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PERFORMANCE ANALYSIS FOR WASTE REPOSITORIES IN THE NORDIC COUNTRIES

Report for Project AFA-1.2

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Informationsservice, Risø, 1997

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Summary

The Nordic Nuclear Safety Research (NKS) project (AFA-1) focused on safety in the final disposal of long-lived low and medium level radioactive waste and its subproject (AFA-1.2), where this report has been produced, is dealing with the performance analysis of the engineered barrier system (near-field) of the repositories for low- and medium level wastes. The topic intentionally excludes the discussion of the characteristics of the geological host medium. Therefore a more generic discussion of the features of performance analysis is possible independent of the fact that different host media are considered in the Nordic countries.

The different waste management systems existing and planned in the Nordic countries are shortly described in the report. In the report main emphasis is paid on the general discussion of methodologies developed and employed for performance analyses of waste repositories. Some of the phenomena and interactions relevant for a generic type of repository are discussed as well. Among the different approaches for the development of scenarios for safety and performance analyses one particular method - the Rock Engineering System (RES) - was chosen to be demonstratively tested in a brainstorming session, where the possible interactions and their safety significance were discussed employing a simplified and generic Nordic repository system as the reference system. As an overall impression, the AFA-project group concludes that the use of the RES approach is very easy to learn even during a short discussion session. The use of different ways to indicate the safety significance of various interactions in a graphical user interface increases the clarity. Within the project a simple software application was developed employing a generally available spreadsheet programme. The developed tool allows an easy opportunity to link the cell specific comments readily available for the 'reader' of the obtained results.

A short review of the performance analyses carried out in the Nordic countries for actual projects concerning repositories for low and medium-level waste is also included in the report.

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Sammandrag

Inom NKS (Nordisk kärnsäkerhetsforskning) genomförs bl a ett projekt om slutförvaring av långlivat låg- och medelaktivt avfall (AFA-1). Föreliggande rapport har producerats inom ramen för ett underprojekt om funktionsanalys för närområden till slutförvar för långlivat låg- och medelaktivt avfall (AFA-1.2). Med närområde avses här själva förvaret med dess allra närmaste omgivning. Geosfären har inte inkluderats eftersom förutsättningarna här varierar från land till land.

I början av rapporten görs en kort beskrivning av existerande, planerade eller tänkbara lösningar på hantering av radioaktivt avfall i de nordiska länderna. Tyngdpunkten i rapporten ligger på en allmän diskussion om utvecklade och tillämpade metoder för funktionsanalys avseende slutförvar för radioaktivt avfall. Något om fenomen och interaktioner av betydelse för en allmän typ av slutförvar diskuteras också. Bland de olika metoderna vid utveckling av scenarier för säkerhets- och funktionsanalys valdes en metod - the Rock Engineering System (RES) - som i samband med ett möte demonstrerades och tillämpades på ett förenklat och allmänt nordiskt slutförvarskoncept. Mötesdeltagarna konstaterade att metoden är mycket lätt att sätta sig in i på kort tid. Genom att markera olika interaktioners betydelse grafiskt i en matris ges en klar och strukturerad presentation av informationen. Inom ramen för projektet utvecklades också ett verktyg i form av en datafil, som kan användas som hjälp när en ny RES-matris ska tas fram.

Rapporten avslutas med en kort översikt om funktionsanalyser, som genomförts i Norden för konkreta projekt beträffande slutförvar för låg- och medelaktivt avfall.

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1 Introduction

The fifth four-year NKS program for the period 1994-1997 includes one waste project, two reactor safety projects and a total of six projects about radioecology, emergency preparedness and information [1.1]. The waste project (AFA-1) is focused on safety in the final disposal of long-lived low and medium level radioactive waste. It is divided into three sub-projects dealing with [1.2]:

- Waste characterisation (AFA-1.1),
- Performance analysis (AFA-1.2),
- Environmental impact statement (AFA-1.3).

The present report is related to the subproject on performance analysis.

A detailed description and documentation of the technical concept and comprehensive characterisation of the site is made before proceeding to final site selection and construction of the repository. In this report phenomena and modelling related to the external geological characteristics of the site, the groundwater movements and the geochemical conditions are not discussed. Instead, the report will concentrate on the evaluation of the performance of the engineered barrier system (near-field) of the repositories. This approach enables a more generic discussion of the topic independent of the fact that different host media are considered in the Nordic countries.

Performance analysis and assessment

In simple terms, *performance assessment* is an analysis to predict the performance of a system or a subsystem, followed by comparison of the results of such analysis with appropriate standards or criteria. Without the latter phase concerning the judgement on the meeting of the safety requirements one could simply use the term *performance analysis*. In this report the repository with its engineered safety features (release barriers) and the immediate near-field comprise the suite of subsystems considered in the limited performance analysis. A comprehensive performance assessment (analysis) becomes a *safety* assessment (analysis) when the system under consideration is the overall waste disposal system and the performance measure is radiological impact or some global measure of impact on safety. Thus performance analysis can be used to describe the analysis of systems at a variety of levels while safety analysis is related to the overall system analysis.

The repository design, in many cases, includes additional engineered barriers like thick concrete vaults and backfilling for example by sand or dense clay or a combination thereof. These will enhance the protection against groundwater flow through the disposal volume and will minimise and delay radionuclide transport from the waste to the geological environment.

The long-term safety of a repository for low- and intermediate level nuclear wastes can be systematically assessed through predictive modelling of gradual failure of the engineered barriers, i.e., the waste form, waste package and the backfill (if any) and the potential subsequent transport to man's environment of radionuclides by circulating groundwater. A complete safety assessment will include the evaluation of the performance of both the engineered safety barriers in the near-field (including the waste form) and the natural barriers. In this report the relevant physical/chemical phenomena and methodologies for the analysis of the performance of the repository itself and the immediate interface to the surrounding are discussed. A very important phase of the modelling is the choice of adequate representation of the real system by defining a conceptual model as the basis for mathematical model(s) taking into account the phenomena, factors and interactions that are important for the description of the behaviour of the system to be analysed. The performance assessment includes the analysis of consequences within the engineered barriers of the repository and defines the source term for subsequent phases of the complete chain of analyses. The evaluation of off-site radiological consequences to humans requires in addition the analysis of geosphere and biosphere transport of radionuclides and finally the evaluation of external and internal radiation exposures to humans via different pathways. These latter aspects have not been discussed in this study.

During the licensing procedure, the results of the safety analysis and their inherent uncertainties will be checked and assessed by the regulatory authorities. In most cases the safety criteria or requirements are expressed for the total system in the form of releases to the biosphere, radiation exposure or health risk. However, the national safety authorities may choose to define separate safety criteria or performance targets for the engineered safety barriers. These type of criteria may concern the limits on the concentration of radionuclides in wastes or limits on the total activity of radionuclides to be disposed of at a given facility. Furthermore separate performance standards can be stated for example for the mechanical, physical and chemical stability of waste forms and waste packages. However, it is generally considered useful to keep sufficient flexibility in the requirements for subsystems as less optimal behaviour of one barrier can be compensated by more stringent requirement for the others. The overall system performance, expressed for example in maximum individual dose rate, is decisive in meeting the regulatory requirements. However, due account should be given to redundancy and diversity aspects of the multibarrier approach as well as for the various uncertainties involved.

References

1.1 NKS, Projektplaner för programmet 1994-1997.NKS(94)7 rev, 1994.

1.2 BRODÉN, K., CARUGATI, S., PALMU, J., WALDERHAUG, T. and SØRLIE, A., Nordic Nuclear Safety Research within the Waste Field. Topseal´96, Stockholm June 9-12, 1996. International Topical Meeting Demonstrating the Practical Achievements of Nuclear Waste Management and Disposal. Transactions Volume II, pp. 121 - 124.

2 Waste disposal systems in the Nordic countries

The practice and planning for disposal of radioactive waste is widely different in the Nordic countries. This is mainly because only two of the countries have nuclear power plants but also because different approaches to the need for early disposal have been taken in the various countries.

2.1 Denmark

Radioactive waste from nuclear research and from the use of radioisotopes by other laboratories, industry and hospitals are collected, treated and stored at Risø National Laboratory. Spent fuel from the research reactor at Risø is returned to USA.

Most of the stored waste is low- or intermediate level but some is α-contaminated. Risø plans to store the waste for some 30 to 50 years so that disposal first will take place in connection with complete decommissioning of the nuclear facilities at the research centre.

Some preliminary design of disposal systems for low- and intermediate level waste (LLW and ILW) in Denmark have been made [2.1]. The facilities considered can all be regarded as examples of advanced types of near-surface burial systems. No special studies have been aimed at the small volume of long-lived ILW stored at Risø.

The short-lived LLW and ILW contains primarily β- and γ- emitters with halflives less than 30 years (mainly ^{60}Co, ^{137}Cs and ^{90}Sr). In approximately 300 years or less the radioactivity of the short-lived LLW and ILW will have decayed over a thousand times to harmless level. This type of waste can be disposed of using relatively uncomplicated methods in near-surface facilities.

Long-lived α-emitting elements must not be present in more than trace quantities in such repositories. The contents of ^{238}Pu and ^{239}Pu plus ^{240}Pu in bituminized evaporator concentrates, which is one of the major waste types at Risø, have been measured to be about 2.5 MBq and 1 MBq per ton of conditioned waste, respectively.

In France, waste accepted for disposal in near-surface facilities must not contain more than an average of 0.01 curies (370 MBq) of α-emitters per ton of waste [2.2]. In Denmark the nuclear authorities have not yet specified limits for the content of long-lived α-emitters in short-lived LLW and ILW which are going to be disposed of in a near-surface facility.

Figure 2.1 shows a concept inspired by the French way of disposal of LLW and ILW at "Le centre de la Manche and Le centre de l'Aube". The main features of this concept is what we today think will be the solution for disposal of Danish LLW and ILW albeit before any performance- or safety- analyses have been done.

The following assumptions have been made:

- The facility is constructed in unconsolidated geological formations such as clay or sand.
- The facility is expected to be situated above ground water level.

Sections, in form of square boxes, with internal dimensions 11 x 11 x 5 m (5 m deep) and with walls and bottom consisting of about 1 m thick reinforced ordinary concrete, are constructed in a 9 m deep excavation in a suitable sand or clay formation.

The standard units containing LLW and ILW (210 l steel drums) are transported to the disposal centre by car. The units are placed in position in the section by a travelling crane installed in a light construction which provides some weather protection and which can be moved to a new position when the section is filled. The units are on top of each other in 5 layers with 324 units in each. After a square box is filled up the sealing of the crevices between the units is made with a suitable injection concrete. A 1 m concrete lid is then cast and finally 2 m soil is distributed on top of the construction. Possibly the concrete lid could be covered by a layer impervious to water (bitumen as an example).

The top of the construction will be made in a form permitting easy drainage of percolating rain water. The water should be able to find its way further down into the ground without any risk of water accumulating around the construction.

One square box contains about 1620 standard units (210 l steel drums containing 100 l waste) corresponding to 340 m^3 conditioned waste. It is estimated that the total total volume needed for disposal of LLW and ILW in Denmark will be about 10 000 m^3. This includes waste from dismantling of the nuclear research facilities in Denmark. To satisfy this need it will be necessary to construct 30 square boxes corresponding to 3 - 4 times the area shown in Figure 2.1.

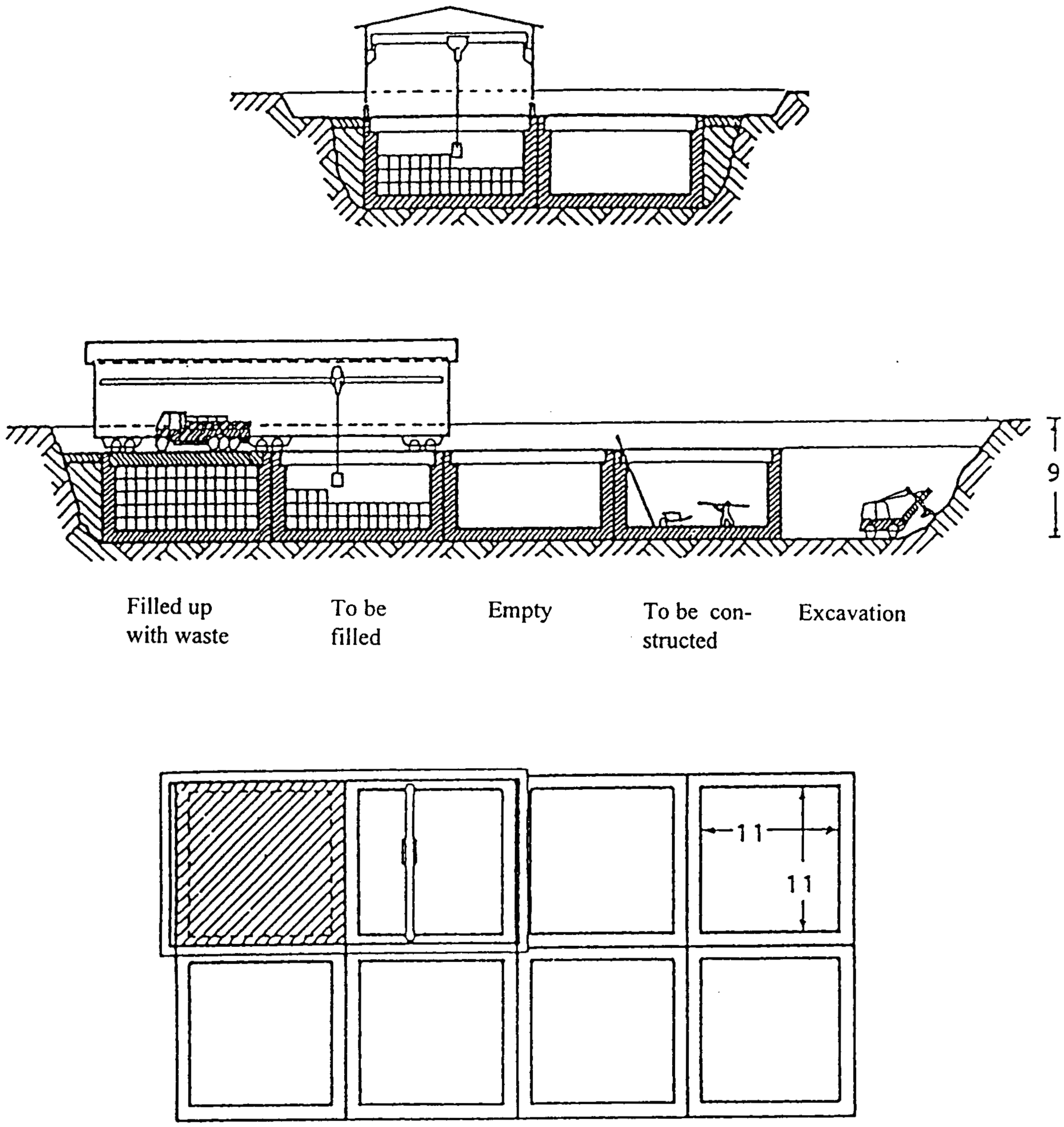

Figure 2.1
A concept for final disposal of LLW and ILW in Denmark in a near-surface facility [2.1].

2.2 Finland

About 30 % of all electricity produced in Finland is generated by nuclear power. Four reactors, with a total capacity of 2 310 MW_e(net), are currently in operation. At Loviisa, there are two 445 MW_e PWR units and at Olkiluoto two 710 MW_e BWR units.

The owner of the two VVER-440 reactors at Loviisa, Imatran Voima Oy (IVO), made initially contractual arrangements for the entire fuel cycle service from the former USSR, including return of spent fuel. However, at the end of 1994 the Finnish Parliament issued an amendment of the Nuclear Energy Act prohibiting practically all export and import of nuclear wastes, including spent fuel from NPPs.

The owner of the Olkiluoto NPP, Teollisuuden Voima Oy (TVO), has opted for storing and, later on, disposing of its spent fuel in a deep geological repository in Finland. A consequence of the amendment of the Nuclear Energy Act, inter alia, was that IVO has to implement the same principles and time schedule as TVO in the management of spent fuel after 1996, when the returning of spent fuel to Russia is no more allowed. Major part of the preparatory work and implementation will be done in a joint company Posiva Oy which was established in October 1995 and has started operating in the beginning of 1996. The total amount of spent fuel to be disposed of is now estimated to consist of 1 700 tU of BWR fuel from Olkiluoto and 740 tU of PWR fuel from Loviisa. The mission of the new company is restricted to the disposal of spent fuel.

Conditioning and storage of low- and intermediate-level wastes from reactor operation, as well as waste from their decommissioning, will take place at the NPP sites. These wastes will be disposed of in underground repositories in the bedrock of the power plant sites.

Figure 2.2 depicts the present arrangements for the management of radioactive wastes in Finland. Presently - according to the amended Nuclear Energy Act - the management of all radioactive waste relies on domestic solution [2.3]. Most of the wastes arise from the operation and decommissioning of the four power reactors in Finland. Limited amount of radioactive waste arising from research activities as well from hospitals and industry are presently stored in an interim storage facility operated by the Finnish Centre for Radiation and Nuclear safety (STUK) at the island of Santahamina in Helsinki. According to the present plans these wastes will be further stored and ultimately disposed of in the VLJ Repository at Olkiluoto.

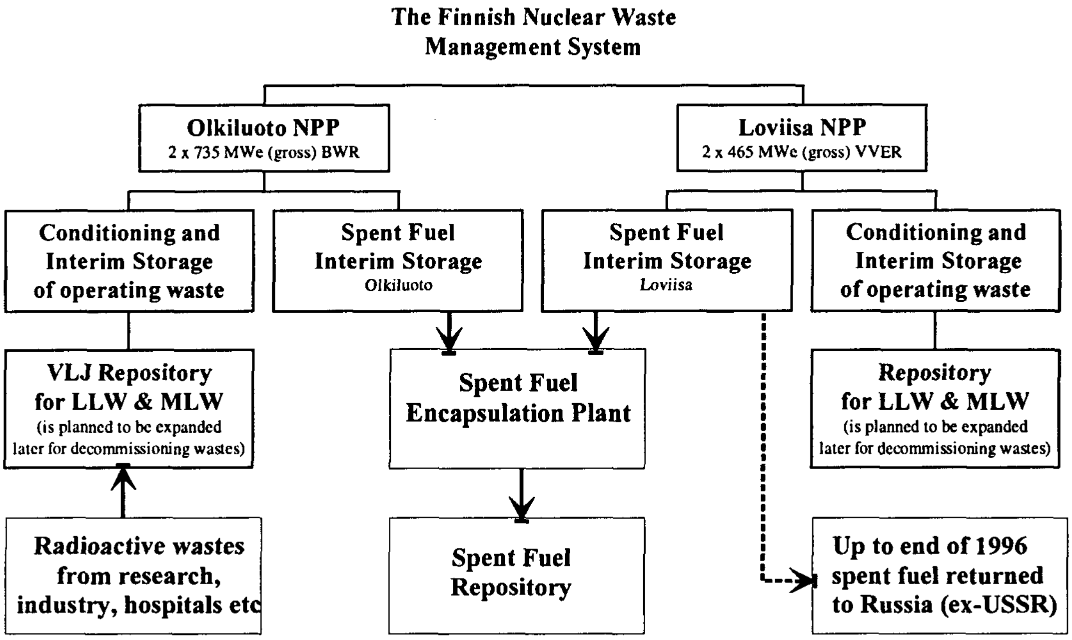

Figure 2.2
Diagram on overall nuclear waste management system in Finland.

