Fuel Safety Project (KBS) has proposed, to fulfill the requirements of the so-called "Conditional law", that reprocessed and vitrified waste from nuclear reactors would be disposed of by enclosure in titanium canisters with 6 mm thick walls and a 100 mm thick lead lining. The canisters are to be placed in vertical drill-holes in rock, 500 m below ground, and embedded in a buffer of 80 - 90 % sand and 20 - 10 % bentonite.

The Swedish Corrosion Institute has been given the job of evaluating the proposal from the point of view of corrosion and of estimating the life of the canisters under the conditions given. To fulfill this task, the Corrosion Institute has appointed an expert group of 10 Swedish specialists mainly from the fields of corrosion and materials technology.

On estimation of the life of the titanium sheath a general corrosion rate of 0.25 µm/year has been taken as a conservative value, which would lead to a life of at least ten thousand years. Pitting and crevice corrosion have been considered very unlikely at the foreseen temperatures and salt contents. Further the risk of delayed fracture, due to hydrogen up-take, is considered as small but cannot be completely excluded at the present state of knowledge. For this reason the titanium sheath cannot absolutely be guaranteed an appreciable lifetime.

If the titanium sheath were penetrated due to mechanical damage or localized corrosion, the exposed lead could suffer localized attack. The corrosion rate would then be determined by the supply of oxygen from the surrounding buffer to the canister surface. Conservative calculations have shown that perforation of

the 100 mm thick lead lining would take about 4 500 years. In any case the life of the lead lining was estimated to at least a thousand years.

In total a titanium canister with a lead lining was estimated to have a life of at least thousands of years, and probably tens of thousands of years.

The expert group was unanimous in its judgement with the exception of professor Gösta Wranglén, who has delivered a statement of his own.

KBS Technical Report No 108

CRITICALITY IN A SPENT FUEL REPOSITORY IN WET CRYSTALLINE ROCK

Peter Behrenz, Kåre Hannerz, ASEA-ATOM, 1978-05-30

ABSTRACT

The KBS project proposes a method for final disposal of spent fuel as waste in a repository in deep mined cavities in wet crystalline rock. Such a repository would contain large amounts of fissile nuclides. This report considers the risks associated with assembly of these nuclides into critical configurations.

Criticality incidents could in theory involve plutonium as long as that remains in appreciable quantity or uranium 235 after the decay of plutonium 239, when the uranium has an enrichment making criticality in a natural water system possible.

Any criticality incident must be preceded by penetration of the long lived copper canisters surrounding the spent fuel waste, the life of which has been estimated to hundreds of thousands of years.

Hence canister failure at a time where sufficient amounts of plutonium 239 still remain is already an unlikely event. It turns out that in order to create a situation where criticality with this nuclide could take place subsequent to canister failure several other unlikely conditions have to be simultaneously assumed. Thus the overall probability of the occurence of criticality is vanishingly small.

Furthermore, it can be shown that the environmental impact of such a criticality, should it occur, will be almost negligible.

Criticality with uranium 235 involves time perspectives far longer than are otherwise considered in connection with nuclear waste disposal. Basically, it would amount to a new Oklo phenomenon, but one where the critical mass is much greater due to the lower enrichment.

For geometric and other reasons it has to take place within the mined cavities of the repository, i.e. the deposition holes for the canisters or the tunnels above them.

Criticality in the tunnels is an extremely remote possibility that could be totally eliminated by admixture of a few per cent of magnetite to the tunnel filling. The relative increase in environmental impact of the repository due to such an incident would be small.

As with plutonium, criticality in the deposition holes is a very low probability event that must be preceded by among others the partial removal of the filling (buffer material) placed in the hole. Its maximum environmental impact will be negligible.

In fact, the scenarios leading to criticality incidents in the repository are based on so unlikely assumptions that they border on physical impossibility and would hardly merit consideration in an ordinary safety analysis. The sole reason for discussing them at length in this report is the fact that statements are frequently appearing in the literature and public debate claiming that criticality represents a major hazard in geologic disposal of spent fuel.

The conclusion of this report is that, at least as far as the KBS proposal is concerned, such claims can be dismissed and that the total risk potential associated with criticality incidents in the repository can be considered negligible.

KBS Technical Report No 109

LEACHABLE GAP ACTIVITY

Lennart Devell, Rolf Hesböl, AB Atomenergi, October 1978

SUMMARY

In the event of ground water penetration into fuel pins after ultimate disposal in crystalline rock a certain fraction of the volatile fission products iodine-129 and cesium-135 may be leached from the fuel within a shorter time period than the rest of the fuel. This fraction based on gap activities and leach experiments has been estimated to be as follows.