The construction of the repository for the low- and intermediate wastes from the operation of the Olkiluoto plant began in 1988 and the operation of the repository commenced in May 1992. The construction of the repository at the Loviisa plant was started in February 1993 and its operation is planned to be started in 1999.

The designs of the Olkiluoto and Loviisa repositories are somewhat different mainly because of the local geological conditions. At Olkiluoto the host rock massif favours vertical silo-type caverns, whereas at Loviisa horizontal tunnels are more suitable.

At the Olkiluoto site the bedrock consists of an intact tonalite massif surrounded by micagneiss. Groundwater of the site is of fresh or brackish type with no great variations in salinity.

In the Olkiluoto repository two separate silos were constructed at the depth of 60...100 m, one for bituminised intermediate level wastes, the other for dry maintenance waste. The diameter of the silos is 24 m and the height 34 m. The silo for maintenance waste is a shotcreted rock silo. The silo for bituminised waste consists of a thick-walled concrete silo inside the rock silo. No backfilling will be used inside the concrete silo. The empty space between the concrete silo and the rock will be filled with crushed rock. In both silos the waste drums will be emplaced within concrete boxes each containing 16 drums.

The bedrock of the Loviisa site on the island of Hästholmen consists of rapakivi granite. The groundwater on the island contains two zones of different salinity. The boundary between the upper, lens-like zone of fresh groundwater and the lower zone of saline, stagnant groundwater lies in a fracture zone varying between

-60 and -140 m. The repository is being constructed at the level of approximately -110 m below the gently dipping fracture zone.

The plans for the decommissioning of the Finnish NPP's are updated every five years. The latest plans were published at the end of 1993 and it included the disposal plans and preliminary safety assessments for the decommissioning waste of the Olkiluoto nuclear power plant (two BWR units and the spent fuel interim store). According to the new plan the existing VLJ repository for low and medium level operating waste will be extended with three new silos at the depth 60 - 100 m (Figure 2.3). Besides dismantling waste also used fuel boxes, control rods and other activated metal components accumulated during the operation of the reactors will be disposed of in the repository. Activated waste will be packed in concrete boxes which are emplaced in a concrete silo constructed inside the rock silo. Contaminated waste will be emplaced in two rock silos and very low level contaminated waste will be placed in the excavation tunnel and the auxiliary rooms of the repository. The disposal rooms for decommissioning waste will be excavated in the 2040's and therepository will be sealed around the year 2055.

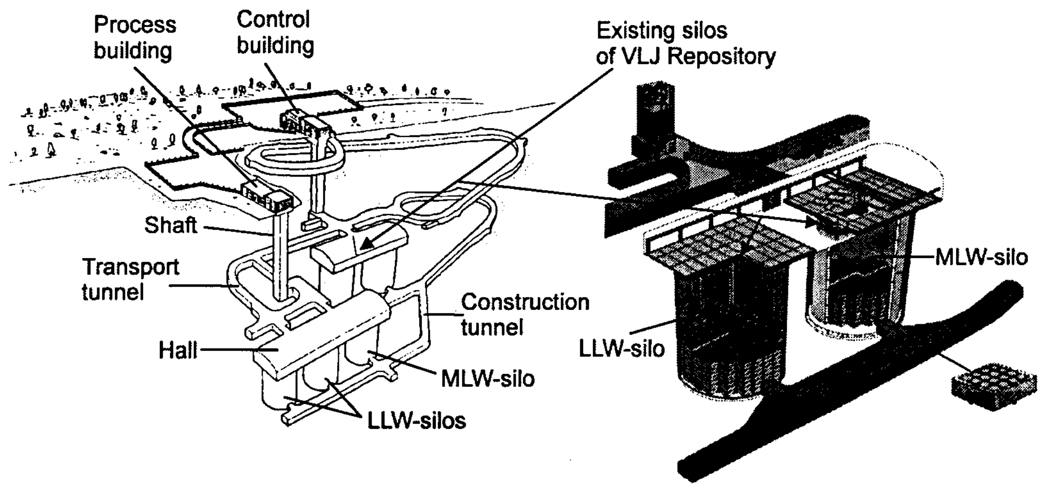

Figure 2.3
The VLJ Repository for low and medium level operational waste disposal at Olkiluoto and the planned extension of the repository for disposal of decommissioning wastes from the facilities at Olkiluoto. A detailed view of the existing silos for LLW and MLW is shown as well.

2.3 Iceland

Iceland has only accumulated small amounts of radioactive waste and no specific waste disposal plans have so far been developed. International solutions, i.e. the waste is disposed off in another country against payment, have been discussed, but the growing international consensus of each country managing its own waste, makes this solution not so feasible. An alternative involving a repository in Iceland must therefore also be studied and compared with an international solution. The obvious disposal facility for Iceland is in the form of an engineered repository in hard rock, and the possibility of finding a suitable disposal site of this kind has been studied [2.4]. The geology of Iceland differs considerable from the other Nordic countries, and a near field function analysis of a repository will have to be based on a different set of physical parameters which is important to identify.

The potential radioactive waste in Iceland consist mainly of three nuclides, ^{60}Co, ^{137}Cs and ^{241}Am, from industrial, medical or domestic use [2.5]. It is anticipated that the future growth in waste will also be concentrated on those nuclides. Therefore, two time periods of waste isolation from the environment have been considered, 10.000 years for ^{241}Am, and 1000 years for ^{60}Co and ^{137}Cs.

Iceland is situated on the Mid-Atlantic Ridge or on the boundary between the North-American and Eurasian plates, and is one of the most active volcanic and seismic regions in the world. A search for a suitable disposal site must first of all take this into account, but other factors are also of importance, such as the thermal gradient and ground water flow.

Volcanism takes place in Iceland in the main neo-volcanic zone located along the center of the country from SW to NE. An additional zone is in the Snæfellsnes peninsula, see Figure 2.1. Bordering the zones is a rock group consisting of intercalating lavas and palagonite ridges, with an age from 0.78 - 3.1 million years, and still further from the zone is tertiary rock with an age of 3.1 - 15 million years. A volcanic eruption in the two latter regions is considered virtually impossible on the time span for a waste repository (appr. 10.000 years).

The seismic activity is most abundant within the accreting plate boundary, where earthquakes occur frequently, but of relativly low magnitude (less than 5 on the Richter scale). On the other hand, earthquakes related to a fracture zone that enters the north-east part of the country (Tjörnes fracture zone) and the South Icelandic Seismic Zone, which runs east-west through the southern parts of Iceland, are less common, but can reach a magnitude of 7 or larger.

Outside the neo-volcanic zone the thermal gradient ranges roughly from 60-140°C/km at one km depth, compared to 25-30°C/km in the other Nordic countries. The thermal gradient decreases laterally from the neo-volcanic zone, but there are several examples of high thermal gradient regions also outside the neo-volcanic zone. The temperature is one of the main parameters in a function analysis of a waste repository, and it is important to note that the temperature of a repository in Iceland can be quite different from the other Nordic countries.

The ground water flow in Iceland rates among the highest in Europe. Unfortunately it is also very high in the north-western and eastern parts of the country, the tertiary rock regions. These otherwise stable areas have high rock permeability and are therefore not favourable as sites for a radioactive waste repository.

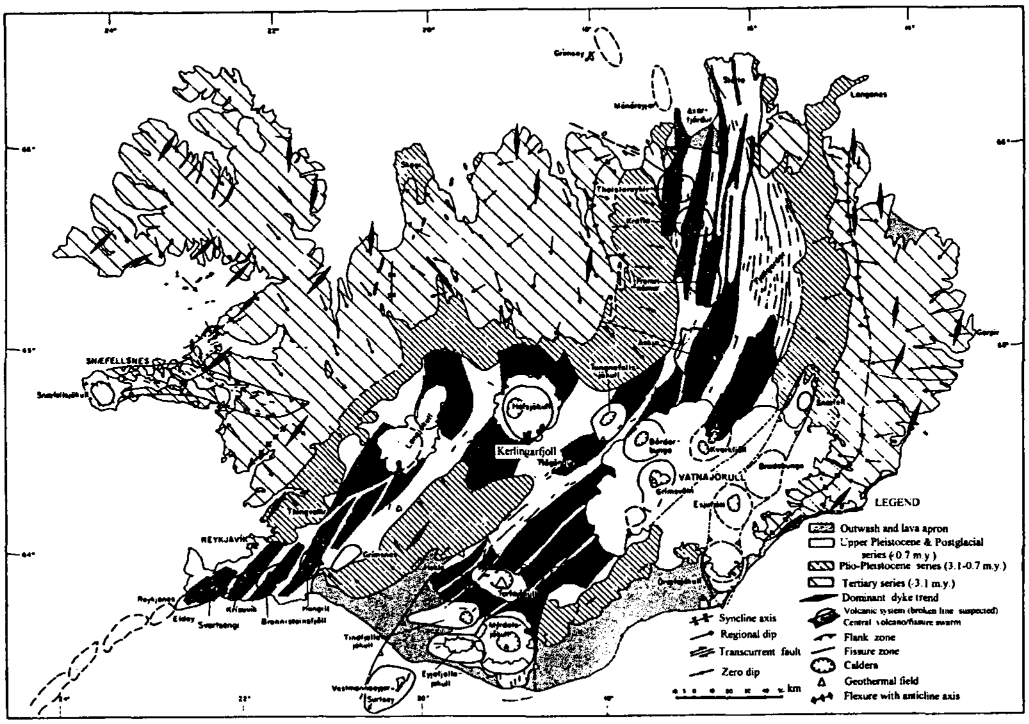

Figure 2.4
Main geological division of Iceland (from ref. [2.6]).

When all of above are taken into account, only two areas are left in Iceland which might be suitable candidates for hosting a radioactive waste repository, a palagonite rock formation in the north-east and some large intrusions in the south-east. The former concist of zeolitized palagonite tuff (Icelandic: moberg), hydrothermally altered with low permeability. A favourable characteristic of zeolites are their ability to absorb large ions such as cesium and strontium. The latter are intrusions in south-east of the country which have cooled down at considerable depth, are fairly homogenous with large lateral continuity and some are also quite extensive in a three dimensional sense.

2.4 Norway

According to plans a combined storage and disposal facility for low- and intermediate level waste will be established in the **Himdalen** in Aurskog-Høland municipality in Norway. The facility is planned to be built in hard rock as a near surface rock cavern facility with 50 metres of rock covering located in a small hill and will be accessible through a tunnel that declines slightly from the facility to the tunnel entrance. As inflow of water is unavoidable in such an formation this will give the facility self draining capacity through a drainage system. According to the Parliament decision plutonium bearing waste containers will be placed in the storage part waiting for future decision on how this waste shall be disposed of. The short lived waste that can be disposed of will be placed in concrete structures (sarcophagi) with a waterproof cover. The drainage systems and the self draining capacity will ensure that the caverns and the sarcophagi will be kept in a dry state.

According to the latest plans building of the facility will start in the spring 1997 and be finished in the spring 1998.The facility will be in operation up to the year 2030. In the year 2030 based on the knowledge and experience gained during the operational phase, a decision will be made whether the storage part containing plutonium bearing waste should be transformed into a repository or this waste should be retrieved. The repository will then be closed but be submitted to institutional control for a period of 300 - 500 years.

2.4.1 The concept and design of the storage and disposal facility

In the current plans the storage and disposal facility consists of the following installations:

- Four rock caverns, three for disposal of waste and one for storage of plutonium bearing waste.
- An entrance tunnel 138 metres long.
- A building inside the entrance tunnel containing service facilities.
- A facility for visitors placed inside the entrance tunnel.
- Tunnel entrance and parking area.

Figure 2.5 and Figure 2.6 show a layout of the facility. Entrance to the four rock caverns is through the common entrance tunnel [2.7]. The caverns are installed at right angles to the entrance tunnel. The outermost cavern will be used as a storage area. The tunnel will be built to admit heavy motor-lorries. Plans exist for construction of a common turning area for these lorries instead of four separate areas as can be seen in Figure 2.5. Plans also exist for combining the service and visitor facilities.

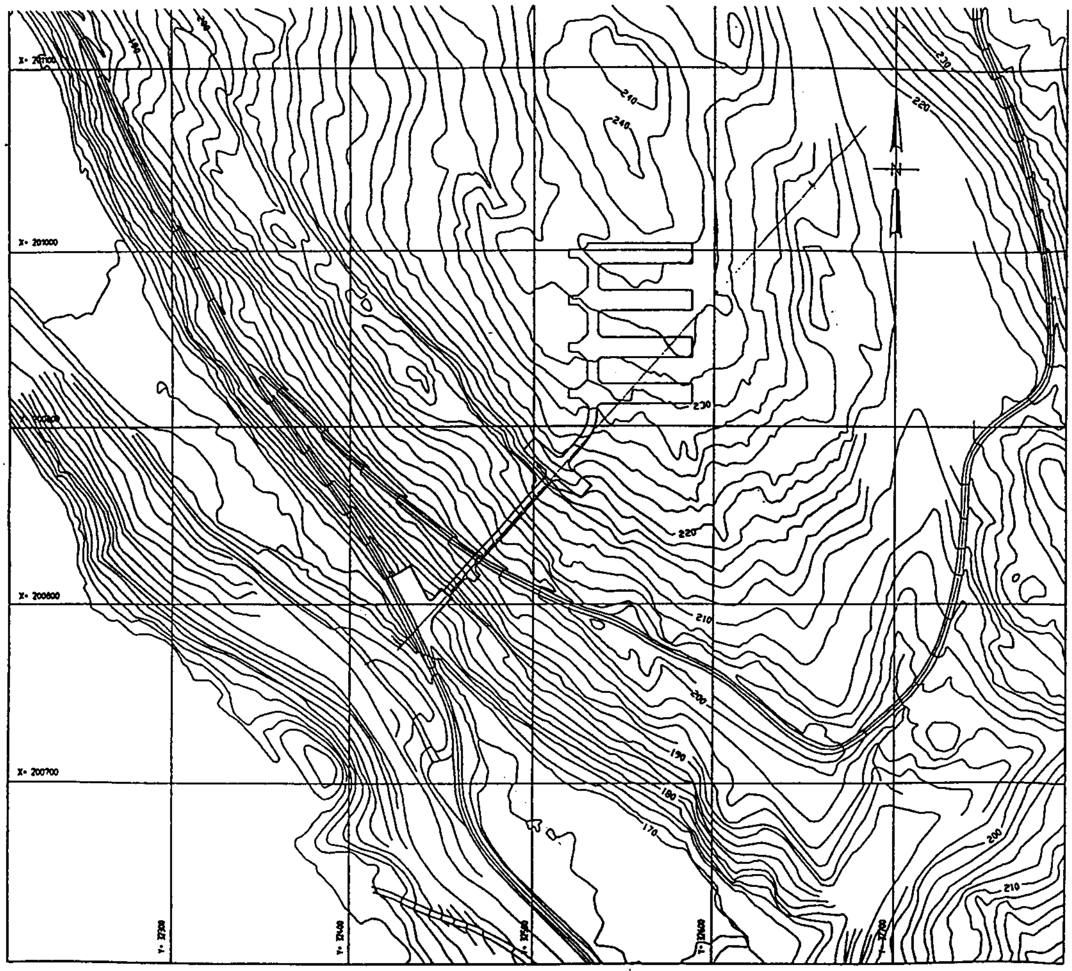

Figure 2.5
The Himdalen storage and repository.

The halls will have a concrete floor while the entrance tunnel will have a gravel floor. Both the caverns and the entrance tunnel will descend in a proportion of 1:50 down to the tunnel entrance.

Inside each cavern there will be built four separate concrete bunks connected two by two. In this way the structure will better able to withstand earth quakes without cracking compared to one long structure. Each separate bunk will have a capacity of 576 or 672 barrel equivalents giving a capacity of 2 496 barrels equivalents per cavern and 7 488 barrel equivalent in the deposit part and 2 496 barrel equivalents in the storage part. Each of the bunks will be filled up with waste containers. In the repository part each layer of waste containers will be surrounded by concrete and given a layer of 15 cm concrete before the next layer is loaded onto this layer of concrete. When a bunk is filled to capacity a ceiling will be mounted with a water tight sealing. The ceiling and seal will be constructed to withstand rock falling from the cavern ceiling. The filling and completion of each part of the sarcophagus is planned to be a continuos operation. Except for the first period of operation when the existing waste stored at the Institute for Energy Technology

(IFE) will be transported to the new repository/storage containers will be intermediately stored at IFE until a sufficient number is present for filling a bunk. Loading of waste containers from motor-lorries into the bunks will be performed by cranes with railings mounted on the walls of the bunks.