Nuclide	Leachable fraction (%) in the gap BWR-fuel	PWR-fuel
I-129	0.1 - 1	1 - 10
Cs-135	0.1 - 1	1 - 10

KBS Technical Report No 110

IN SITU EXPERIMENTS ON NUCLIDE MIGRATION IN FRACTURED CRYSTALLINE ROCKS

Ove Landström, Carl-Erik Klockars,
Karl-Erik Holmberg, Stefan Westerberg
Studsvik Energiteknik and The Geological Survey of Sweden, July 1978

SUMMARY AND CONCLUSIONS

Nuclide migration in fractured rocks was studied in field experiments in a granite-gneissic outcrop within the Studsvik research centre area. Radionuclides representing long-lived fission products of the elements selenium, technetium, tin, cesium, iodine, neodymiun and strontium were injected into a fracture zone intersecting one of the testholes at a depth of 72 metres below the ground surface. Concentration-time curves of activities arriving at the pumping borehole were measured. The distance between the boreholes was 51 metres and the pumping rate 0.1 litre/s. The mean transit time and the dispersion coefficient for water were calculated to be 57 h and 8.8 m^2/h, respectively. A shorter flow path (22 metres) could be utilized by intermittent sampling in a third borehole.

The radioactivity measurement techniques included gamma-spectrometric borehole logging, continuous activity measurements of pumped water and water sampling followed by laboratory analysis. Different combinations of radionuclides were injected simultaneously and selectively registered by gamma-ray spectrometry using a Ge(Li) detector in the field as well as in the laboratory.

Technetium and iodine travelled as anions with the same

velocity as that of water. Strontium was retarded by a factor of about 6. Neither cesium nor neodymium could be detected. This is in agreement with their high retardation factors, estimated from mass distribution constants, determined in the laboratory. Selenium was partly and tin probably almost completely precipitated when injected into the borehole, which complicated the interpretation. A selenium pulse of short duration arrived simultaneously with the bromine pulse. About 20 days later the beginning of a second breakthrough of selenium and a first breakthrough of tin was observed in the shorter flow path.

In a second series of experiments it was shown that the fracture system surrounding the injection borehole could be effectively grouted with bentonite. Subsequent tracer tests with strontium-85 showed that after more than twelve months all the injected activity still remained in the borehole. It was found that bentonite injection is a very effective way of reducing the radionuclide migration through strongly fractured rock.

KBS Technical Report No 111

NUCLIDE LEVELS IN SPENT LWR FUEL AND IN HIGH LEVEL WASTE FROM THE RECYCLING OF PLUTONIUM IN PWR

Nils Kjellbert, AB Atomenergi, 1978-07-26

ABSTRACT

An investigation of the importance of physical simplification in the ORIGEN code has been made by comparison with the BEGAFIP and CASMO codes. The most significant discrepancy seems to be due to the ORIGEN overestimation of the self-shielding effect on the neutron capture resonance integral of plutonium-240. This leads to an underestimation of the inventories of heavy nuclides with masses 241 through 246. In connection with the investigation, a recalculation of earlier presented inventories in spent fuel has been made, including studies of helium pressure build-up due to alfa decay, and neutron-induced activity in fuel rod construction materials.

A rough estimation of plutonium recycle impact on radionuclide inventories in high-level waste has also been made. It is shown that recycling will have no dramatic effects on the inventories of radiologically significant nuclides.

The study was initiated and sponsored by KBS.

KBS Technical Report No 112

SAFETY ANALYSIS IN THE HANDLING PROCEDURE IN THE ENCAPSULATION OF SPENT FUEL IN COPPER CANISTERS

Erik Nordesjö, ASEA-ATOM, 1978-03-20

Summary

In the encapsulationplant the irradiated fuel will be encapsulated in a copper container before delivery to the final underground waste disposal. In this report the process for encapsulation is described and an analysis of offsite radiological consequences both for yearly normal activity discharges and from unlikely accidents is performed.

From the report it is concluded that the radiological effects will be very small, less than 10^{-4} mrem/year to surrounding people from normal operation. Even at extremely unlikely accidents, the individual doses will be negligably small.

It is further concluded that the transport down to the final underground waste disposal can be performed in a safe manner. Even considering a total failure in the transport device down to the wastedisposal, significant damages on the capsul can be avoided with a special design of the transportshaft.

KBS Technical Report No 113

STUDIES OF CERAMIC MATERIAL FOR ENCAPSULATION OF HIGH-LEVEL WASTE

Lennart Hydén et al, ASEA-ATOM, September 1978

Canisters with high-active nuclear waste that are buried in the ground, e.g. at 500 m depth in crystalline rock, are exposed to water and, for some time, to elevated temperature. In this case, the temperature is maximized to $80^{o}C$. At the time these studies started, the information regarding the chemical composition of the ground water was not complete. Therefore, the leaching tests reported here have been performed in a synthetic ground water somewhat different from the later established "normal" composition.

In order to present the knowledge of "corrosion" on ceramics L Bergström and A Liljestrand compiled a report on stability of minerals at hydrothermal conditions (1) and at Svenska Silikatforskningsinstitutet R Carlsson and L Hermansson made a litterature survey on the chemical stability of ceramics (2). R Carlsson also presented a proposal of candidate materials (3).

Bergström and Liljestrand, as a principle, only studied those minerals which are thermodynamically stable at low pressures, low temperatures. This should exclude i.a. mullite (the high-temperature variety of Al_2SiO_5); cordierite (α-Mg_2 $(Al_4Si_5O_{18})$) as well as the plagioclases (certain Na- and Ca-feldspars). Only SiO_2, Al_2O_3 and TiO_2 should meet that requirement. (KBS presents studies on Al_2O_3 in other reports.)

An investigation of the resistance of glass-ceramics based on diopside and feldspar is reported by Carlsson and Hermansson; in that report the glass-ceramic material showed a corrosion rate equivalent to 10 mm in 10 000 years in boiling 10% NaOH. No leaching studies were performed on this material, because it should take too long to get the samples.

Samples of cordierite could be obtained quicker as the ceramic group at Metallforskningsinstitutet already worked with cordierite. Glass-ceramics seemed to be of great interest and contact was established with Corning Glass Works, USA, certainly the most important in this field. It was agreed that Corning should develop the technique to manufacture a full-scale container and investigate the corrosion properties of such a material. The first part of this work soon had to be abandoned for economic reasons, but the corrosion studies are reported in KBS report no 93. Samples of their selected material were incorporated in the corrosion studies performed at Studsvik.

Experience from the manufacture of ceramic bodies this size (Ø 0,5 m; L 3 m) had to be considered also. Ifö Electric Högspänning AB had such experience and their porcelain LD was selected for corrosion investigations.

The corrosion testing was performed by AB Atomenergi, Studsvik. The quality of the water to be used was given from the information available for KBS on ground water chemistry in crystalline rock (4). The composition of the water led to the interesting observation that the first period of disolution was followed throughout the testing by a weight gain from precipitation of new minerals on the surface of the samples. So, the gravimetric methods could not be used for measuring the corrosion rate. The material from Corning, which contains 3,5% Li_2O, could be studied by measuring the content of Li in the water; there is no Li in the synthetized ground water. For the measurements to be valid, the leaching must not be selective. At 90^o a loss of material equal to 11 ± 6 μm/year was measured between 40 and 150 days (4). The results from Cornings own measurements say 5-18 μm/year, so the agreement is quite good.

For chemical analysis of the surfaces two electron-spectroscopic methods were selected. Augerelectronspectroscopiœ (AES) is a method to scan the surface with an electron beam (Ø 1 μm), which releases "Augerelectrons" and their energies are analyzed. The depth of information is 5-15 Å (6).

Electron Spectroscopice for Chemical Analysis (ESCA) was also used. This technique is based upon exciting the electrons with X-rays. The smallest area that can be studied is 1 mm^2 and the depth of information is 10-30 Å (7).

Both methods can be combined with an etching (by argon-ions) of the surface, and if the etching has done no harm to the remaining material, the analysis can be repeated meaningfully. However, the information from these analysis is precluded by the precipitation of new minerals and no information on the corrosion rate of the base materials was gained.

The same phenomenon also disturbed the SEM-studies of the corroded surfaces (5).

In case the studies of glass-ceramics for nuclear waste canisters will be continued, the author wants a few things not to be forgotten: the development of a technique to make a full-size canister looked promising; the material to be used for sealing the lid to the container showed a corrosion resistance similar to that of the base material; an interesting method to measure the corrosion rate without too long testing times is proposed in the report from Corning Glass Works (KBS report 93).

KBS Technical Report No 114

γ-RADIOLYSIS OF ORGANIC COMPOUNDS AND α-RADIOLYSIS OF WATER

Hilbert Christensen, Studsvik Energiteknik AB

1978-09-07

SUMMARY

This KBS-report is a collection of five technical reports. Various radiolytic problems in connection with the disposal of high-active waste are dealt with. The English titles of the five reports are:
1) Radiolysis of fulvic acids; 2) Radiolysis of organic compounds in bentonite; 3) α-radiolysis of water during the disposal of fuel without reprocessing; 4) Radiolysis of water during the disposal of unreprocessed spent fuel. Oxidation by hydrogen peroxide; 5) Formation and decomposition of hydrogen peroxide by α-radiolysis. These reports (in Swedish) are attached as Appendices A-E.