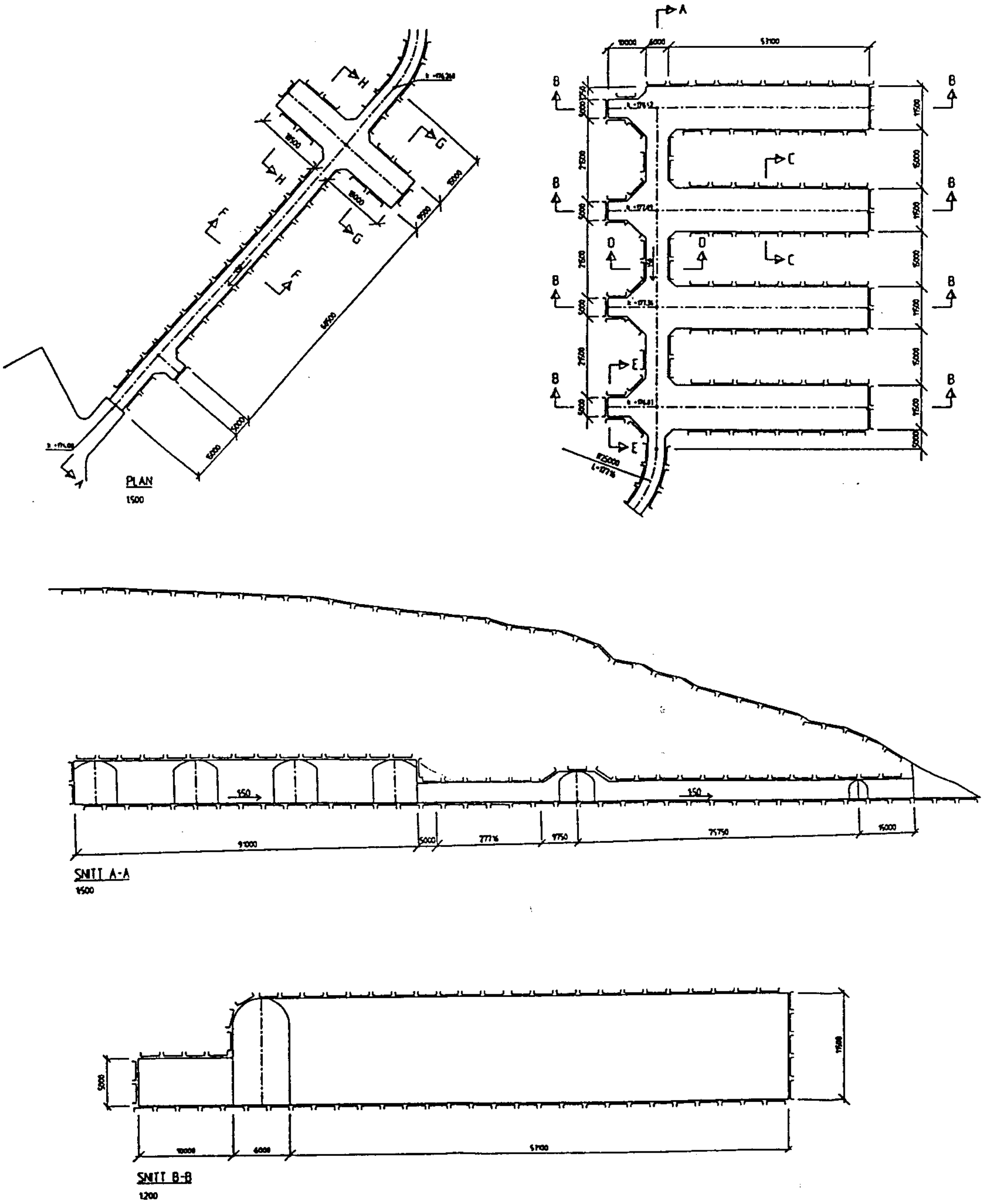

Figure 2.6
Layout of the combined storage and disposal facility (KLDRA)-at Himdalen.

Two drainage systems are to be installed. One system will collect water flowing into the caverns and drain it out of the entrance tunnel by self drainage. The second drainage system will collect water leaking form the concrete structures if this should happen. Due to the self draining capacity of the installation water will drain out through the drainage systems or the floors of the caverns and the tunnel if the drainage system should be blocked. Two separate drainage sumps will be installed in the service area, one for each drainage systems. Monitoring of the drainage water will continue for the operational period. Plans for monitoring of the drainage water for radionuclides will be described in the safety assessment for the operation of the facility. During the period of institutional control of 300 - 500 years the water drained out of the repository will be monitored for its radionuclides content.

2.4.2 Waste volumes, waste types and nuclide inventory

A review of the waste volumes and waste types are summarised below. A more detailed description is given in Ref. [2.5]. The reported nuclide inventory is based on a report to the Directorate of Public Construction and Property given 12. Feb. 1996 [2.8]. The amount of waste already stored at the Kjeller site and anticipated until the year 2030 is given below.

Buried at IFE, Kjeller	1 000	barrels/cases
Stored at IFE, Kjeller	2 000	barrels/cases
Anticipated 1995 - 2030	4 000	barrels/cases
Anticipated from decommissioning of reactors and other facilities	2 000	barrels/cases
Sum	9 000	barrels/cases

These figures include the third stage of the recent decommissioning of the Uranium Reprocessing Pilot Plant. Taking this into account the new anticipated number of barrels/cases from decommissioning of the reactors and other facilities is increased from an earlier estimate of 800 barrels/cases up to 2000 barrels/cases [2.5].

The total number of barrels/cases that will be deposited and stored is estimated to 9 000. The new combined storage and depository is therefore constructed to contain 10 000 barrels/cases.

The radioactive waste is categorised according to:

- Physical form (solid-, liquid- or gaseous waste).
- Activity level and radiation types.
- Exposure level.
- Nuclide content and half lives.
- Chemical composition.

The treatment and conditioning of the waste and the arrangements inside the waste containers depend to a large degree on the physical form of the waste classified according to:

- Solid waste.
- Metal waste.
- Semifloating waste.
- Liquid waste.

The total nuclide inventory planned to be disposed or stored in the KLDRA-Himdalen has been estimated in the following way [2.5]:

- Calculation of the nuclide inventory in the ground repository at IFE, Kjeller.
- Calculation of the nuclide inventory in waste stored above ground up to year 1992 at IFE, Kjeller.
- Estimates of the nuclides and activity levels in waste from operation of research reactors and other nuclear activities from year 1992 to year 2030.
- Estimates of the nuclides and activity levels in waste from decommissioning of research reactors and other nuclear installations before year 2030.
- Estimates of the nuclides and activity levels in waste from other sources between 1992 and 2030.

Since waste containing plutonium shall be stored and not deposited estimates of the total inventory of plutonium have been given special attention.

2.5 Sweden

About 50 % of all electricity produced in Sweden is generated by nuclear power. Figure 2.7 shows a flow diagram over the management of radioactive waste in Sweden [2.9, 2.10]. Sweden has 12 fission power reactors: 4 at Ringhals, 2 at Barsebäck, 3 at Oskarshamn and 3 at Forsmark. The spent fuel from these reactors is sent to the water pool intermediate storage facility CLAB at Oskarshamn. The spent fuel will finally after encapsulation be disposed of in a deep repository, SFL, that will be built somewhere in Sweden.

The medium-level waste and part of the low-level waste from the nuclear power plants are treated and conditioned at the sites. The produced packages are sent to the repository SFR at Forsmark. Part of the low-level combustible waste from the nuclear power plants is sent to Studsvik for treatment and part of the low-level waste is disposed of at shallow land burial sites. The nuclear power plants at Ringhals, Oskarshamn and Forsmark have shallow land burial sites.

At Studsvik waste from many different producers are treated. Packages with treated waste from Swedish nuclear power plants, from the facilities at Studsvik and from hospitals, universities and industry in Sweden are stored at Studsvik. An interim store in rock is used for medium level waste and an interim storage building is used for low-level waste. The packages with comparatively short-lived nuclides are successively sent to the repository SFR while the packages with long-lived nuclides will be sent to SFL.

Ashes produced at incineration of combustible low-level waste from foreign nuclear power plants and combustible low-level waste from fuel element factories are sent back. Ingots from melted contaminated scrap of foreign origin are also sent back if the material cannot be reused.

Very low-level waste from decommissioning of the research reactor R1 in Stockholm has been disposed of at a shallow land burial site at Studsvik.

All the four nuclear power stations and Studsvik are situated on the coast. Therefore sea transportation is the main route in Sweden. A special ship called M/S Sigyn is used for the transport. Special vehicles are used for waste transport between intermediate storage facilities and the ship.

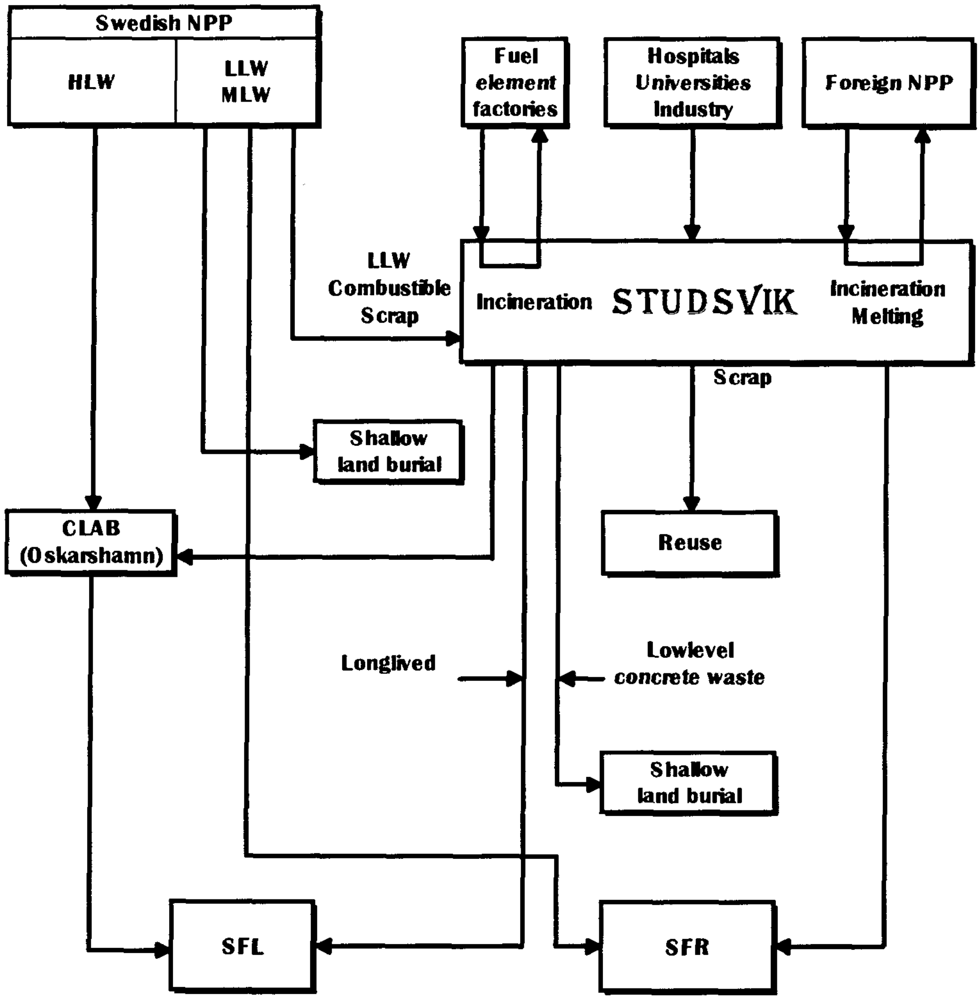

Figure 2.7
Flow diagram over the management of radioactive waste in Sweden

The shallow geological repository, SFR, for short-lived waste from reactor operation, nuclear institutions and industry has been in operation since 1988. It is placed in granite at a depth of about 50 m from the sea bottom, just outside the Baltic shore line, near the Forsmark reactor site. SFR-1 which is now in operation, consists of the following compartments:

- Silo, a shaft type compartment for intermediate-level waste in concrete
- BMA, a tunnel type compartment for intermediate-level waste, mostly in concrete
- BLA, a tunnel type compartment for solid low-level waste (scrap and trash)
- BTF, a tunnel type compartment for dewatered ion exchange resin (not considered in this study)

The total waste volume in SFR-1 is 90 000 m^3.

The repository SFL is planned to be situated at about 500 m depth in crystalline bedrock. Encapsulated spent fuel will be disposed of in SFL 2 and other long-lived waste will be disposed of in SFL 3-5. A previously planned repository for vitrified waste from reprocessing, SFL 1, has been omitted. SFL 3-5 is planned to be situated about one kilometre away from SFL 2. SFL 2 and SFL 3-5 will be reached through the same centrally located shafts.

The SFR 3-5 repository comprises of three areas that will be used for different categories of waste:

- SFL 3 for waste from Studsvik, the central interim storage for spent fuel, CLAB and the encapsulation plant, EP
- SFL 4 for decommissioning waste
- SFL 5 for reactor components.

Not all of the waste intended for disposal in SFL 3-5 falls into the category of long-lived waste. In fact, only the waste that comes for Studsvik, the core components and the reactor internals are long-lived waste. Operational waste and later decommissioning waste form CLAB and EP could in principle be disposed of in the final repository for reactor waste, SFR. However, SFL 3-5 is intended to receive all low-level and medium-level waste that arises in the post-closure period of SFR. That will include for example filter masses consisting of ion exchange resins which would also have been suitable for disposal in SFR.

Svensk Kärnbränslehantering AB, SKB (the Swedish Nuclear Fuel and Waste Management Company), which is jointly owned by the nuclear utilities in Sweden, has been commissioned by them to plan, construct, own and operate systems and facilities for the management of spent fuel and radioactive waste. OKG AB and Vattenfall AB have been assigned by SKB to take charge of operation and maintenance of CLAB and SFR, respectively.

AB SVAFO, which also is jointly owned by the nuclear utilities in Sweden, is responsible for the management of the waste originated form the early research activities at Studsvik. SVAFO owns facilities for waste treatment and intermediate storage at Studsvik. The operation is carried out by Studsvik RadWaste AB as contractor.

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3 Performance analysis methodologies

3.1 Role and scope of performance and safety analyses

Performance and safety analyses are required in various phases of a project to develop and construct a facility for disposal of radioactive wastes [3.1]. In the initial phase, general strategic studies aim at determining the major options for the management and disposal of different types of wastes. In that phase the analyses are quite generic in nature and only few data are likely to be available. Similarly the methodology to be relied upon can be quite simple.

In the next phase the disposal and repository options are identified and analysed in more detail to determine their feasibility for a particular purpose. The type of the pertinent facilities and how big potential hazards are involved determine the role and scope of analyses required. In case of very low level wastes or wastes that can be exempted from regulatory control it is not usually necessary to employ sophisticated sets of modelling tools. For other wastes - including low and medium level wastes and particularly those including significant amount of longer-lived radionuclides - increasingly detailed and concept- and site-specific performance analyses are required later during the repository development project.

In the Nordic countries the regulatory process calls for a preliminary safety analysis report (PSAR) to be prepared in order to obtain the acceptance by the authorities and to receive a permit for construction of the disposal facility. During the construction period more detailed data are obtained on the characteristics of the waste products, packages, engineered safety barriers and on the site-specific features of the geological host medium. These data are employed in the preparation of the final safety analysis report (FSAR) which is required for the application to receive a licence to commission and operate the repository.

Furthermore, the extent and type of performance and safety analyses are dependent on the purpose for which they are carried out and on which organisation is conducting the studies. For example the regulatory body may consider that an independent performance analysis needs to be undertaken to judge the analyses performed by the facility developer. These independent studies may be less comprehensive and concentrate on points where additional information is considered necessary for example to judge the importance of remaining uncertainties and whether these have been adequately covered by the use of conservative assumptions in models and data or by robustness in the facility design.

3.2 Choice of employed methodology

How a performance analysis needs to be undertaken and what type of methodology is required depend on the regulatory requirements, safety or performance indicators (i.e. release rates, individual/collective doses, fluxes to biosphere etc.) considered, the target audience and the timescales necessary to be covered for the considered repository [3.2]. The safety requirements vary from country to country, but within the Nordic countries and in the framework of the NKS safety research programme one is aiming at employing as far as possible common methods, procedures and criteria and to explain remaining differences. The end points of analyses or the performance indicators could be individual or collective doses, or maybe radionuclide fluxes as compared to the flow of natural radionuclides in the environment. The target audience is quite an important factor affecting the type of evaluations needed and especially the way of presenting the results. The same full and technically complex set of analyses needed to obtain the regulatory approval is unlikely to be appropriate and understandable to the political decision makers and general public.

The required complexity of models depends on in which phase of the repository development the analyses are carried out. In addition, there is usually more room for conservatism in analyses for low and medium level repositories as they present lower potential hazards. Consequently, the model validation efforts are not equally important as is the case for high-level wastes and the use of simplified assessment models is usually sufficient with less reliance on comprehensive and detailed research models. In addition, simplifications are also necessary due to the often very complex nature of the low- and intermediate level waste.

3.3 Methodologies for different stages of performance and safety analyses

There are various approaches and techniques for carrying out the analysis of the performance of repository subsystems. Regardless of the detailed methodologies employed it is important to first go carefully through different safety issues that could potentially be important for the performance and behaviour of the engineered safety features as well as the pertinent natural barriers. The second major phase of the performance analysis is then the prediction of consequences in selected scenarios that take into account the identified key safety issues. The major components of a full-scope performance analysis may include the following aspects:

- Development and choice of important scenarios and - depending on the type of methodologies applied - also the evaluation of event probabilities unless they are not conservatively assumed to take place with a probability of one.
- Development of conceptual models for the subsystems to be analysed together with the definition of interaction modes taken into account
- Formulation of the conceptual models in the form of mathematical models for the phenomena accounted for in the performance evaluation
- Analysis of the performance and behaviour of subsystems concerned as well as consequences brought about by the chosen scenarios by numerically solving the equations of the mathematical models developed
- Evaluation of model and data uncertainties and sensitivity of the results on the assumptions made and the variability of parameters describing the characteristics of the technical and natural barriers.
- Confidence building by making comparisons between modelling results and available compatible experimental results from laboratory and field studies.

The term "**conceptual model**" is used in two different, although related, senses:

– the simplified geometrical structure of geological features or arrangement of engineered barrier components assumed in calculations,

– the physical or chemical description of a process, sometimes including its mathematical formulation; cf. 3.3.2

3.3.1 Scenario development

The performance analysis requires as a starting point a number of assumed courses of events or scenarios by which one wishes to analyse the performance of the considered subsystems in a broad spectrum of different conditions. The compilation of the scenarios can be accomplished in a number of different ways. The scenario development methods range from judgemental analyses to systematic approaches [3.1]. Regardless of the sophistication level of the models applied there is no absolutely rigorous and objective procedure to assure scenario completeness and consequently strong reliance must be placed on human judgement.

For relatively simple systems the scenario compilation process can be based on quite simple judgements by a team of experts. In the case of more complicated systems involving many mutual interactions between different phenomena and components more sophisticated and formalised methods have been developed especially for the case of long-lived high level waste. For already operating repositories for low and medium level wastes in Sweden (SFR) and in Finland (VLJ Repository) the scenario analyses were employing expert judgements based on comprehensive research to gain a profound understanding of the safety importance of different factors. In more recent scenario analyses -for example in Canada for a near-surface low-level radioactive waste disposal facility [3.3] an extensive search for important safety issues was carried out using the methods and previous experience of scenario analyses for high-level waste disposal.