KBS Technical Report No 115

ACCELERATED DISSOLUTION OF URANIUM FROM α-ACTIVE UO_2

Gösta Nilsson, Studsvik Energiteknik AB, 1978-04-27

Abstract

The loss of uranium atoms from the surface of UO_2 due to its alpha activity has been estimated. The loss may be due to sputtering, evaporation from "thermal spikes", chemical etching of the alpha tracks and blister formation and exfoliation of the surface.

The effect of sputtering was estimated by a calculation of the number and energy of the different knock-ons in the collision cascade. Only the primary knock-ons were found to have sufficient energy to be ejected. The emission rate of uranium due to sputtering was found to be $1.3 \times 10^{-14} \times Ci$; g U cm^{-2} $year^{-1}$, where Ci is the alpha activity in curie ton^{-1}. The loss of uranium due to evaporation, blistering and exfoliation is probably less than that caused by sputtering. When UO_2 is in contact with water chemical etching of the alpha tracks by the oxidazing radicals formed by the radiolysis of water may give a very large contribution to the uranium loss. This loss is impossible to calculate and should be found experimentally.

KBS Technical Report No 116

LEACHING OF Al_2O_3 UNDER SIMULATED REPOSITORY CONDITIONS

Britt-Marie Svensson, Lennart Dahl
Studsvik Energiteknik AB, 1978-06-02

SUMMARY

Al_2O_3 material has been leached at 90°C in:

1) simulated ground water at pH 8.5
2) embedded in bentonite + silica sand saturated with the same water
3) in simulated ground water at pH 6 and pH 10.

Leaching periods varied from 30 days to 300 days.

We observed slight weight increments in all cases from deposits on samples from the environment. These mask weight losses from Al_2O_3 that may have occurred.

KBS Technical Report No 117

LEACHING OF Al_2O_3 IN TWICE-DISTILLED WATER

Britt-Marie Svensson, Göran Blomqvist
Studsvik Energiteknik AB, 1978-05-29

SUMMARY

Al_2O_3 material has been leached at 90°C in pure double-distilled water and in double-distilled water buffered to pH 9.3 with sodium carbonate. The corrosion rate of Al_2O_3 has been calculated after determination of the Al-content of the leaching solution by atomic absorption spectrometry.

The corrosion rate in the pure water was $10 \cdot 10^{-6}$ mm·year^{-1} all through the investigation, 184 days hitherto.

The corrosion rate in the solution at pH 9.3 was $\sim 230 \cdot 10^{-6}$ mm·year^{-1} during the first days and then steadily decreasing to $10\text{-}12 \cdot 10^{-6}$ mm·year^{-1} after 120 days and stabilising there during the leaching period 120 to 184 days.

The leached material has been sent to other laboratories for further investigations.

KBS Technical Report No 118

SLUTRAPPORT Al_2O_3 CANISTER

The Swedish Corrosion Institute and its reference group

The report is expected to be printed in the autumn of 1979

KBS Technical Report No 119

FINAL STORAGE OF ACTIVATED STEEL COMPONENTS IN CONCRETE

Lars Rombén, Kyösti Tuutti
The Swedish Cement and Concrete Research Institute,
1978-07-14

SUMMARY

A part of the medium-level nuclear waste problem is the disposal of metal parts which have been activated during use in nuclear reactors. Some alloys contain Ni-59 with a half -life or 75000 years. A proposed method for final disposal consists in storing the waste in rock tunnels at a depth of 300-500 m inside containers which are surrounded by a buffer consisting of compacted quartz-bentonite mixture. This paper deals with the possibilities to use containers made of concrete and designed as cubic boxes enclosing the waste material cast and incorporated into cement mortar matrix. Special regard is given to the containment problem in relation to the mechanical and chemical processes that are operating. A calculation of the rate of release of Ni-59 through the walls of a container has given a figure of about 10^{-5} kg Ni per year for undeteriorated container walls.

KBS Technical Report No 120

SOME NOTES IN CONNECTION WITH THE KBS STUDIES OF FINAL DISPOSAL OF SPENT FUEL

Ivars Neretnieks,
Royal Institute of Technology, September 1978

Abstract

This report contains four short notes on problems briefly studied in connection with the KBS investigations on final disposal of spent unreprocessed fuel.

The first note discusses the possibility for thermal convection flow of water in the buffer material. It is concluded that thermal convection will not have a noticable effect on the transport of water dissolved species.

The second note reports some experimental data on the sorption of strontium and cesium on granite.

The third note discusses the time required to accumulate a critical mass of uranium in the buffer material. It is concluded that it would take millions of years even if very unlikely series of events are postulated.

The fourth note discusses the consequences of hydrogen production in the repository. It is concluded that no grave consequences are expected if escaping hydrogen opens a channel in the clay and in the rock.