A comparison between different formalised methods developed for scenario analyses as well as the their benefits and drawbacks has been presented in [3.4]. The safety and performance analyses for any kind of radioactive waste repository involves the consideration of broad spectrum of relevant Features, Events and Processes, FEPs, that could, directly or indirectly, influence the release and transport of radionuclides within the repository and subsequent migration and transport in geosphere and biosphere. The stages generally included in a scenario development are [3.2] are:

- Identification and classification of all phenomena relevant to the performance of the repository and site,
- Screening of phenomena according to well-defined criteria,
- Identification and grouping of scenarios relevant to the performance of the repository and site and screening of the safety significance of the scenarios
- Specification of scenarios for consequence analysis.

In addition to the identified FEPs, other safety related factors may also be important to the acceptability of the performance of subsystems or the whole disposal system [3.3]. The following examples can be mentioned: deviations of the real facility from the reference design evaluated in the performance analysis, limitations of methods and modelling used in the performance analysis (e.g. applicability of assumptions and models) and evolution of regulatory requirements.

The most demanding and time-consuming task is the screening of FEPs and joint Swedish SKI/SKB efforts have been devoted to develop alternative ways to define so called *Process System (PS),* which according to the definition in [3.4] is "the organised assembly of all phenomena (FEPs) required for description of barrier performance and radionuclide behaviour in a repository and its environment, and that can be predicted with at least some degree of determinism from a given set of external conditions". Several approaches to create and visualise a PS in a systematic fashion have been compared in [3.4]:

In system analysis of e.g. nuclear power plants the event and fault tree analyses have been extensively applied. In waste disposal performance analysis the fault tree method was found less suitable as it is primarily intended to be employed for cases where the events and processes are well-known and supported by extensive statistical data. The performance analysis of repositories can be supported by comprehensive understanding of the processes and it has been concluded that a so called *reversed event-tree* structure can better be used to visualise the Process System. That approach starts with a top event, e.g. the release of radionuclides from the near-field to the geosphere, and then moves inwards barrier by barrier to the initial source, namely the waste form.

The second way to structure the Process System (PS) is to construct an *Influence Diagram* of the PS where FEPs within the PS are represented by boxes and interactions between FEPs are illustrated by lines between these boxes. The construction of the Basic Influence Diagram for the system to be analysed implies the following actions:

- Definition of the system,
- Selection of FEPs relevant for the defined system,
- Identification of influences between the FEPs.

All these steps need to be well documented and compiled. In addition to the influence diagrams an extensive database with descriptions of all FEPs and interactions between them have to prepared with links from each FEP and influence-arrow to the pertinent detailed description or definition in the database.

The development of Influence Diagrams is an iterative process during which FEPs could be combined or alternatively split into more FEPs. Similarly the influences have to be reviewed to remove negligible influences and thereby also those FEP-boxes having only negligible influences to activate their interactions with other FEPs can be removed as well. The reduced Influence diagrams can then be used as the basis for the formulation of scenarios to be analysed in the consequence analysis. The influence diagram approach has been applied in Sweden e.g. for the performance assessment exercise SITE-94 [3.5] and for the SFL 3-5 repository [3.6]. Based on the reduced Influence Diagram, containing about 900 influences, the Reference Scenario and further a Reference Case was formulated to carry out the subsequent quantitative consequence estimation of releases from the repository. After this first attempt of application the general impression on the capabilities of this methodology seems promising and has produced useful results as an input to the planning of forthcoming studies and investigations of the SFL 3-5 concept. A summary of the prestudy on SFL 3-5 is presented in Chapter 5.4.

The third method for scenario analysis as described in [3.4] is the so called *Rock Engineering System (RES)* approach which is a methodology developed to structure problems in rock engineering to ensure that all aspects of the problem are being covered. The approach is, however, not restricted to rock engineering and can be applied to discover the important characteristics and interactions in any kind of complex problems. In the RES approach one starts with the overall objective and then establishes which variables and interactions between variables comprise the mechanism pathways for all the factors. The basic device used in the

RES approach is the *interaction matrix* in which the main variable or parameters (more generally FEPs) are identified and listed along the leading diagonal of a square matrix. The interactions between the FEPs are presented with clockwise convention in the off-diagonal elements of the matrix (Figure 3.1). For example the element I_{12} describes an interaction where parameter P_1 has an effect on parameter P_2 and similarly the element I_{21} depicts an opposite effect of parameter P_2 on parameter P_1. An important aspect of the interaction matrix is that it is generally not symmetric. In the above example the interactions are not identical. For example in case P_1 is canister and P_2 is porewater within the backfill material, the interaction I_{12} could be corrosion and thereby the chemical composition of porewater is changed with subsequent impacts on a number of other phenomena. On the other hand interaction I_{21} could also be corrosion, but now in such a direction that the chemical composition or characteristics of water within the backfill is decisive on the rate of gas generation by a reaction between the canister material and water. In addition to direct interactions there are possibilities for a multitude of indirect impacts. For example in Figure 3.1 the interaction between parameters P_4 and P_2 could be direct or rather complicated indirect effect through a *pathway* M_{4132} involving three subinteractions I_{41}, I_{13} and I_{32}

Thus the RES methodology is comprised of:

- Statement of the project objective,
- Consideration of the necessary variables for the leading diagonal terms,
- Establishment of all the interactions to fill the matrix,
- Study of individual pathways through the matrix,
- Study of the 'matrix evolution' as all the interactions take place.

P1	Interaction I_{12}	Interaction I_{13}	Interaction I_{14}
Interaction I_{21}	**P2**	Interaction I_{23}	Interaction I_{24}
Interaction I_{31}	Interaction I_{32}	**P3**	Interaction I_{34}
Interaction I_{41}	Interaction I_{42}	Interaction I_{43}	**P4**

Figure 3.1
Principle of the interaction matrix in the RES approach.

The application of the RES approach can be preceded and/or combined by a comprehensive analysis of features, events and processes (FEPs) that are important in the consideration of the (sub)system involved. Previously identified FEPs in the context of the studies for the pertinent (sub)system or international databases on similar system studies can be taken into consideration when the main parameters or phenomena are chosen for the diagonal elements. Similarly the consistency of the direct and indirect impacts - as well as processes taking place within one diagonal element - described by the interaction matrix and the list of relevant FEPs can be checked. By this procedure it is possible to structure and complete the set of FEPs considered to be of key importance to the safety of the system. The RES interaction matrix can describe the whole repository system with all the technical and natural release barriers at the same time or subdivided into several individual interaction matrices for a group of parameters describing a subsystem. For example, within this subproject a restricted demonstrative exercise on the near-field subsystem of an idealised repository system was organised in the connection of one working meeting. This simple application is summarised in Chapter 5.5. In the comparative evaluation of scenario analysis techniques [3.4] the applications of RES approach on the scenario analysis of subsystems included in the spent fuel disposal systems have been described. The RES methodology has also been applied in Finland [3.7] for identifying FEPs for the near-field analysis of copper-steel canister.

After the collection of a comprehensive set of interactions between the main parameters, the next step is the evaluation of the significance of the interactions. For example the following type of significance coding can be applied:

- 'Critical' interaction (4; red)
- Strong interaction (3; orange)
- Medium interaction (2; yellow)
- Weak interaction (1; green)
- No interaction (0; white)

The colour codes are useful in graphical illustration of the strength of the interactions. The numerical values can be employed for estimating the overall significance of given diagonal parameter. The horizontal sum of significance values gives an indication of how much a diagonal variable on that row affects the other variables in relation to the other variables. Similarly the vertical sum of significance values describes how much other variables have an effect on the diagonal variable on that particular column. Usually there are a number of individual interactions within one off-diagonal element. Their safety significance can vary greatly and the indication of the overall significance of a particular interaction element have to be judged.

What has been described above on the contents of the RES approach can be categorised as 'soft' application aiming at defining the scenarios of performance analysis. The consequences of these scenarios are then evaluated with relatively simple models. Alternatively, in the 'hard' application of RES approach a fully-coupled model is established for the system to be analysed involving the explicit equations for all the interactions. Usually, however, it is more useful and illustrative to start the consequence evaluation process with only the interactions that are subjectively judged to be significant. These aspects are closely linked to the topic of the next subchapter.

3.3.2 Conceptual and mathematical model development

For each significantly different class of scenarios there is a need to develop a *conceptual model* that describes the possible structures and behaviour of the analysed system in the desired degree of details. The conceptual model involves the set of hypotheses or assumptions describing the physical and chemical processes that affect the time dependent behaviour of the system and the surrounding other systems together with various characteristics of the system as well as the boundary and initial conditions. The choice of appropriate conceptual model is dependent on the purpose and aims of the pertinent study. For example the estimation of total flux of groundwater through a repository requires a less detailed conceptualisation as compared to the case where a detailed distribution of fluxes among the substructures is needed. Several alternative conceptual models might be developed for the same purpose and hence a critical evaluation of the possible uncertainties related to the choice of conceptual model can be evaluated as well. Some sort of systematic approach has to be employed and anyway all the modelling assumptions have to be carefully documented and justified.

Mathematical models are required as the primary tool of performance analysis. Together with the appropriate system- and site-specific model parameters they present a collection of multidisciplinary scientific understanding of the relevant processes determining the behaviour of the system. Mathematical models translate the assumptions of a conceptual model into the formalism of mathematics - usually a set of coupled algebraic, differential and/or integral equations with appropriate initial and boundary conditions within the domain of the (sub)system to be analysed quantitatively by the model. The defined equations are solved analytically, semianalytically or numerically using a computer code corresponding to the model.

Some sort of simplification of processes or geometries is often required and depending on the aims of the study one can for example omit the transient phase of the processes and restrict oneself to stationary solutions. Usually there is a whole spectrum of models available having a varying degree of details involved. At the most detailed level *research models* are needed and employed to build a sufficient understanding of the relevant phenomena and confidence on the capabilities of the models to describe the processes in a way that is compatible with experimental results.

At the other end of spectrum of models lie the *assessment models* for subsystems, such as near-field, and the whole repository system. These models usually have simplified geometry and otherwise less detailed presentation of the processes and their interactions. In addition, these simplified models often apply very pessimistic scenarios, conceptual & mathematical models and parameter values so that the consequences are likely to be clearly overestimated. In most cases the safety margins are sufficient to allow the use of this type of upper bounds. To get a quantitative view of the safety margins, analyses with realistic or best estimate parameter values and for most probable courses of events are useful. With increasing power of computer hardware simplifications are not so necessary, but increasingly complex phenomena can in principle be accounted for also in the applications of assessment models. Nevertheless simplified modelling procedures are needed to increase the transparency of performance analyses in view of the needs of broader audiences involved in the decision-process involved in nuclear waste management projects.

3.3.3 Analysis of consequences and their uncertainties for key scenarios

The most simple ways of describing the consequences of chosen key scenarios of performance analysis can be accomplished by straightforward scoping and bounding analyses in order to get an idea of the order of magnitude of the risks involved in the management and disposal of wastes in a particular case study considered.

In more advanced calculation of consequences two main categories of models can employed. In the *deterministic* approach each individual calculation scenario or case is analysed with a single set of fixed parameter values and quite sophisticated models can be applied. In this approach a base case can represent either the best estimate or conservative set of parameters. A number of other scenarios spanning the range of interest for model parameter values and alternative conceptual models as well as disturbed evolutions and hypothetical events can then be considered separately. In the deterministic analyses no attempt is made to differentiate the considered scenarios by assigning a certain probability for occurrence.

In *probabilistic* consequence analyses uncertainties are quantified by defining probability density functions for model parameters and these distributions are propagated through the chain of models describing a subsystem or the whole repository system. The final result of consequence analysis is then also expressed in a form of statistical distribution, which thus gives a direct measure of uncertainty. Although probabilistic methods seem to provide a comprehensive spectrum for the description of the phenomena considered, there is a danger that parameter values outside their range of validity are sampled and less transparent understanding of important phenomena and interaction is obtained. Consequently, some sort of combination of the different type of methods is often needed. In the consideration of uncertainties probabilistic methods have been extensively applied to describe the impacts of parameter uncertainties. However, it is important to cover also the uncertainties related to the choice of scenarios and conceptual models.

Another aspect related to uncertainties arises from the fact that performance analyses and related uncertainty considerations are carried out iteratively in the various stages of a repository development project. Consequently certain factors identified in preliminary analyses to bring about major uncertainties to the expected behaviour of the repository system can be overcome or avoided by appropriate modifications to the waste management practices or to the design of the repository concept.

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4 Phenomena and interactions in the near-field of a repository

This Nordic study is focused on the safety in the final disposal of long-lived low and medium level waste. In the preceding chapter general methodological aspects of the analyses of performance of a repository system were discussed. The topic of this chapter is devoted to the consideration of those physical and chemical phenomena - as well as their mutual interactions - that are expected to have decisive roles in the behaviour of the repository. Because of the differences among the Nordic countries concerning the existing or planned facilities for the disposal of the waste types considered in this report, the discussion of the pertinent phenomena and interactions is by necessity generic. However, certain common features among the disposal concepts exist especially as regards the repository near-field.

4.1 Typical repository concepts for low and medium level wastes

A succinct summary of the planned and existing repository concepts for low and medium level waste repositories has been presented in Chapter 2. There are obvious differences as concerns the geological host medium and its form and depth among these disposal systems. The discussion of the (hydro)geological aspects of the far-field is, however, outside the scope of this report. As concerns the topic of this study, i.e. the repository near-field, there are nevertheless quite many common features:

- Similar waste forms; for example activated or contaminated metal waste, some sort of organic material etc,
- Same type of waste conditioning methods (bituminisation, cementation) applied,
- Steel drums or concrete containers are used for waste packages,
- Backfill material (sand, clay or their micture) used to fill the space between waste packages,
- Repository silos, tunnel sections, bunkers may include concrete walls
- Repository region is saturated with water (either below groundwater table or assumed to be saturated in some performance analysis scenarios).

A simplified repository concept containing some of these common features is considered as a topic of the demonstrative applications of the interaction matrices according to the RES approach in Chapter 5.5

4.2 Effects caused by the characteristics of the waste form

Although the basic design philosophy of a waste repository presumes that the waste is either not actively reacting chemically or physically with its surrounding environment or is converted into as stable as possible chemical and physical form, there are a number of less obvious effects. For example, the slow penetration of moisture into the canister can initiate the dissolving or leaching of the waste material and thereby change the chemical characteristics of the pore solution in the solidification matrix (bitumen or concrete). The corrosion of metal pieces could cause gas formation (hydrogen). For metal wastes galvanic reaction might occur between the waste form and engineered barriers. On the other hand the presence of corroding inactive metal in the repository could diminish the solubility of metallic radioisotopes such as nickel. The presence of cementitious materials will normally ensure strongly alkaline pore solutions over long periods. However, for wastes containing organic material, such as cellulose, biological degradation may cause gas formation (carbon dioxide) which reacts with the cement producing less alkaline conditions so that the solubility of waste products and radionuclides is enhanced. Colloids or complexing agents may also be produced resulting in increased migration rates in the geosphere. Inorganic complexing agents are in comparison in most cases less important. The strength of nondesirable interactions and direct contacts can be avoided or suppressed by the presence of engineered barriers.

4.3 Effects originating from the behaviour of engineered barriers

One function of the engineered barriers is to avoid or defer as long as possible the direct contact of groundwater percolated into the repository area with the waste form and the radionuclides contained in it. The chemical conditions within the engineered safety features are decisive in predicting the release of radionuclides from the repository to the surrounding geological host medium. The concentration of many elements and their diffusion rate are restricted by sorption into the barrier materials. In addition, the solubility limitations within the packages or other engineered barriers may limit the concentrations and thereby the release of some radionuclides.

During the gradual degradation of engineered barriers, for example the concrete barriers, the chemical environment within the repository remains unfavourable for the release and subsequent transport of radionuclides as the solubility is suppressed and the retention capacity is enhanced. Furthermore, there are interactions that additionally improve the functioning of the repository. These phenomena include: sorption/coprecipitation of radionuclides with corrosion products and coprecipitation with calcite. Reprecipitation of certain leached components from concrete waste packages or concrete walls may reduce the permeability of internal or external backfill layers to water and gas transport.

Regardless of the many planned positive impacts certain undesirable side effects might be caused during the degradation process of engineered barriers. For example, finally the concrete barriers might be altered by long-term leaching away of cementitious components from the concrete and subsequently the permeability of these barrier structures will increase and the characteristics favourable for suppressing the release and enhancing retention will cease to function. Similar degradation processes may take place also in the backfill materials and hence their sorption and diffusion characteristics may be impaired. Also the chemical characteristics, swelling capacity and plasticity of backfill materials may be affected by chemical reactions with cementitious or waste components. Furthermore the degradation and subsequent removal transfer of the materials in the concrete barriers may cause loosening of the backfill layers and hence changes in hydraulic and transport characteristics.

The iron reinforcements within concrete structures may be corroded and the precipitation of corrosion products cause internal volume expansion and hence impaire mechanical stability and increase permeability.

Further the corrosion of steel drums is a major source of gas formation within the repository and therefore the disposal system has to be designed insensitive to this gas formation and the resulting potential for releases in gaseous form or mechanical failures due to pressure build-up within the engineered structures. The colloid generation and subsequent transport might also be caused by degrading concrete and corroding metal components as well as from backfill materials. It is, however, likely that only minor fractions of colloids could be transported through concrete barriers and/or backfill layers.

4.4 Effects caused by groundwater flow through the near-field and repository structures

The flow of groundwater is important primarily because it provides the most likely mechanism by which humans might eventually get into contact with radionuclides released from a waste disposal facility. Within the near-field of the disposal system the amount of groundwater flowing through the repository determines how much water is available and thereby dictates the release rate of solubility-limited radionuclides. In addition, other internal transfer rates within the repository are also proportional to flow rate of the carrying element, i.e. groundwater. The groundwater flow through engineered barriers, such as concrete walls or backfill layers, is the basic reason for removal of some important chemical components from these barrier materials during their gradual degradation and hence the reason for increased permeability and reduction of sorption capacity.

In the case of repositories of long-lived low and medium level wastes, the heat generation rate is low as compared to the heat transport within the repository and the surrounding host rock and therefore the elevation of temperatures for this reason within the repository can be neglected. Consequently, also the potential of heat generation for enhancing the flow of groundwater is negligible.

4.5 Impact of groundwater composition on behaviour of barriers

After the sealing of the repository the groundwater present in the surrounding geological host medium will ultimately infiltrate and fill the repository structures and cause water saturated conditions. For repository concepts where groundwater table lies below the vault episodes might be caused by extreme weather conditions that temporarily cause water infiltration into the repository and hence these type of scenarios cannot be overlooked. Even in quite dry soil formations the soil pore atmosphere is always of high humidity and some types of waste are hygroscopic. The combination may lead to condensation af water inside the waste units, swelling and eventual release of contaminated solution. Interactions of water and its constituents with the construction and backfill materials as well as with the radionuclides ultimately outleached from waste products are dependent on the initial composition of inflowing groundwater. Besides the water composition important characteristics include the redox properties and pH of the water. Also the salinity of groundwater has to be taken into account although in many cases chloride anions react only to a minor extent with the mineral or components of the engineered barriers. However, the presence of chloride is likely to enhance the corrosion of iron. After infiltration inside the concrete barriers there will be a strong chemical influence on the composition and other characteristics of water. For example concrete porewater has a high pH.

4.6 Estimation of the near-field releases and scenario selection

In the modelling of transport within the repository and through its barriers both advective and diffusive mechanisms have to be considered. The reliance given on the retention capability of the waste form and the waste package or container material is waste type specific. Especially for low level wastes the package is often assumed to be degraded fast and subsequently the radionuclides are assumed to become instantaneously distributed evenly inside the repository vault. The resulting concentration of radionuclides in the water volume inside the repository and in the outgoing waterflow is determined on the basis of amount of water available, the sorption properties of structures and backfill barriers regarded as a homogenous mass.. For certain elements also the solubility limitations have to be accounted for. However, such a simplification is not necessarily conservative because the flow is likely to occur preferentially through a minor part of the structure. The amount of water available for leaching and dissolving radionuclides is dependent on the turnover rate of water through the repository system. Besides normal evolution the estimation of near-field releases involves the consideration of various alternative scenarios where varying number of barriers are assumed to cease to fulfil their planned function. As an extreme case there might be requirements from authorities to consider a case involved with total and simultaneous loss of function of all technical barriers and hence complete reliance on the functioning of the natural barriers.

5 Examples of performance assessments for low and medium level waste repositories in Nordic countries

5.1 Examples from Denmark

Site investigations and performance assessments of deep disposal of spent reactor fuel in a salt dome were carried out by the Danish utilities in 1970's. A preliminary review of geological formations potentially suitable for deep disposal was also done and included in an EU inventory of such areas. No actual preparations for repositories for low and medium level radioactive waste have so far been made, but some generic considerations of hydraulic phenomena which must be taken into account in connection with repositories situated above or below the groundwater level are collected in [5.1].

Laboratory investigations of material properties relevant for disposal of low and medium level waste of the types stored at Risø National Laboratory have been carried out through many years in connection with Nordic or EU financed research projects.

Major topics have been the hygroscopicity of bituminized or cemented evaporator concentrates, volume stability and leaching under various conditions of similar materials, and migration through cementitious barriers including cracks in such barriers. Information relevant for definition of FEP's have been attained.

5.2 Examples from Finland

5.2.1 Safety studies for Olkiluoto Repository

Construction of the VLJ Repository was finalised in summer of 1991. The VLJ Repository is an underground disposal facility for low and medium level operational waste arising at the Olkiluoto nuclear plant. A summary of the main features of this repository has been presented in Chapter 2.2 and in [5.2]. The repository consists of two silos (Figure 2.3) excavated at a depth of 60...100 meters in the bedrock. The silo for low level waste is a shotcreted rock silo. For medium level waste a reinforced concrete silo has been constructed inside the rock silo.

Safety requirements

The Finnish regulations for disposal of low- and medium-level radioactive waste [5.3] include rather stringent requirements on the safety of a repository, as well as detailed guidance for the preparation of the Final Safety Analysis Report (FSAR). From the point of view of the post-closure performance analysis the most important requirements are:

- The ALARA principle (As Low A Reasonably Achivable).
- The maximum allowable expectation value for the annual dose of an individual belonging to the critical group is 0.1 mSv.
- Accidents considered as possible may lead at highest to an annual dose of 5 mSv. At least the following accidents shall be considered: rock movements, a bored well, and human intrusion.
- The increase of concentrations of radioactive substances in the biosphere due to the disposed waste shall remain insignificant everywhere.
- The disposal concept shall be based on multiple engineered and natural release barriers. Engineered barriers must efficiently restrict release of radionuclides for at least 500 years. After 500 years primarily natural barriers shall be sufficient for fulfilling the safety requirements.
- At least the following phenomena shall be considered in the safety analysis: degradation of engineered barriers due to interactions with groundwater and the waste, swelling of the waste, gas generation, potential changes in the groundwater flow and chemistry due to e.g. land uplift and sea level changes, as well as changes in the local hydrological system and the land use.
- Administrative surveillance can be assumed to prevent intrusive human activities at most during 200 years after the sealing of the repository.
- Detailed optimisation analyses are not required provided that it can be shown that a realistic estimate of the collective dose commitment integrated up to 10 000 years is at most of the order of 1 manSv.
- The safety analysis must be performed with reliable models, conservative data, and assumptions.

The Final Safety Analysis Report including the post closure performance analysis and the application for the operation license were submitted to the authorities until May 1991 The post closure performance analysis is comprehensively summarised in [5.2]. The safety analysis is based on detailed site investigations performed before and during the construction of the repository. Also properties of waste products and engineered barriers have been comprehensively studied during more than ten years. The safety analysis includes detailed groundwater flow modelling of the site, evaluation of the performance of engineered barriers, as well as analyses of release and transport of radionuclides in the repository, the geosphere and the biosphere. The aim has been to produce a robust and transparent safety case. Conservative assumptions, data, and deterministic models have been used throughout the analysis.

Scenarios considered in the performance analysis

Several scenarios have been employed to evaluate the performance of the disposal system in a wide range of conditions and disruptive events. The reference scenario provides conservative upper limits for the consequences in the anticipated conditions. In the realistic scenario, less pessimistic data and realistic activity inventories have been used. The disturbed evolution and accident scenarios are "what if" type cases where the consequences of various extreme evolutions and events are evaluated. They include a case where the concrete silo for MLW is assumed to be severely impaired immediately after the sealing of the repository. The ability of natural barriers to restrict release of radionuclides into the biosphere has been evaluated by means of scenarios where the degradation of engineered barriers has been assumed to be much faster than can be expected on the basis of experiments, natural analogues and theoretical studies. Effects of gas generation and consequences of human intrusion have been evaluated, too. Most important features of the scenarios are summarised in Table 5.1.

Performance of engineered barriers and near-field performance analysis

In the reference scenario radionuclides are assumed to be released from the bituminised waste at a constant rate in the course of 500 years, although experiments suggest much lower release rates. The bituminised cation exchanger may have ion exchange capacity even when bituminised. With their maximum ion exchange capacity, the resins could absorb a significant proportion of the calcium in the concrete boxes. In the reference scenario it is conservatively assumed that the concrete boxes will disintegrate to a state similar to that of the crushed rock in 1 000 years. Also the outside of the concrete silo is assumed to disintegrate in a similar way at a rate of 10 cm in 1000 years The degradation rate is a conservative choice based on the experiences of the behaviour of concrete structures in various environments, geochemical modelling indicates a much lower degradation rate for concrete in the repository conditions. The disintegration of concrete is assumed to result in a gradually increasing groundwater flow through the concrete silo. In the reference scenario, it is assumed that 1% of the total flow through the rock silo (10 m^3/year) will go through the concrete silo at the beginning and a half of this amount through the concrete boxes inside the silo. The flow through the wall of the concrete silo is assumed to take place via fractures with no retardation of radionuclides in the concrete. This passage flow through the concrete silo is increased stepwise as follows: to 2% after 1 000 years, 3% after 2 000 years, 5% after 3 000 years, and 10% after 4 000 years. The structures of the concrete silo are assumed to cave in after 5 000 years and all the concrete is assumed to be also chemically depleted after 6 000 years. The groundwater flow through the MLW silo is further increased to 20 m^3/year after 9 000 years and to 30 m^3/year after 12 000 years, as the sealing structures of the repository are assumed to be disintegrated.

Table 5.1
Main features of the scenarios considered in the FSAR for VLJ Repository at Olkiluoto.

MLW-REF	Reference scenario for MLW – release from bitumen within 500 years – collapse of the concrete silo at 5 000 years – all concrete chemically depleted within 6 000 years – deterioration of seals within 12 000 years – groundwater flow through the rock silo: 10--20 ~ 30 m^3/yr
MLW-REAL	Realistic scenario for MLW – realistic activity inventory – release from bitumen within 1 000 years – collapse of the concrete silo at 10 000 years – all concrete chemically depleted within 12 000 years – deterioration of seals within 24 000 years – groundwater flow through the rock silo: 5 ->10->15 m^3/yr
MLW-MIGSA	Sensitivity analysis on far-field migration
MLW-DAM	Concrete silo for MLW damaged immediately after the sealing resulting in that 10% of the total groundwater flow through the rock silo goes through the concrete silo
MLW-EB5	5-fold degradation rate of engineered barriers in the MLW silo – release from bitumen within 100 years collapse of the concrete silo at 1 000 years – all concrete chemically depleted within 1 200 years – deterioration of seals within 2 400 years
MLW-EB10	10-fold degradation rate of engineered barriers in the M LW silo – release from bitumen within 50 years – collapse of the concrete silo at 500 years – all concrete chemically depleted within 600 years – deterioration of seals within 1200 years
MLW-EBT	Total loss of performance of all engineered barriers in the M LW silo at 500 years
LLW-REF	Reference scenario for LLW – engineered barriers have an insignificant role – groundwater flow through the rock silo: 10 m^3/yr
LLW-REAL	Realistic scenario for LLW – realistic activity inventory – engineered barriers have an insignificant role – groundwater flow through the rock silo: 5 m^3/yr
GAS	Release of 10% of the total activity of LLW within 10 years
REF-TOT	MLW-REF + LLW-REF
REAL-TOT	MLW-REAL + LLW-REAL

The assumed chemical depletion of concrete has a strong influence on the release of ^{14}C. In a concrete environment, carbonate released from the waste precipitates as calcite. There will be a sharp increase in the solubility of calcite when pH in the repository drops due to the depletion of concrete. In the reference scenario, ^{14}C is assumed to be released from the MLW silo at a constant rate after the chemical depletion of concrete. The release period is conservatively chosen to be a half of the timespan during which calcite - mainly originating from carbonate of the incoming groundwater--accumulates in the repository. Accordingly, all ^{14}C is assumed to be released from the MLW silo between 6 000 and 9 000 years after the sealing of the repository.

The release of radionuclides from the waste products and their subsequent transport by diffusion and convection through engineered barriers has been analysed by the numerical compartment model REPCOM [5.4]. For crushed rock and depleted concrete the conservative "concrete water" K_d-values taking into account complex forming agents in a concrete environment have been used even in the long-term, when all concrete in the repository is assumed to be chemically depleted. In the case of LLW, the K_d-values have been further decreased by a factor of ten to take into account complex forming agents in the miscellaneous waste.

In the scenario, where the consequences of an early damaging of the concrete silo are studied, 10% of the total groundwater flow through the rock silo is assumed to go through the concrete silo immediately after the sealing of the repository. In the scenarios, where the ability of the natural barriers to restrict the releases is evaluated, engineered barriers are assumed to deteriorate with five- and ten-fold rates. As an ultimate "what if" case, a sudden and total loss of the performance of all engineered barriers is set to take place at 500 years

Gas generation and release

No backfilling is used inside the concrete silo for MLW. It has been planned to provide the opening in the lid of the silo with a gas lock cutting the water pathway. A gas lock might, however, be plugged with degradation products of concrete. Also the estimated low gas generation rate in the MLW silo gives cause for a simpler solution; a big hole filled only with crushed rock could be left in the lid of the concrete silo. A gas pressure difference loading the wall of the concrete silo can hence be ruled out. Waterborne diffusion of radionuclides through the hole, as well as a potential minor turnover of water in the silo due to the hole have been taken into consideration in the release calculations. Microbial decomposition of low-level waste is the most important gas generation process in the repository. The gas transport capability of the rock above the repository has been estimated to exceed manyfold the maximum gas generation rate. The remaining concern associated with gas generation is whether gases can be released through the 10cm thick layer of shotcrete on the rock walls. To ensure a harmless release of gases, a partial removal of the shotcrete lining on the ceiling of the repository cavern would be recommendable. Although the possibility of a significant, gas-induced displacement of contaminated water from the repository is considered very remote, the safety analysis includes a "what if" scenario (GAS) where 10% of the total activity inventory in the LLW silo is assumed to be released into the geosphere within ten years.

Results of the safety analysis

The results of the safety analysis show that significant radiation doses can be caused only if a well is bored in the vicinity of the repository or if the groundwater discharge spot is used for farming. For the reference scenario MLW-REF (cf. Table 5.1) the maximum dose rate via a well assumed to be located at a distance of 150 meters (well 2) is $2 \cdot 10^{-5}$ Sv/year. The dominant nuclides are ^{14}C, ^{239}Pu & ^{240}Pu, ^{129}I, and in the short-term, ^{90}Sr. The dose rate maximum is associated with the chemical depletion of concrete and is not affected by the assumed mechanical damaging of the concrete silo soon after the sealing of the repository in the scenario DAM--in the short term, the dose rate is increased in the DAM scenario. Short-circuiting of the migration path in the geosphere (scenario MIGSA) increases the maximum dose rate by a factor of two compared to the reference scenario, the dominant nuclides are now $^{239/240}Pu$. Even in the hypothetical scenario with a ten-fold degradation rate of engineered barriers (scenario EB10) the maximum dose rate via a well located at the distance of 150 m is no more than $3 \cdot 10^{-4}$ Sv/year. In the realistic scenario the maximum dose rate is only $2 \cdot 10^{-7}$ Sv/year.

Maximum expectation values of the dose rates via the wells are presented in Table 5.2 In these estimates, the only probability taken into account is the likelihood of the wells. After 200 years, the probability of a well at the distance of 50 meters is estimated to be 0.05, whereas the probability of a well at the distance of 150 meters is conservatively chosen to be one. The disturbed evolution and accident scenarios analysed are hypothetical "what if" cases. The probabilities of their occurrence cannot be quantitatively estimated and hence only very conservative upper limits can be given for the dose rates.

Taking into account all possible exposure environments (well 1 + well 2 + sea/lake/lake sediment) the maximum expectation value of the individual dose rate is found out to be $3 \cdot 10^{-5}$ Sv/a in the reference scenario. With realistic values for activity inventories, near-field, and migration data, the corresponding value is $2 \cdot 10^{-7}$ Sv/a. The realistic estimate of the collective dose commitment integrated up to 10 000 years is clearly below one manSv.

In the disturbed evolution and accident scenarios the maximum dose rate via a well assumed to be located at the distance of 50 meters is 1 mSv/year. More realistic activity inventories and data would decrease this value at least by a factor of 20.

Table 5.2
Maximum expectation values of the individual dose rate (Sv/a) (taking into account the likelihood of the wells).

Scenario	Well 1 (at the distance of 50 meters)		Well 2 (at the distance of 150 meters)	
	Time (a)	Expectation value of dose rate (Sv/a)	Time (a)	Expectation value of dose rate (Sv/a)
MLW-REF	$1.5\ 10^{4}$	$.5\ 10^{-6}$	$6.3 \cdot 10^{3}$	$2 \cdot 10^{-5}$
MLW-REAL	$1.3\ 10^{4}$	$5 \cdot 10^{-8}$	$1.4 \cdot 10^{4}$	$2 \cdot 10^{-7}$
MLW-MIGSA	$1.3 \cdot 10^{4}$	$<1 \cdot 10^{-5}$	$1.4 \cdot 10^{4}$	$<4 \cdot 10^{-5}$
MLW-DAM	$2.0 \cdot 10^{2}$	$<1 \cdot 10^{-5}$	$6.3 \cdot 10^{3}$	$<2 \cdot 10^{-5}$
MLW-EB5	$1.4 \cdot 10^{3}$	$<3 \cdot 10^{-5}$	$1.7 \cdot 10^{3}$	$<1 \cdot 10^{-4}$
MLW-EB10	$8.1 \cdot 10^{2}$	$<<5 \cdot 10^{-5}$	$9.0 \cdot 10^{2}$	$<<3 \cdot 10^{-4}$
MLW-EBT	$5.2 \cdot 10^{2}$	$<<5 \cdot 10^{-5}$	$5.5 \cdot 10^{2}$	$<<3 \cdot 10^{-4}$
LLW-REF	$1.0 \cdot 10^{2}$	$5 \cdot 10^{-6}$	$1.0 \cdot 10^{2}$	$1 \cdot 10^{-5}$
LLW-REAL	$10 \cdot 10^{2}$	$1 \cdot 10^{-8}$	$1.0 \cdot 10^{2}$	$1 \cdot 10^{-8}$
GAS			$1.2 \cdot 10^{1}$	$<6 \cdot 10^{-5}$
REF-TOT	$1.5 \cdot 10^{4}$	$5 \cdot 10^{-6}$	$6.3 \cdot 10^{3}$	$2 \cdot 10^{-5}$
REAL-TOT	$1.3 \cdot 10^{4}$	$5 \cdot 10^{-8}$	$1.4 \cdot 10^{4}$	$2 \cdot 10^{-7}$

Note: Unlikely scenarios are marked with <.
Very unlikely (hypothetical "what if" scenarios) are marked with <<

The most important barriers and safety features of the disposal system are 1) the 60 meters thick bedrock crust above the repository which efficiently restricts the amount of water coming into contact with the waste, provides stable conditions for the waste and the engineered barriers, and reduces the probability of harmful interactions with the activities of living beings, 2) the bituminised waste form, 3) the concrete silo, and 4) dispersion and dilution in the geosphere and the biosphere. In the safety analysis, all the principal barriers have been modelled applying conservative assumptions and data. It can hence be concluded that the VLJ repository with good margin fulfils the stringent safety requirements established by the authorities.

The Finnish Centre for Radiation and Nuclear Safety (STUK) has reviewed and accepted the FSAR. On the basis of the review STUK assessed that the repository fulfils well the safety requirements. The post closure safety analysis was considered to be of good quality and hence STUK recommended for the Ministry of Trade and Industry that an operation license could be granted for the repository. STUK also recommended that the total activity inventory is limited to 1000 TBq in the MLW silo and to 10 TBq in the LLW silo; these values are based on the long-term safety analysis and are somewhat higher than the conservative activity inventories used in the safety analysis. Later on during the operation the repository complementary studies have been carried out to consider the possibilities to employ the waste silos of the VLJ Repository to also provide a possibility to dispose of miscellaneous other radioactive wastes that have arisen in various applications in research institutes, hospital and certain industrial applications. Presently STUK takes care of the intermediate storage of these wastes.

STUK considers that it is necessary to carry on certain long-term research on the performance and safety of the repository during the operation phase. Especially, the following subjects have been mentioned: sealing of the repository and the long-term performance of the seals and backfills, gas generation of LLW and release of gases from the repository into the rock, possible rock movements and changes in the water conductivity in the fissure cluster intersecting the MLW silo, activity inventories, and geochemical analysis of the groundwater. If necessary, STUK can require updating of the safety assessment during the operation phase. Before the final sealing of the repository, detailed updated plans for the sealing and an updated safety analysis must be presented.

5.2.2 Preliminary analyses for the safety of disposal of wastes arising from the decommissioning of Olkiluoto plant

The decommissioning plans presented in 1987 included comprehensive safety analyses of the final disposal of the wastes for both the Finnish nuclear power plants. At the end of 1993 an updated safety analysis of disposal of the decommissioning wastes arising from the Olkiluoto NPP in an expansion part of the VLJ repository was presented to the authorities

The safety analysis [5.5] for the disposal of decommissioning wastes from Olkiluoto NPP is based on a detailed technical plan and on the comprehensive safety analysis carried out for the FSAR of the VLJ-Repository. Groundwater flow in the repository and in the rock has been analysed in detail taking into account the new parts of the repository. The results of the safety analysis show that the planned disposal concept provides good protection and isolation for the waste and efficiently hinders releases into the biosphere. The most important barriers and phenomena restricting releases are good corrosion resistance of metals in concrete environment, the low solubility of zirconium and nickel, and the large amount of concrete around the most active components emplaced in thick-walled concrete boxes in the middle of the concrete silo. The maximum dose rate via a well at the groundwater discharge spot (the assumed dilution volume is 1000 m^3/yr and the amount of well water ingested is about 1 m^3/yr per person) is less than one hundredth of the dose rate due to the natural background radiation. The extension of the VLJ Repository does not harm the post-closure safety of the existing disposal rooms.

5.3 Example from Norway

In the process of finding a suitable site for a low- and intermediate level waste repository in Norway performance analyses for the near-fields of possible sites have not been performed specifically. The process of selecting and recommending a site has been based on ecological-, social- and economical considerations. After the decision was made that this new combined storage and repository facility shall be located in Himdalen in Aurskog-Høland municipality, the Norwegian Atomic Energy Act require that licence applications for building of the facility and operating the facility must be submitted.

The Directorate of Public Construction and Property is responsible for the construction and will be the owner of the facility. They have therefore applied for a construction licence and have conducted an investigation program in order to confirm that geological-, hydrological- and other characteristics of the site are satisfactory. Their application for a construction licence is therefore based on a safety assessment study of the site and the environment. This study is carried out according to the Planning and Building Act which require that the impact on environment, natural resources and society shall be described before facilities are established and on the Radiation Protection Act which require that radiation protection for the public shall be maintained both in short- and long-term. The Norwegian Radiation Protection Authority (NRPA) will verify compliance of the construction with the licence requirements and the characteristics of the site before construction can start and before operation licence can be granted.

The Institute for Energy Technology (IFE) will be the operator of the storage/repository and must therefore submit an application for a licence for operating the facility. According to the Atomic Energy Act and the Radiation Protection Act this application must be based on a safety assessment study describing safe transport and handling of radioactive waste and radiation protection for occupational exposed workers and the public during operation. This work is under way. The NRPA will verify compliance with requirements before licence for operation is granted.

Since the facility shall be a combined storage and repository further licences will be required at later stages in order to convert the storage to a repository and for the closure of the facility.

The safety assessment study [5.6] for licensing the construction of the storage/repository includes a description of the near-field and overall performance analysis for this facility. The performance analysis is based on computer calculations of different scenarios for releases of radioactivity from the repository and describes the consequences for the far-field and doses to the public. These calculations include a description of how the barriers will behave and disintegrate in different situations. A separate performance analysis of the near-field has therefore not been made.

5.3.1 Engineered safety barriers

The types and number of barriers preventing or delaying the movement of radionuclides into the surrounding environment in the KLDRA-Himdalen facility can be described as follows:

1. The first barrier surrounding the waste depends on the inner construction of the waste containers. Dependent of the type of waste, the level of activity in the waste and the dose rates measured in contact with the waste the first barrier can be:
   - a steel container
   - a lead container
   - a combined steel and lead container
   - a concrete barrier as described in point 2 below
2. The next barrier will in most cases be a concrete barrier constructed as a shielding layer inside the containers or filling all spaces in an between the waste inside the barrels and containers.
3. For liquid waste absorbed in concrete and other additives the next barrier will be the polyethylene lining inside the 210 litre steel barrels.
4. The steel walls of the outer barrels and containers or the concrete walls of the concrete containers will serve as the next barrier.
5. Waste containers of all types in the repository will be surrounded by concrete in the sarcophagi. This structure including the outer walls of the sarcophagi with the addition of a ceiling with a water tight sealing will perform as a barrier against leakages of radioactivity into the rock caverns.
6. The repository is constructed with self draining system. This design will prevent leakages into other directions than through the draining systems or the access tunnel if the drainage systems should be blocked. Calculations performed in the safety assessment study shows that leakages into and though the surrounding rock formation will be negligible. In addition the drainage systems are equipped with drainage sumps and the facility shall be submitted to institutional control for 300 - 500 years to come.
7. The rock formations surrounding the repository and the safety installations preventing access to the repository are barriers for intrusion into the facility.

Details for closure of the repository in year 2030 have not been decided, but there has been some discussion on back filling of the caverns with rock before closure. It can be foreseen that additional barriers will be established during the process of closure of the repository.

5.3.2 Geology, earthquakes and cavern stability

The facility will be built in a hill consisting of Precambrian gneiss. The structure in which the cavern will be excavated is a part of the old Scandinavian tectonic structure. It is dry and without large defects. The nearest weakness zone is in the bottom of the Himdalen valley, but significant seismic activity has not been observed for the last 100 million years. Based on a return period of 10 000 years, an evaluation of earthquake loads has been performed [5.7]. The conclusions of this evaluation are that it is highly unlikely that the facility will suffer significant damage due to the loads imposed by such an earthquake[5.8].

Dynamic loads enhance the potential for individual rock wedges barely in equilibrium to shake loose from the cavern roof and walls, but it hardly affects the overall stability of the facility. Provided the cavern has sufficient safety margin against instability under static conditions, this conclusion is valid. The static stability will be ensured by a rock bolt support system if necessary. Potentially loose rock wedges will also be removed as a part of the construction work.

5.3.3 Hydrology and water flows

At present the ground water level equals the terrain level except at the top of the hill. After excavation of the cavern, the ground water level will fall as the rock is drained, and after about 15 years the water level will be at the floor of the cavern. The ground water level is calculated to be stabilised after about 55 years. This means that all ground water gradients are directed towards the cavern and no water flow away from the cavern is possible. This makes transport of radionuclides through the rock nearly impossible. The inflow of water is estimated to be between 0.1 and 2 m^3/h for the whole facility [5.9]. Much of this inflow will probably be through a flaw in the entrance tunnel.

There are only two possible release pathways. The main way is by water drained through the drain system which will be much more open than the rock. (That excludes flow through the rock as a way of release.) The second possibility is as gas, which only apply to carbon-14 as carbon dioxide and tritium as water vapour.

5.3.4 Scenario selection and releases of radioactivity

In the impact assessment presented in 1992 [5.10] some release scenarios were described, showing that no one would ever receive doses larger than about 1 μSv/a in the most probable scenario. However, the Waste Management Assessment and Technical Review Programme (WATRP) and NRPA criterias concluded that an evaluation of these scenarios would be necessary and that other scenarios should be identified. The identification of new scenarios was performed according to the OECD/NEA list, and the following scenarios have been identified as the most important to evaluate [5.6]. These scenarios cover the main aspects of the main scenarios from the Environmental Impact Assessment (EIA).

The latest work done by AEA Technology defines four groups of scenarios named W, G, ND and HI. W means release by water from the facility, G denotes release by gas, ND denotes natural disturbances and HI human intruition. The water and gas scenarios have been described briefly above. The water scenario will be discussed briefly below.

Base scenario, diffusive release. W1

In this scenario, representing the base case condition, it is assumed that the repository drainage system continues to function according to its design specification. The space around the sarcophagus is therefore maintained essentially dry, with any incoming water being collected by the drain and discharged into a stream. There is no water flowing through the waste packages, and release can only occur by diffusion of pore water. The pore space surrounding the waste is assumed to be filled with water arising from the concrete grout. It is assumed that the radionuclides are taken into solution under conditions determined by the concrete that will limit their solubility.

Release from Flooded Repository. W3

If the drainage system fails it might happen, though very unlikely, that the cavern could become partially or totally flooded. In this scenario it is assumed that the cavern is totally flooded, but that the main drainage is still along the drainage channel into the local stream. It is further assumed that flow through the waste can occur. A solubility limited source term model is again assumed, with the specific discharge in this case being determined by the rate of water seepage into the cavern and the effective hydrological properties of the cavern and host rock. Sorption processes are taken into account.

5.3.5 Gas generation

Two main sources of gas have been considered:

1. Hydrogen generated from corrosion of appreciable amounts of metals and alloys present under the anaerobic conditions expected to be found in a repository.
2. Methane and carbon dioxide, produced in approximately equal amounts by microbial degradation, mainly of wood, paper and textiles in low level waste. The carbon dioxide would be expected to react with cement forming calcium carbonate.

For low level waste a third process, radiolysis, has been found to generate only very small volumes of gas and this process is not considered further.

In the study of gas generation performed by AEA [5.6] generation rates have been calculated for both container corrosion and microbial degradation under both aerobic and anaerobic conditions. The ease of gas migration though the concrete and the potential for pressure built up has been assessed based on calculated generation rates. Of critical importance is the nature of the waste form, such as volumes and types of reactive and biodegradable waste, the amount of water present and the pH in the near field environment.

Calculations of gas generation are based on the following estimations:

a) The repository will contain approximately $2.6 \cdot 10^4$ kg biodegradable waste.

b) It is assumed that approximately 10 % of the waste volume comprises reactive metal taken to be mild steel. In addition there will be mild steel and stainless steel plates in the waste container walls. The total amount of metals and alloys in the repository is estimated to $1.1 \cdot 10^6$ kg.

Gas generation and releases are calculated for three scenarios [5.6]:

1. Scenario G1: Gas generation and releases in the base case scenario (W1).
2. Scenario G2: Gas generation and releases in the flooded repository scenario (W3).
3. Flammability of released gases, explosion risks.

Scenario G1: Gas generation and releases in the base case scenario (W1)

It is assumed that the repository drainage system functions correctly. The cavern access tunnel provide an open access to the environment and hence maintain conditions inside the caverns at atmospheric pressure. In this scenario it is assumed that the barrels have been penetrated during the emplacement period after the closure by pitting or crevice corrosion. In this situation there is enough oxygen in the caverns that can diffuse through the concrete sarcophagi and into the waste containers.

Microbial decay of biodegradable waste will occur by aerobic mechanisms and in the calculation of gas generation it is assumed that the pH value of the pore water is buffered to a fixed value of 11. Calculation over a 500 years period shows that the gas generation is dominated by production of CO_2 and that the generation rate peaks at 600 m^3/year after 30 years and then drops sharply.

Gases escaping from the sarcophagi can not cause pressurisation of the cavern spaces because they are maintained at atmospheric pressure through the access tunnel and the drainage systems. The sarcophagus itself is designed as an impregnable barrier to water flow and may present a substantial barrier to gas migration. From the peak production of CO_2 at 600 m^3/year the maximum internal pressure of the sarcophagus has been estimated to $1.3 \cdot 10^5$ Pa (1.3 Atmospheres). It is considered that this degree of pressurisation should not lead to serious degradation of the sarcophagus structure and therefore is not a significant safety issue. The calculations take no account of the effect of CO_2 reacting with the concrete which reduces the amount of gas being released but also tends to block some of the pores.

Scenario G2: Gas generation and releases in the flooded repository scenario (W3)

In the flooded repository scenario (W3) a situation is described where the drainage systems have failed and the caverns are flooded. With the caverns now saturated the oxygen supply is limited and microbiological degradation mechanisms and corrosion under anaerobic conditions will become important. In this situation the pH value of the water also becomes important. Calculations of the gas generation are based on the amount of metal and biodegradable waste given above.

Based on a pH value of the pore water buffered to a fixed value of 11 it has been calculated that the corrosion of the metal containers by anaerobic corrosion will produce approximately 18 m^3/year of H_2 gas. This production is closely matched by production of CO_2 in the first 20 years. After about 20 years another microbial mechanism becomes dominant producing CH_4 and for a short period consuming H_2 and CO_2. The gas generation peaks after 30 years due to equal contribution from CH_4 and CO_2. Thereafter the gas generation rate drops sharply and CH_4 and H_2S are produced at very much lower rates.

Any gas escaping from the sarcophagi in the flooded caverns is assumed to form bubbles at the top of the cavern roofs. The question is whether the gas will cause significant overpressurisation of the caverns. This is dealt with by the two coupled processes of pressurisation and migration though capillary tubes in the cap rock. In the migration process the gas must displace the water in the capillary tubes in order to escape to the surface. Calculations show that the transit time from the cavern to the surface is between 0.9 and 9.5 days dependent of the aperture size of the capillaries. These transit times corresponds to hydrostatic pressures in the caverns of between 0.6 and 1.0 MPa which does not represent serious overpressurisation of the caverns. Once these channels have been established the permeability of the cap rock is sufficiently high that gas generation rates of nearly four times higher than the highest generation rate calculated could be supported.

Flammability of released gases, explosion risks

Both the hydrogen gas and the methane are flammable in air providing that the gas/air ratios are exceeding certain limits. The risk depends on the rate of gas generation and the extent of confinement. Assuming that the waste containers have not been breached within the first 30 to 40 years as in the foregoing scenarios there is a possibility of some containers becoming pressurised by degradation gases. Steel corrosion rates in concrete is usually very slow.

At the peak generation rate of H_2 inside a single 210 litre barrels of 24 litres/year this gas will be mixed with other gases within the available space of 80 litres. Since this gas generation persists after the O_2 has been consumed this occurs under anaerobic conditions. After this peak production H_2 is consumed by the degradation process producing up to a few hundred litres of CH_4 and CO_2 inside a single drum. However this depends on a multitude of other factors as well. If the waste containers have been breached on an early stage these gases will be vented out though the sarcophagi and caverns and should not present a significant hazard.

5.4 Example from Sweden

SKB has performed a prestudy on long-term performance for the SFL 3-5 repository. The work was carried out in the form of a project, which contained the following parts [5.11]:

- Inventory and characterisation of the waste
- Inventory of Features, Events and Processes (FEPs) that may influence the performance of the repository barriers to radionuclide release
- Selection of data and calculations of near-field releases
- Laboratory experiments and literature studies of important chemical properties.

The prestudy has involved a first attempt to characterise the waste presently planned for SFL 3-5, testing of a systematic scenario methodology and a first evaluation of barrier performance and containment of radionuclides and chemotoxic elements. The waste characterisation was based on rough estimates and the evaluation of barrier performance and containment was restricted to a defined Reference Case which only includes parts of identified mechanisms.

Inventory and characterisation of the waste

Radionuclide content and other safety relevant components were estimated in an attempt to come as close as possible to the actual content of radionuclides, metals, organic materials, etc.

SFL 5 with metallic waste from the reactors will determine the total activity in the repository SFL 3-5. ^{63}Ni will dominate during the first 1000 years and ^{59}Ni thereafter.

Complexing agents in the waste can potentially enhance the release of contaminants by decreasing sorption abilities and increasing solubilities. Organic material and cyanide precipitates are examples of potential sources of complexing agents. It has been established that waste containing organic material will be concentrated to SFL 3 and that the cellulose content will be small. Waste packages with cyanide precipitates conditioned with cement are also foreseen to be allocated to SFL 3.

Steel will be present in all repository parts, as waste, waste packagings and as reinforcement in concrete containers and structures. Much of the steel in the waste is stainless steel. Other metals and metal alloys present are aluminium, Zircalloy, lead, brass, copper, cadmium, etc.

Concrete/cement will be present in all repository parts.

Not only the radionuclide content, but also the content of potentially chemotoxic elements must be considered in a safety assessment. SFL 3-5 will according to first estimates contain chemotoxic elements in some waste types of noticeable quantities, for example certain metals like cadmium lead and beryllium.

Inventory of Features, Events and Processes (FEPs) that may influence the performance of the repository barriers to radionuclide release

Influence Diagrams were used in order to structure graphically the Features, Events and Processes. An advantage of using Influence Diagrams is the possibility to schematically represent the actual lay-out of the repository system. A drawback is a complex system of boxes and arrows, which is a consequence of illustrating all phenomena and their interactions involved in mobilisation and not only the transport paths through the system.

Selection of data and calculations of near-field releases

The near-field releases of radionuclides were calculated for a Reference Case. The calculations revealed that ^{137}Cs and ^{63}Ni would dominate the annual release from all repository parts during the first 1000 years after repository closure and that ^{59}Ni would dominate at longer times.

Near-field releases were also calculated for lead and beryllium. The results showed that some of the barriers effective to prevent release of radionuclides, such as sorption in concrete, are also efficient for chemotoxic elements.

Laboratory experiments and literature studies of important chemical properties

The retention of radionuclides in the concrete dominated environment has turned out to be an important barrier function. The following experiments were started in order to determine how the radionuclides would react in groundwater and cement in a repository:

- Sorption of Eu, Th, Np, Am, Cm, Pm, Co, Ni and Cs in concrete
- Diffusion of Ni, Cs and T in cement paste
- Solubility of Ni, Pu and Eu in cement paste water

Some preliminary results from the experiments were used in the prestudy.

Concrete has important barrier functions in SFL 3 and 5, which are the parts that will contain highest activities. In particular the chemical stability of concrete is important, because concrete pore water will in general enhance retention and suppress solubility of radionuclides in the waste. Investigations of concrete durability have strengthened the conviction that the typical concrete dominated high pH-conditions will remain in the repository during the time when the hazards of the waste has to be considered.

An experimental program on cellulose degradation has been started since degradation of cellulose in concrete may have an influence on the chemistry of radionuclides. A possible explanation for this is that cellulose is hydrolysed at high pH and that the new compounds formed as a result of the hydrolysis are strong complexing agents. The expected products are polyhydroxo-carboxylic acids.

Results so far indicate e.g. that the occurrence of cellulose degradation will enhance the solubility of particularly tetravalent elements at high pH and that the major potential complexing acid that can be formed at substantial yield by alkaline degradation of cellulose is D-glucoisosaccharinic acid.

5.5 Use of an interaction matrix for a simplified Nordic repository concept

To demonstrate the use of the RES-approach (Rock Engineering System) described in Chapter 3.3.1 a workshop session was organised among the participants in the AFA-1.2 project. To illustrate some of the common features in the existing or planned repository systems in the Nordic countries for final disposal of low and intermediate level wastes, a simplified repository system was adopted as the basis for the RES-excercise. Figure 5.1 depicts the main components included in that reference system.

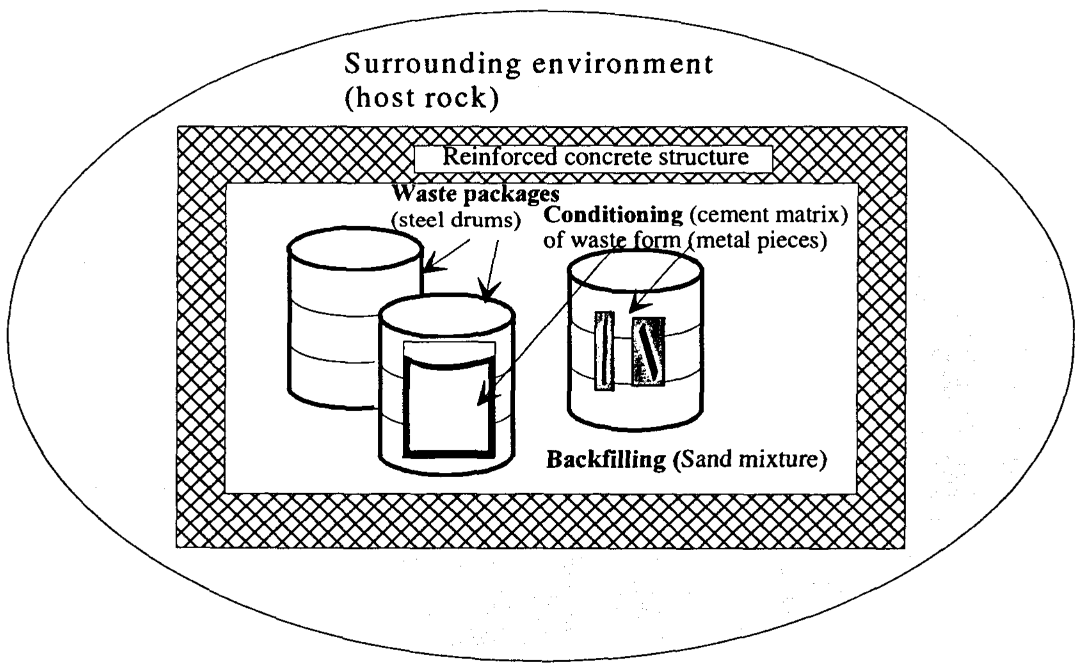


Figure 5.1
A simplified system of engineered safety barriers used as the basis for RES interaction matrix workshop session within the AFA-1.2 subproject.

The following main components or key variables and parameters were chosen as the diagonal elements of the RES-interaction matrix:

1.1	Metallic Waste Form	7.7	Hydrology (near-field)
2.2	Cement Matrix	8.8	Gas
3.3	Steel drum	9.9	Temperature
4.4	Sand backfill	10.10	Radionuclide transport
5.5	Reinforced concrete	11.11	Environment (host rock)
6.6	Water chemistry		

In the workshop session only part of the interactions (mostly in the upper triangle of the interaction matrix) were discussed to demonstrate the basic possibilities of the RES approach emphasising the pedagogical aspect. The following coding was used to describe the strength of interactions:

Priority		**Description**
Number	Colour	
3	Red	**Important interaction- part of the Performance Analysis (PA).** The interaction can be either a prerequisite for the PA or handled by assumptions or modelling efforts in the PA
2	Yellow	**Interaction present - probably part of the Performance Analysis.** Limited or uncertain influence directly or via this interaction on the other parts of the Process System
1	Green	**Interaction present - do not have to be considered in the Performance Analysis.** Negligible influence on other parts of the Process System
0	White	**No identified interactions**

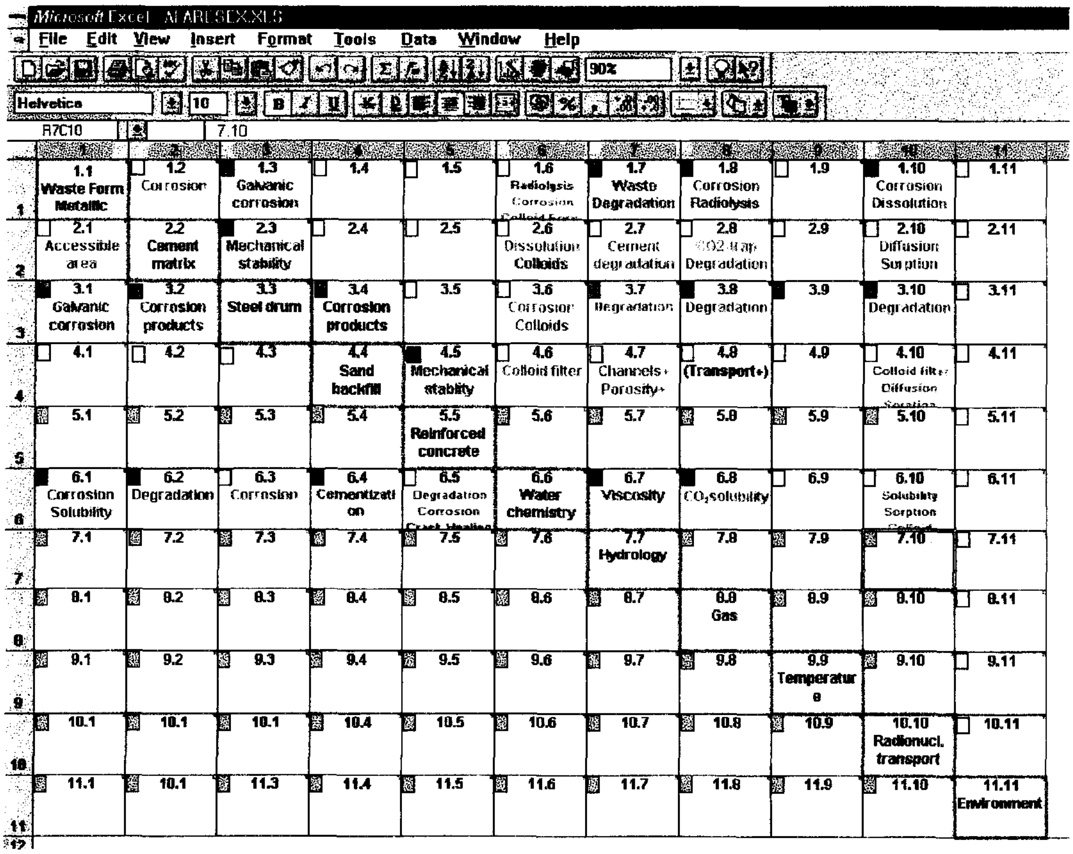
The discussion session was organised in such a way that two persons took note of the discussion. A separate sheet was reserved for each diagonal and interaction element. For each identified interaction the following subjects were discussed:

- Element number, interaction number, name of interaction
- Description of the interaction (which factors/phenomena are causing the interaction and which factors/phenomena have an effect on it)

- Given priority coding/colour and a short verbal reasoning for the choice
- Discussion group, date of session and the competence of the group.

The discussion notes were after the session converted into a tabular form employing a commercial spreadsheet programme (Microsoft Excel-5/7). For each cell a note was attached, which included the main conclusions of the discussion. The outside appearance of each interaction element cell was designed in such a way that the names of interactions identified in a particular element are coloured corresponding to the above coding. Furthermore an overall conclusion of the significance of the interaction is indicated by the colour of a small square in the upper left corner of each cell. The developed software application is shortly described in Annex 1.

A general view of the developed interaction matrix is depicted in the Figure 5.2. A magnified view on one part of the matrix is shown in Figure 5.3. A complete listing of the element-specific notes describing the conclusion derived in the discussion session is included in the Annex 2.



	1	2	3	4	5	6	7	8	9	10	11
1	1.1 Waste Form Metallic	1.2 Corrosion	1.3 Galvanic corrosion	1.4	1.5	1.6 Radiolysis Corrosion	1.7 Waste Degradation	1.8 Corrosion Radiolysis	1.9	1.10 Corrosion Dissolution	1.11
2	2.1 Accessible area	2.2 Cement matrix	2.3 Mechanical stability	2.4	2.5	2.6 Dissolution Colloids	2.7 Cement degradation	2.8 CO2 trap Degradation	2.9	2.10 Diffusion Sorption	2.11
3	3.1 Galvanic corrosion	3.2 Corrosion products	3.3 Steel drum	3.4 Corrosion products	3.5	3.6 Corrosion Colloids	3.7 Degradation	3.8 Degradation	3.9	3.10 Degradation	3.11
4	4.1	4.2	4.3	4.4 Sand backfill	4.5 Mechanical stablity	4.6 Colloid filter	4.7 Channels+ Porosity+	4.8 (Transport+)	4.9	4.10 Colloid filt+ Diffusion	4.11
5	5.1	5.2	5.3	5.4	5.5 Reinforced concrete	5.6	5.7	5.8	5.9	5.10	5.11
6	6.1 Corrosion Solubility	6.2 Degradation	6.3 Corrosion	6.4 Cementizati on	6.5 Degradation Corrosion	6.6 Water chemistry	6.7 Viscosity	6.8 CO_2solubility	6.9	6.10 Solubility Sorption	6.11
7	7.1	7.2	7.3	7.4	7.5	7.6	7.7 Hydrology	7.8	7.9	7.10	7.11
8	8.1	8.2	8.3	8.4	8.5	8.6	8.7	8.8 Gas	8.9	8.10	8.11
9	9.1	9.2	9.3	9.4	9.5	9.6	9.7	9.8	9.9 Temperatur e	9.10	9.11
10	10.1	10.1	10.1	10.4	10.5	10.6	10.7	10.8	10.9	10.10 Radionucl. transport	10.11
11	11.1	10.1	11.3	11.4	11.5	11.6	11.7	11.8	11.9	11.10	11.11 Environment

Figure 5.2
The RES-interaction matrix (partially developed) derived in an AFA-1.2 workshop session for a simplified Nordic repository concept.

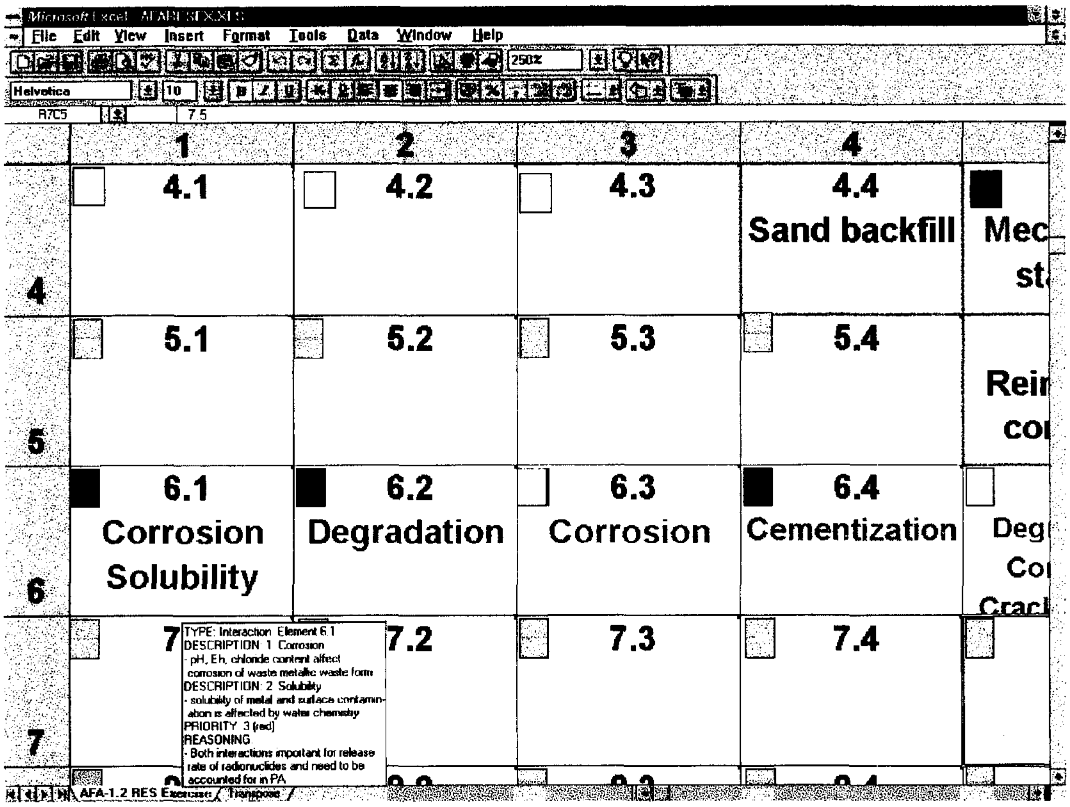


Figure 5.3
A detailed view of the RES-interaction matrix for a simplified Nordic repository concept. Overall conclusions on significance of interactions are indicated with colours.(For the sake of visibility violet colour is used instead of yellow for names of interactions having limited or uncertain influence on safety).

As an overall impression one can conclude that the use of the RES approach was considered by the group to be very easy to learn during the discussion session of restricted length. The use of different ways to indicate the safety significance of various interactions increases the clarity. Furthermore, the developed software application employing a generally available spreadsheet programme provides an easy opportunity to link the cell specific comments readily available for the 'spectator' of the obtained results.

References

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5.6 93046 KLDRA-Combined Storage and Repository for Low- and Intermediate Level Radioactive Waste in Himdalen. Application for licensing the construction of a storage and repository according to the Atomic Energy Act no. 28 - 1972. Detailed scenario analysis. STATSBYGG, Oslo 14th of March 1996

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5.10 Repository for Low- and Intermediate Level Radioactive Waste. Consequence study according to Planning and Building Act (in Norwegian). The Directorate of Public Construction and Property. November 1992.

5.11 WIBORGH, M., Prestudy of final disposal of long-lived low and intermediate level waste, Swedish Nuclear Fuel and Waste Management Co., Technical Report 95-03, 131 p.+ app. 15 p.

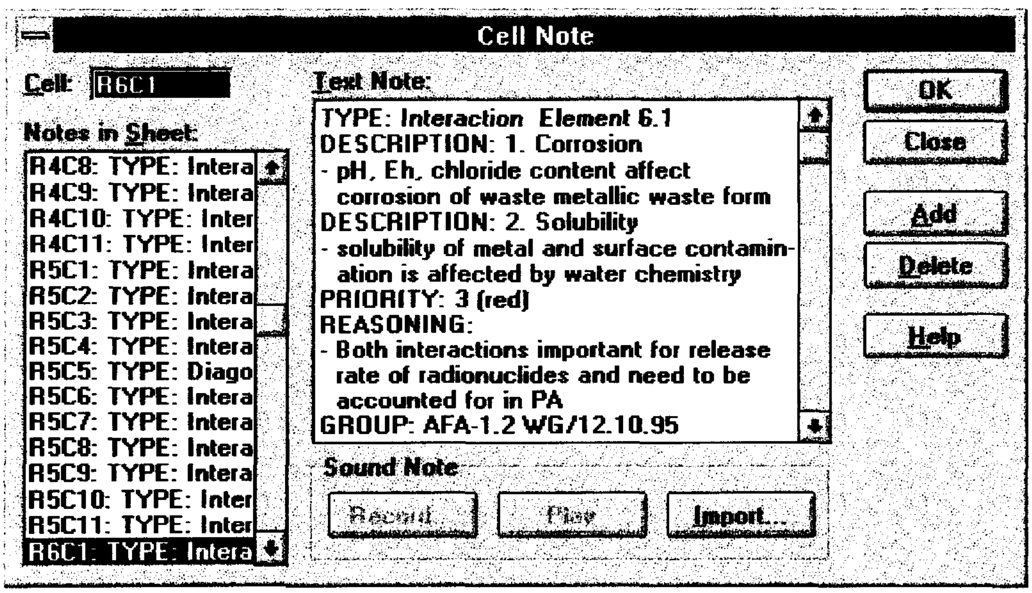
Description of a spreadsheet tool for documenting the conclusions of a discussion session based on RES Interaction - matrix approach

In the documentation of the results of the demonstrative AFA-1.2 exercise the use of Microsoft Excel spreadsheet program turned out to be quite flexible tool for both indicating the safety significance of various element specific interactions and for showing the succinct descriptions of the conclusions of a workshop-type brainstorming session.

Two Excel-files are available from VTT Energy containing on the one hand the documentation of the results (cf. Annex 2) of the AFA-RES-session arranged in October 1995 in Studsvik and on the other hand an empty file that can possibly be employed in other applications.

The names of diagonal and interaction elements have to be first given for the pertinent cells of the main sheet. During the RES session it is probably still useful to take notes on cell and interaction specific sheets and to introduce the comments to the cell-specific notes of the spreadsheet afterwards.

The adding and editing of notes differs slightly in the Excel 5- and Excel 7-versions. In Excel 7 a new note can be added or an existing one edited by choosing the ***Note***-option from the ***Insert*** pulldown menu. After that a specific cell is chosen and the note text added/edited. For each cell note the addition/edition of text have to be accepted by clicking the ***Add*** button. Text can be brought to the cell note also from a separate MS Word-document by choosing there the text that one wishes to use in the note and then copying it (ctrl+c) and then going to the opened cell note editing window and then pasting the text to the new destination by (shift+Insert) command. Also the transfer in opposite direction is possible and the cell note contents can be copied for example into a table like that in Annex 2 (firstly ctrl+c in the note and then shift+Insert in the Word-document). The cell note window is illustrated in the figure below. In the main spreadsheet the cell-specific notes can in the case of Excel7-version easily be seen by moving the cursor by mouse to the pertinent element.



In the Excel5-version the introduction of a new note or the editing of an existing one has to be started by choosing from the ***Tools*** pulldown menu first ***Auditing*** and then ***Show Auditing Toolbar***. After that the note addition/editing can be started by pushing the ***Attach Note*** button in this toolbar. The same procedure can also be employed for Excel7 in addition to the way described on the previous page. By pushing the ***Info*** button in the auditing toolbar one can see the cell note in a fullscreen mode that can be closed from the ***Close*** option in the ***File*** pulldown menu.

The main spreadsheet and the notes can be printed either separately or as combined. In case one wishes to only print the main spreadsheet (without notes) one should remove the tick in the notes box in the ***Sheet***-section of ***Page Setup*** (from ***File*** pulldown menu).

	1	2	3
1	TYPE: **Diagonal Element 1.1** DESCRIPTION: ***Metallic Waste Form*** Pieces of aluminium &steel no organic material	TYPE: Interaction Element 1.2 DESCRIPTION: *Corrosion* – Corrosion of metal affects mechanically the cement matrix – Corrosion changes the hydraulic properties (through expansion & deposition of corrosion products) of matrix (porosity & hydraulic conductivity) PRIORITY: 2 (yellow) REASONING: – Can be of importance in certain cases – Depends whether one accounts matrix as a safety barrier	TYPE: Interaction Element 1.3 DESCIPTION: *Galvanic corrosion* – Potential impact on corrosion of steel drums PRIORITY: max. 1 (green) REASONING: – Low significance compared to other mechanisms – Normally certain distance between waste product metal and steel drum
2	TYPE: Interaction Element 2.1 DESCRIPTION: *Accessible area* – Accessible area of waste form increases via changes of matrix properties PRIORITY: 2 (yellow) REASONING: May increase release rate	TYPE: **Diagonal Element 2.2** DESCRIPTION: ***Cement Matrix*** – Homogenous mixture – Completely filled – Standard cement – No organic extra additives – Processes within this diagonal element, such as ageing caused by internal reasons requires separate RES scheme	TYPE: Interaction Element 2.3 DESCRIPTION: *Mechanical stability* – Changes in characteristics of cement matrix may reduce mechanical stability of steel drums PRIORITY: 1 (green) REASONING: – Steel drums have low release barrier importance
3	TYPE: Interaction Element 3.1 DESCRIPTION: *Galvanic corrosion* – Galvanic corrosion affects the corrosion of metal pieces in waste form PRIORITY: 1 (green) REASONING: – Other types of corrosion probably more important – Normally distance between metals within cement matrix and drum wall	TYPE: Interaction Element 3.2 DESCRIPTION: *Corrosion products* – Corrosion products may fill pore space in cement matrix and thereby change the properties of matrix PRIORITY: 1 (yellow) REASONING: – Would probably reduce release rates	TYPE: **Diagonal Element 3.3** DESCRIPTION: ***Steel drum*** – Non-tight steel container & lock – Painted with ordinary paint TYPE: Interaction Element 4.3 DESCRIPTION: No direct interaction PRIORITY: 0 REASONING:
4	TYPE: Interaction Element 4.1 DESCRIPTION: No interaction PRIORITY: 0 REASONING: Physically separated	TYPE: Interaction Element 4.2 DESCRIPTION: No interaction (unless very poor properties of cement or very heavy sand) PRIORITY: 0 REASONING:	TYPE: Interaction Element 4.3 DESCRIPTION: No direct interaction PRIORITY: 0 REASONING:
5	TYPE: Interaction Element 5.1 Not considered	TYPE: Interaction Element 5.2 Not considered	TYPE: Interaction Element 5.3 Not considered
6	TYPE: Interaction Element 6.1 DESCRIPTION: 1. *Corrosion* – pH, Eh, chloride content affect corrosion of waste metallic waste form DESCRIPTION: 2. Solubility – Solubility of metal and surface contamination is affected by water chemistry PRIORITY: 3 (red) REASONING: – Both interactions important for release rate of radionuclides and need to be accounted for in PA	TYPE: Interaction Element 6.2 DESCRIPTION: *Degradation* – E.g. sulphate, magnesium, carbonate, chlorides in water degrade properties of cement matrix PRIORITY: 3 (red) REASONING: – Significant impact on barrier performance	TYPE: Interaction Element 6.3 DESCRIPTION: *Corrosion (&solubility ?)* – Cf. interaction 6.1 – Corrosion produces also gases PRIORITY: 2 (yellow) REASONING: – Reliance on steel drums as safety barrier low – Gas formation has to be accounted for in PA
7	TYPE: Interaction Element 7.1 Not considered	TYPE: Interaction Element 7.2 Not considered	TYPE: Interaction Element 7.3 Not considered
8	TYPE: Interaction Element 8.1 Not considered	TYPE: Interaction Element 8.2 Not considered	TYPE: Interaction Element 8.3 Not considered
9	TYPE: Interaction Element 9.1 Not considered	TYPE: Interaction Element 9.2 Not considered	TYPE: Interaction Element 9.3 Not considered
10	TYPE: Interaction Element 10.1 Not considered	TYPE: Interaction Element 10.2 Not considered	TYPE: Interaction Element 10.3 Not considered
11	TYPE: Interaction Element 11.1 Not considered	TYPE: Interaction Element 11.2 Not considered	TYPE: Interaction Element 11.3 Not considered

	4	5	6
1	TYPE: Interaction Element 1.4 DESCRIPTION: No interaction PRIORITY: 0 REASONING: Physically separated	TYPE: Interaction Element 1.5 DESCRIPTION: No interaction PRIORITY: 0 REASONING: Physically separated	TYPE: Interaction Element 1.6 DESCRIPTION: 1. *Radiolysis* – Radiation from waste changes water chemistry – pH changes due to radiolysis PRIORITY: 1 (green) REASONING: – Low significance owing to waste type (low-activity) DESCRIPTION: 2. *Corrosion* – Corrosion affects pH & Eh PRIORITY: 2 (yellow) REASONING: – Normally cement has strongest effect on water chemistry – Later-on corrosion may have increasing effect DESCRIPTION: 3. *Colloid formation* – Colloid formation in contact with water (probably colloids cannot be transported out of (intact) cement matrix) PRIORITY: 1 (green) REASONING: – Cement has greater (dominant) significance
2	TYPE: Interaction Element 2.4 DESCRIPTION: No interaction PRIORITY: 0 REASONING: – Physically separated while steel drums remain intact; potential impact via water chemistry in case of damaged drums	TYPE: Interaction Element 2.5 DESCRIPTION: No interaction PRIORITY: 0 REASONING: – Physically separated while steel drums remain intact and backfill undamaged; potential impact via water chemistry (in case of damaged drums)	TYPE: Interaction Element 2.6 DESCRIPTION: 1. *Dissolution* – Dissolution of concrete affects composition of water and its chemical characteristics (e,g. pH) PRIORITY: 2 (yellow) REASONING: – Might increase release rates DESCRIPTION: 2. *Colloids* – Dissolution of cement affects colloid build-up and indirectly water chemistry PRIORITY: 1 (green) REASONING: – Other mechanisms presumably more important
3	TYPE: Interaction Element 3.4 DESCRIPTION: *Corrosion products* – Similar to interaction 3.2 PRIORITY: 1 (green) REASONING	TYPE: Interaction Element 3.5 DESCRIPTION: No interaction – Cf. interactions 1.2 & 2.5 PRIORITY: 0 REASONING:	TYPE: Interaction Element 3.6 DESCRIPTION: *Corrosion & Colloids* PRIORITY: 2 (yellow) REASONING: – Cf. interaction 1.6
4	TYPE: **Diagonal Element 4.4** DESCRIPTION: ***Sand backfill*** – Inert sand material – Permeability = ? – Other physical, chemical & mechanical properties – Not fully homogenous – Saturated (water level above concrete structure)	TYPE: Interaction Element 4.5 DESCRIPTION: *Mechanical stability* – Sand backfill supports concrete structure (prevents collapse) PRIORITY: 1 (green) REASONING: – Favours barrier integrity	TYPE: Interaction Element 4.6 DESCRIPTION: *Colloid filter* – Sand backfill traps colloids that have been produced elsewhere PRIORITY: 2 (yellow) REASONING: – Inhibits rapid transport & subsequent release of radionuclides attached to colloids
5	TYPE: Interaction Element 5.4 Not considered	TYPE: **Diagonal Element 5.5** DESCRIPTION: ***Reinforced concrete*** – Concrete structure – Separate lid (cover) – Spricks between lid and walls	TYPE: Interaction Element 5.6 Not considered

	4	5	6
6	TYPE: Interaction Element 6.4 DESCRIPTION: *Cementization* – Dissolution of cement by water of suitable characteristics may bring about partial cementization of sand PRIORITY: 1 (green) REASONING: – Probably beneficial effect	TYPE: Interaction Element 6.5 DESCRIPTION: 1. *Degradation* – Chemical "attack" degrades concrete properties DESCRIPTION: 2. *Corrosion* – Corrosion of reinforcing steel bars degrades mechanical properties – DESCRIPTION: 3. *Crack healing* – Tightening of walls and cracks by calcination (fills up pores) PRIORITY: 2 (yellow) REASONING: – Interactions may have marked positive or negative impacts on barrier performance	TYPE: **Diagonal Element 6.6** DESCRIPTION: ***Water chemistry*** – Contains chloride, sulfate, biocarbonate – pH = – Eh = – Contains complexing agents, colloids &microbes – Important both inside canisters and in concrete structure
7	TYPE: Interaction Element 7.4 Not considered	TYPE: Interaction Element 7.5 Not considered	TYPE: Interaction Element 7.6 Not considered
8	TYPE: Interaction Element 8.4 Not considered	TYPE: Interaction Element 8.5 Not considered	TYPE: Interaction Element 8.6 Not considered
9	TYPE: Interaction Element 9.4 Not considered	TYPE: Interaction Element 9.5 Not considered	TYPE: Interaction Element 9.6 Not considered
10	TYPE: Interaction Element 10.4 Not considered	TYPE: Interaction Element 10.5 Not considered	TYPE: Interaction Element 10.6 Not considered
11	TYPE: Interaction Element 11.4 Not considered	TYPE: Interaction Element 11.5 Not considered	TYPE: Interaction Element 11.6 Not considered

	7	8	9
1	TYPE: Interaction Element 1.7 DESCRIPTION: *Waste Degradation* – Indirect effect via interaction 1.2 on water flow through drums PRIORITY: 1 (green) REASONING: – Of lower importance compared to the effect of cement matrix	TYPE: Interaction Element 1.8 DESCRIPTION: 1. *Corrosion* – Corrosion of metal causes gas (hydrogen) formation PRIORITY: 3 (red) REASONING: – Relatively large amounts of gas can be produced – Time aspects and quantities of metal (Fe & Al) modify significance DESCRIPTION: 2. *Radiolysis* – Radiolysis causes gas build-up PRIORITY: 1 (green) REASONING: – Low significance compared to other effects	TYPE: Interaction Element 1.9 DESCRIPTION: No interaction PRIORITY: 0 REASONING: – Non-heat generating waste
2	TYPE: Interaction Element 2.7 DESCRIPTION: *Cement degradation* – Degradation of cement matrix changes the hydraulic (permeability, porosity, cracking) properties of matrix PRIORITY: 2 (yellow) REASONING: – Could increase release rate	TYPE: Interaction Element 2.8 DESCRIPTION: 1. *CO_2-trap* – Cement matrix can trap extra carbon dioxide – Gas transport characteristics may change PRIORITY: 2 (yellow) REASONING: – May increase RN release rates DESCRIPTION: 2. *Degradation* – Degradation of cement matrix may change transport velocities of gases PRIORITY: 2 (yellow) – May increase source-term	TYPE: Interaction Element 2.9 DESCRIPTION: No interaction PRIORITY: 0 REASONING:
3	TYPE: Interaction Element 3.7 DESCRIPTION: *Degradation* – Degradation (e.g. corrosion) of steel drums changes properties of sand backfill and thereby hydrological conditions PRIORITY: 2 (yellow) REASONING: – May affect release rates (SV)	TYPE: Interaction Element 3.8 DESCRIPTION: *Degradation* – Gas formation by steel drum corrosion PRIORITY: 2 (yellow) REASONING: – Gases may act as carrier of radionuclides	TYPE: Interaction Element 3.9 DESCRIPTION: No interaction PRIORITY: REASONING:

	7	8	9
4	TYPE: Interaction Element 4.7 DESCRIPTION: 1. *Channel formation* – Water flow in sand brings about new channels or expands existing ones DESCRIPTION: 2. *Porosity increase* – Settling/agglomeration of sand changes porosity distribution (homogenity) PRIORITY: 2 (yellow) REASONING: – Both impacts may speed up release process	TYPE: Interaction Element 4.8 DESCRIPTION: *Conductivity increased by gas transport* – No significant effect due to high initial porosity of sand PRIORITY: 0 REASONING:	TYPE: Interaction Element 4.9 DESCRIPTION: No interaction PRIORITY: 0 REASONING:
5	TYPE: Interaction Element 5.7 Not considered	TYPE: Interaction Element 5.8 Not considered	TYPE: Interaction Element 5.9 Not considered
6	TYPE: Interaction Element 6.7 DESCRIPTION: *Viscosity changes* – Hypothetical effect (= increase in flow rate) – Presumably of low importance for this waste type PRIORITY: 0-1 REASONING: see above	TYPE: Interaction Element 6.8 DESCRIPTION: *CO_2-solubility* – Water chemistry (especially pH) affects solubility of CO_2 PRIORITY: REASONING:	TYPE: Interaction Element 6.9 DESCRIPTION: No interaction PRIORITY: 0 REASONING:
7	TYPE: **Diagonal Element 7.7** DESCRIPTION: ***Hydrology (near-field)*** – Hydraulic head (water pressure) – Water flow rate in near-field – Flow directions	TYPE: Interaction Element 7.8 Not considered	TYPE: Interaction Element 7.9 Not considered
8	TYPE: Interaction Element 8.7 Not considered	TYPE: **Diagonal Element 8.8** DESCRIPTION: ***Gas*** – Gas pressure – Gas quantities – Gas transport – Radioactive gases (also decay products) – Corrosion gases – Only near-field relevant	TYPE: Interaction Element 8.9 Not considered
9	TYPE: Interaction Element 9.7 Not considered	TYPE: Interaction Element 9.8 Not considered	TYPE: **Diagonal Element 9.9** DESCRIPTION: ***Temperature*** – Temperature in the near-field (within concrete structure)
10	TYPE: Interaction Element 10.7 Not considered	TYPE: Interaction Element 10.8 Not considered	TYPE: Interaction Element 10.9 Not considered
11	TYPE: Interaction Element 11.7 Not considered	TYPE: Interaction Element 11.8 Not considered	TYPE: Interaction Element 11.9 Not considered

	10	11
7	TYPE: Interaction Element 7.10 Not considered	TYPE: Interaction Element 7.11 DESCRIPTION: No interaction PRIORITY: 0 REASONING:
8	TYPE: Interaction Element 8.10 Not considered	TYPE: Interaction Element 8.11 DESCRIPTION: No interaction PRIORITY: 0 REASONING:
9	TYPE: Interaction Element 9.10 Not considered	TYPE: Interaction Element 9.11 DESCRIPTION: No interaction PRIORITY: 0 REASONING:
10	TYPE: **Diagonal Element 10.10** DESCRIPTION: ***Radionuclide transport*** – Includes both release & transport in the near field – Main aim of analysis: release from near-field	TYPE: Interaction Element 10.11 DESCRIPTION: No interaction PRIORITY: 0 REASONING:
11	TYPE: Interaction Element11.10 Not considered	TYPE: **Diagonal Element 11.11** DESCRIPTION: ***Environment*** – Saturated with water – Below groundwater table – Water chemistry of surface water – Reducing geochemical conditions – Constant hydraulic gradient & water flow – Constant temperature in the environment – Mechanically stable environment

	10	11
1	TYPE: Interaction Element 1.10 DESCRIPTION: 1. *Corrosion* – Corrosion of metal leads to release of activation products PRIORITY: 3 (red) REASONING: – Determines the source-term of activation products DESCRIPTION: 2. *Dissolution* – Radionuclides from surface contaminated waste can be dissolved – Water chemistry dictates the solubility PRIORITY: 3 (red) REASONING: – Important for source-term (from surface contamination)	TYPE: Interaction Element 1.11 DESCRIPTION: No interaction PRIORITY: 0 REASONING:
2	TYPE: Interaction Element 2.10 DESCRIPTION: 1. *Diffusion* – Porosity of cement matrix affects diffusion rate PRIORITY: 2 (yellow) REASONING: – May increase release rates DESCRIPTION: 2. *Sorption* – Sorption mechanisms can (temporarily) attach (adsorb) radionuclides and hence retard RN transport – Accessible area and mineralogical properties of cement constituents affect sorption PRIORITY: 2 (yellow) REASONING: – Same as for interaction #1	TYPE: Interaction Element 2.11 DESCRIPTION: No interaction PRIORITY: 0 REASONING:
3	TYPE: Interaction Element 3.10 DESCRIPTION: *Degradation* – Diffusion properties of sand backfill change owing to precipitation of corrosion product from steel drums {SV} – Transport velocity increased {AS} – Permeability of steel drums increased {AS} PRIORITY: 2 (yellow) REASONING: – Release rate of radionuclides may be affected (reduced {SV} or increased {AS})	TYPE: Interaction Element 3.11 DESCRIPTION: No interaction PRIORITY: 0 REASONING:
4	TYPE: Interaction Element 4.10 DESCRIPTION: 1. *Colloid filter* – Cf. interaction 4.6 DESCRIPTION: 2. *Diffusion* – Main transport mechanism for release DESCRIPTION: 3. *Sorption* – Radionuclides are attached to/adsorbed on sand (in this respect sand is not inert) PRIORITY: 2 (yellow) REASONING: – Filtering inhibits rapid transport & subsequent release of radionuclides attached to colloids sorption retards movement of RN	TYPE: Interaction Element 4.11 DESCRIPTION: No interaction PRIORITY: 0 REASONING:
5	TYPE: Interaction Element 5.10 Not considered	TYPE: Interaction Element 5.11 DESCRIPTION: No interaction PRIORITY: 0 REASONING:
6	TYPE: Interaction Element 6.10 DESCRIPTION: 1. *Solubility* – Complexing agents, pH, Eh, ionic strength, Ca& Cl- contents affect solubility DESCRIPTION: 2. *Sorption* – Cf. interaction 6.1 DESCRIPTION: 3. *Colloid stability* – Ionic strength affects stability of colloids PRIORITY: 2 (yellow) REASONING: – All interactions relevant for PA	TYPE: Interaction Element 6.11 DESCRIPTION: No interaction PRIORITY: 0 REASONING: