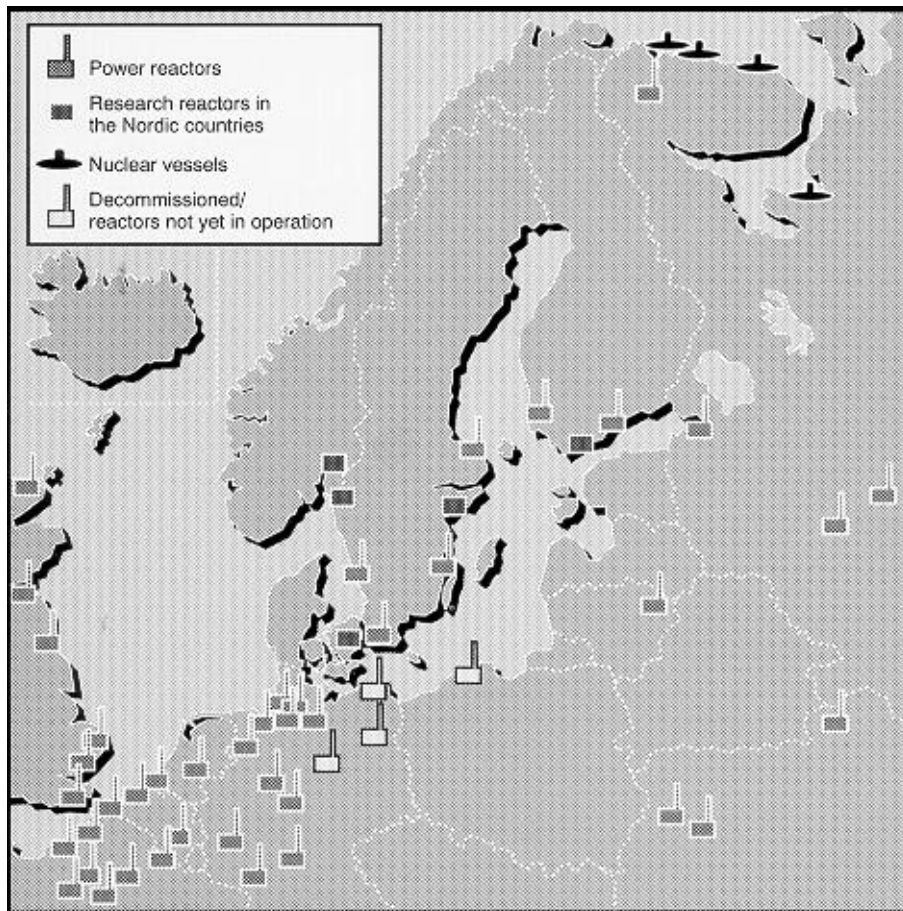


Safety in the Final Disposal of Radioactive Waste



nks

Nordic Nuclear Safety Research (NKS)

organizes joint four-year research programs involving some 300 Nordic scientists and dozens of central authorities, nuclear facilities and other concerned organizations in five countries. The aim is to produce practical, easy-to-use background material for decision makers and help achieve a better understanding of nuclear issues.

To that end, the results of the fifth four-year NKS program (1994 – 1997) are herewith presented in a series of final reports comprising reactor safety, waste management, radioecology, nuclear emergency preparedness and information issues. Each report summarizes one of the ten projects carried out during that period, including the administrative support and coordination project. A special Summary Report, with a brief résumé of all ten projects, is also published. Additional copies of the reports on the individual projects can be ordered free of charge from the NKS Secretariat.

The final reports – together with some technical reports and other material produced during the 1994 – 1997 period – have been collected on a CD-ROM, also available free of charge from the NKS Secretariat.

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Safety in the Final Disposal of Radioactive Waste

Final Report of the Nordic Nuclear
Safety Research Project AFA-1

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December 1997

This is NKS

NKS (Nordic Nuclear Safety Research) is a scientific cooperation program in nuclear safety, radiation protection and emergency preparedness. Its purpose is to carry out cost-effective Nordic projects, thus producing research results, exercises, information, manuals, recommendations, and other types of background material. This material is to serve decision-makers and other concerned staff members at authorities, research establishments and enterprises in the nuclear field.

The following major fields of research are presently dealt with: reactor safety, radioactive waste, radioecology, emergency preparedness and information issues. A total of nine projects have been carried out in the years 1994 - 1997.

Only projects that are of interest to end-users and financing organizations have been considered, and the results are intended to be practical, useful and directly applicable. The main financing organizations are:

- The Danish Emergency Management Agency
- The Finnish Ministry for Trade and Industry
- The Icelandic Radiation Protection Institute
- The Norwegian Radiation Protection Authority
- The Swedish Nuclear Power Inspectorate and the Swedish Radiation Protection Institute

Additional financial support has been given by the following organizations:

In Finland: Ministry of the Interior; Imatran Voima Oy (IVO); Teollisuuden Voima Oy (TVO)

In Norway: Ministry of the Environment

In Sweden: Swedish Rescue Services Board; Sydkraft AB; Vattenfall AB; Swedish Nuclear Fuel and Waste Management Co. (SKB); Nuclear Training and Safety Center (KSU)

To this should be added contributions in kind by several participating organizations.

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Disclaimer

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In particular, neither NKS nor any other organization or body supporting NKS activities can be held responsible for the material presented in this report.

Abstract

During 1994-1997 a project on disposal of radioactive waste was carried out as part of the NKS program. The objective of the project was to provide authorities and waste producers in the Nordic countries with background material for determinations about the management and disposal of radioactive waste. The project was called NKS/AFA-1. It was divided into three sub-projects: AFA-1.1, AFA-1.2 and AFA-1.3. AFA-1.1 dealt with waste characterisation, AFA-1.2 dealt with performance assessment for repositories and AFA-1.3 dealt with Environmental Impact Assessment (EIA). The studies were mainly focused on the management of long-lived low- and intermediate-level radioactive waste from research, hospitals and industry.

Representatives from all Nordic countries have participated in each of the sub-projects. Most of the work has been performed in a broad circle. This has contributed to a better understanding of the waste situation in the different countries and has also made it possible to learn from each other. Furthermore, in some cases, it has contributed to common recommendations. Recommendations have e.g. been given regarding the characterisation of waste under treatment and the characterisation of existing and old waste packages. The waste characterisation study (AFA-1.1) included also a survey of waste types, methods used for the measurement of activity content, methods used for the determination of chemical content- and an inventory of new available methods.

The performance assessment study (AFA-1.2) covered performance assessment methodologies, interactions in the near-field of a repository and examples of performance assessment in the Nordic countries. A survey was also made of waste disposal systems in the Nordic countries.

The Environmental Impact Assessment (EIA) study (AFA-1.3) included differences and similarities between the principles of handling the EIA in the Nordic countries. Examples were provided of nuclear-related EIA in Finland, Norway and Sweden. From Denmark and Iceland, examples outside this area were given.

Key words

Radioactive waste, waste management, waste disposal, waste characterisation, safety analysis, environmental impact statement.

Summary with conclusions and recommendations

General

During 1994-1997 a project on the disposal of radioactive waste was carried out as part of the NKS program. The objective of the project was to give authorities and waste producers in the Nordic countries background material for determinations about the management and disposal of radioactive waste. The project was called NKS/AFA-1. It was divided into three sub-projects: AFA-1.1, AFA-1.2 and AFA-1.3. AFA-1.1 dealt with waste characterisation, AFA-1.2 dealt with performance assessment for repositories and AFA-1.3 dealt with Environmental Impact Assessment (EIA). The studies mainly focused on the management of long-lived low- and intermediate-level radioactive waste from research, hospitals and industry.

Representatives from all Nordic countries have participated in each of the sub-projects. Most of the work has been performed in a broad group of experts. This has contributed to a better understanding of the waste situation in the different countries and has also made it possible to learn from each other. Furthermore, in some cases it has contributed to common recommendations.

Waste characterisation (AFA-1.1)

The AFA-1.1 study included an overview on waste categories in the Nordic countries and methods to determine or estimate the waste content. New available methods were presented based on answers to questionnaires that were sent out to suppliers.

The study includes also recommendations regarding the characterisation of waste under treatment and the characterisation of existing and old waste packages. It is advisable to, if possible, obtain information concerning waste under treatment. Classification of the waste according to physical and chemical composition is also most simply achieved during the treatment. However, when radioactive waste is handled, the dose rate measurement should be the first precaution prior to any other procedures. Reference nuclides can sometimes be used for estimations on isotopes which are difficult to measure.

New regulations for the inventory of a repository may demand new assessments of old radioactive waste packages. The existing documentation of a waste package is then the primary information source although additional measurements may be necessary.

Performance analysis (AFA-1.2)

The AFA-1.2 project dealt with the performance assessment of the engineered barrier system (near-field) of the repositories for low- and intermediate-level radioactive waste. The topic intentionally excluded the discussion of the characteristics of the geological host medium. Therefore, a more generic discussion of the features of performance assessment was possible independent of the fact that different host media are considered in the Nordic countries.

The results from the AFA-1.2 study include a short overview of different waste management systems existing and planned in the Nordic countries. However, the main emphasis of the study was a general discussion of methodologies developed and employed for performance assessments of waste repositories. Some of the phenomena and interactions relevant for generic types of repository were discussed as well. Among the different approaches for the development of scenarios for safety and performance assessments one particular method, the Rock Engineering System (RES), was chosen to be tested by demonstration. The possible interactions and their safety significance were discussed, employing a simplified and generic Nordic repository system as the reference system.

A short review of performance assessments carried out in the Nordic countries for actual projects concerning repositories for low- and intermediate-level waste was also included in the study.

Environmental impact assessment (AFA-1.3)

The results from the AFA-1.3 study include information on similarities and differences between the EIA in the Nordic countries and also a review of experiences from EIA in the countries, both within the nuclear field and outside the nuclear field.

The system for environmental impact assessment (EIA) in a country depends on the legislative structure, the application of legislation, administrative practice and general social objectives. It is therefore natural that the EIA systems differ from country to country, even if the directives of the European Community and internationally accepted principles are adopted. There are e.g. differences in the objectives for the EIA systems in the Nordic countries. The EIA system in Denmark must provide a guarantee that a specific assessment of environmental consequences for certain projects has been carried out at the level of the planning of site use. Emphasis should be placed on public participation and an open decision process. The EIA systems in Finland, Iceland and Norway must guarantee that a special assessment of environmental consequences has been carried out for certain projects. Emphasis should be placed on project planning and public participation. The EIA system in Sweden shall give the authorities a basis for assessment of the effect on environment, health, safety and general interests in accordance with the Swed-

ish Act on the Management of Natural Resources for a broad spectrum of projects. Differences can also be found regarding responsibilities for the Environmental Impact Statement (EIS). The proponent of the project is responsible for the EIS in Finland, Iceland, Norway and Sweden. The authority is responsible for the EIS in Denmark.

Sammanfattning med slutsatser och rekommendationer

Generellt

Under 1994-1997 genomfördes inom ramen för NKS (Nordisk kärnsäkerhetsforskning) ett projekt om slutförvaring av radioaktivt avfall. Syftet med projektet var att ge myndigheter och avfallsproducenter i de nordiska länderna underlag för undersökningar beträffande behandling och slutförvaring av radioaktivt avfall. Projektet kallades NKS/AFA-1. Det uppdelades i tre under-projekt: AFA-1.1, AFA-1.2 och AFA-1.3. AFA-1.1 handlar om avfallskaraktisering, AFA-1.2 handlar om funktionsanalys för slutförvar och AFA-1.3 handlar om miljökonsekvens-beskrivningar (MKB; MKB är i Finland förkortning för miljökonsekvensbedömning). Studierna fokuserades huvudsakligen på hantering av lång-livat låg- och medelaktivt avfall från forskning, sjukhus och industri.

Representanter från alla de nordiska länderna har deltagit i vart och ett av underprojekten. Huvuddelen av arbetet har genomförts med ett brett deltagande. Detta har bidragit till bättre förståelse för avfallssituationen i de olika länderna och också gjort det möjligt att lära från varandra. Dessutom har arbetet i några fall bidragit till gemensamma rekommendationer.

Avfallskaraktisering (AFA-1.1)

AFA-1.1-studien inkluderade en översikt av avfallskategorier i den nordiska länderna och metoder att bestämma eller uppskatta avfallsinnehållet. Nya tillgängliga metoder presenterades baserade på svar på förfrågningar, som skickats ut till leverantörer.

Inom studien gjordes dessutom rekommendationer beträffande karakterisering av avfall under behandling och karakterisering av befintliga och gamla avfallskollin. Det är tillrådligt att om möjligt få information om avfallet under behandling. Det är också enklast att klassificera avfallets kemiska och fysikaliska sammansättning under behandlingen. När radioaktivt avfall hanteras bör emellertid dosräkningar göras innan alla övriga procedurer. Referensnuklider kan ibland användas för uppskattningar om svårsmätbara nuklider.

Nya regler om avfalls-inventarium kan kräva nya uppskattningar om gamla avfallskollin. Den befintliga dokumentationen om ett avfallskolli är då den primära informationskällan.

Funktionsanalys (AFA-1.2)

AFA-1.2-projektet behandlade funktionsanalys för barriärsystemet i närområdet till slutförvar för lång-livat låg- och medelaktivt avfall. Ämnesområdet begränsades med avsikt så att det omgivande geomediet inte diskuterades. Detta gjorde det möjligt att diskutera genomförande av en mer generell funktionsanalys, som är oberoende av att olika geomedier kan komma i fråga i de nordiska länderna.

Resultaten från AFA-1.2-studien inkluderade en kort översikt över olika befintliga och planerade avfallshanteringssystem i de nordiska länderna. Tonvikten i projektet lades emellertid på en generell diskussion om metodiker utvecklade för och tillämpade på funktionsanalys för avfallsförvar. Några företeelser och interaktioner som gäller för ett generellt förvar diskuterades också. En metod, den så kallade RES-metoden (Rock Engineering System) valdes ut för demonstration. Möjliga interaktioner och deras betydelse diskuterades för en förenklad generell nordisk förvarstyp som referenssystem.

I studien inkluderades också en kort översikt över genomförda funktionsanalyser i Norden för aktuella slutförvarsprojekt.

Miljökonsekvensbeskrivningar (AFA-1.3)

Resultaten från AFA-1.3 studien inkluderar information om likheter och skillnader mellan MKB-processer (miljökonsekvensbeskrivningsprocesser) i de nordiska länderna samt en översikt över erfarenheter från MKB-processer i länderna, både inom och utom det nukleära området.

Systemet för miljökonsekvensbeskrivningar i ett land är beroende av lagstiftningens struktur, lagstiftningens tillämpning, administrativ praxis och allmänna samhällliga målsättningar. Det är därför naturligt att MKB-systemen skiljer sig från land till land även om EU-direktiv och internationellt accepterade principer tillämpas. Det finns t. ex. skillnader i mål för MKB-system i de nordiska länderna. MKB-system i Danmark ska ge garanti för specifik bedömning av miljökonsekvenser för vissa projekt i samband med markanvändning. Tyngdpunkten ska läggas på allmänhetens deltagande och en öppen beslutsprocess. MKB-system i Finland, Island och Norge ska ge garanti för särskild bedömning av miljökonsekvenser för vissa projekt. Tyngdpunkten ska läggas på den projektansvariges planering och allmänhetens deltagande. MKB-system i Sverige ska ge beslutsmyndigheter underlag för bedömning av inverkan på miljö, hälsa och säkerhet samt på allmänna intressen enligt naturresurslagen för ett brett spektrum av projekt. Skillnader finns också beträffande ansvar för miljökonsekvensbeskrivningar. Byggherren för byggprojektet är ansvarig för MKBn i Finland, Island, Norge och Sverige. Myndigheten är ansvarig för MKBn i Danmark.

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Appendix 1: Abbreviations

Appendix 2: Participants

Appendix 3: Financing

1 Introduction

The AFA-1 project focused originally on safety in connection with the final disposal of long-lived low and intermediate-level radioactive waste [NKS 94]. All Nordic countries have this type of waste. The waste originates from power reactors (Finland, Sweden), research/test reactors (Denmark, Finland, Norway, Sweden) and also from medicine, research and industry (Denmark, Finland, Iceland, Norway, Sweden).

However, the project was somewhat extended so that it was no longer limited to long-lived low- and intermediate-level radioactive waste.

Denmark has interim storage facilities for radioactive waste at the research centre Risø, but has not started the planning work for final disposal. Short-lived low- and intermediate-level radioactive waste is handled together with long-lived low- and intermediate-level radioactive waste at Risø.

Denmark has no experience from environmental impact assessments for repositories for radioactive waste but ,on the other hand, has experience from other environmental impact assessment procedures.

The long-lived low- and intermediate-level radioactive waste in Finland will be deposited either in repositories near the reactor waste repositories in Loviisa and Olkiluoto or near to the spent fuel repository. The nuclear power plants are the main producers of this type of waste.

The environmental impact assessment procedures in Finland for the disposal of long-lived radioactive waste mainly focus on the disposal of spent fuel.

Iceland only has small quantities of radioactive waste from hospitals, research and industry. The growing international tendency that each country should manage its own waste makes it important not only to consider international solutions for the disposal of radioactive waste from Iceland but also to study the possibility of finding a suitable disposal site in Iceland.

Iceland has no experience from environmental impact assessment procedures for repositories for radioactive waste but has experience from other environmental impact assessment procedures.

Low- and intermediate level radioactive waste in Norway is handled at Kjeller together with other types of radioactive waste. A combined disposal and storage facility for radioactive waste is built at Himdalen. This facility will be used for long- and short-lived low-level radioactive waste. When the facility is closed, it

will be decided whether to retrieve the stored waste or to convert this part of facility into a repository.

The long-lived low- and intermediate-level waste in Sweden comes from the nuclear power plants and Studsvik. It is planned to dispose of this waste in repositories near to the repository that will be used for spent fuel. The environmental impact assessment procedure for the disposal of long-lived low- and intermediate-level waste is therefore included in the environmental impact assessment procedure for the disposal of spent fuel.

The objective of the AFA-1 project was to give authorities and waste producers in the Nordic countries background material for decision-making concerning the management and disposal of radioactive waste. The project was divided into three sub-projects dealing with:

- Waste characterisation
- Performance assessment for the repository near-field
- Environmental impact assessment

These directions of work were found to be of most common Nordic interest for the project period [Brodén 97]. They had also the advantage of being suitable for one project (see Figure 1.1).

Representatives from all Nordic countries have participated in each of the sub-projects. Most of the work has been performed in a broad group of experts. This has contributed to a better understanding of the waste situation in the different countries and has also made it possible to learn from each other. Furthermore, in some cases, it has contributed to common recommendations.

The results from the sub-projects are presented in the following three chapters (Chapters 2-4). Concluding remarks are presented in Chapter 5. Abbreviations, participants and funding are presented in appendices.

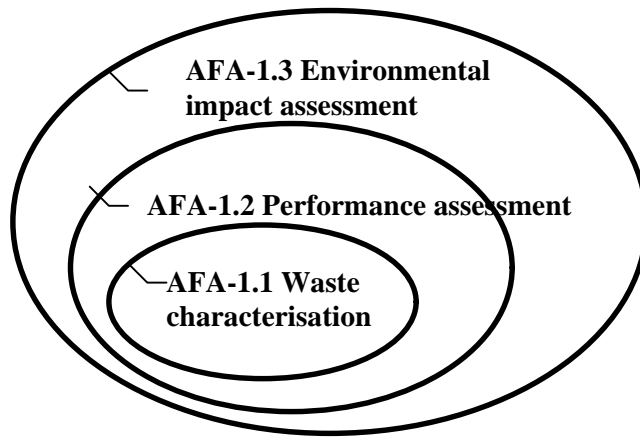


Figure 1.1 Waste characterisation is a part of a performance assessment, a performance assessment is a part of a safety assessment and results from a safety assessment are used in an environmental impact assessment.

2 Waste characterisation (AFA-1.1)

In this chapter, waste properties of importance for the long-term safety of disposal systems will be discussed. The waste is considered as conditioned waste i.e. waste ready for disposal. Unlike high level waste, the demands on waste containers for low- and intermediate-level waste are not particularly high. The safety of the system is ensured through the engineered and geological barriers of the disposal system and by the low activity contents in these types of waste.

2.1 Factors influencing the requirements on waste properties

A disposal facility should be placed and constructed in such a manner that the spread of toxic materials from the site is prevented or at least does not result in significant exposures of the population or other deleterious effects to the environment. Much can be done by selecting suitable sites and engineering solutions for the disposal facility, thereby determining the external conditions in which the waste packages are placed. It is the overall performance of the system which is important. Improved quality in one respect may therefore compensate for a less satisfactory quality in other respects. Some major parameters are:

- Depth of the facility (on- or near-surface, deep geological)
- Type of geological formation (crystalline rock, salt, clay)
- Barrier system (clay, concrete)
- Hydrology of the site (above or below groundwater, gradients, flow rates)
- Water chemistry
- Radioecology and general use of the area (food production)
- Seismic stability
- Climate
- Risk of human intrusion

Interactions between the waste and external conditions such as hydrology, groundwater composition, thermal gradients, geotechnical instability and the associated risk of crack formation in the barriers affect the behaviour of the inner part of a disposal facility. Such interactions are therefore important for the selection of the site and type of repository, and for the derived acceptance criteria for waste packages permitted to be disposed of in that particular facility:

- Upper limits for isotope-specific activity inventories in the facility as a whole are likely to be specified based on overall safety assessments for the system.

- Heat generation might be of concern with deep geological disposal of some types of medium-level waste and is an important feature for high-level waste.
- The contents of long-lived radioisotopes are especially important for near surface disposal. An overall limit for α -emitters might be specified for each waste package or as an average over a certain number of units. Limits for very long-lived β -emitters might also be required.
- Human intrusion by accident is a possibility in a forgotten repository and is an additional motivation for the above-mentioned limits. Intrusion might also be intentional if the waste represents a significant (future) resource value. This situation should be avoided.
- Handling of the units during emplacement in the repository might require limits on external radiation levels (γ and neutrons). The risk of escape of gaseous radionuclides should also be considered (^3H in water vapour, ^{14}C in CO_2 , CH_4 etc., and ^{222}Rn from ^{226}Ra are the important ones).

The waste packages should be compatible with the overall environment of the repository. This is only possible to a certain extent and a general degradation of the waste packages with time must be accepted. It is important that the often very slow degradation processes are understood so that reasonably correct long-term predictions can be made. If possible, the unavoidable changes in the system should not lead to significant releases outside the engineered barrier system.

Concrete or other cementitious materials are widely used as backfill, barrier or construction material in most engineered disposal facilities. The presence of cementitious materials has a profound influence on water chemistry in the system (high pH, high Na^+ , K^+ or Ca^{2+} concentrations) and documentation of the maintenance of such conditions over long periods should be included in the safety assessment for the facility.

2.1.1 Nuclide contents

The total amounts of radiotoxic components in the waste determine the potential hazard associated with a disposal facility. A measure for the potential hazard is obtained when the amounts are compared with the specific hazard associated with uptake of the various isotopes by the human organism. This is expressed by dose coefficients for intakes of radionuclides by workers as defined by ICRP [ICRP 68].

Due to decay, the potential hazard from radioisotopes diminishes with time, and it is therefore primarily the longer-lived radioisotopes which are important.

A list of radioisotopes and toxic materials assumed to be relevant for safety assessments in disposal facilities for low- and medium-level radioactive waste was given in a report for the sub-project AFA-1.1 [Brodén 97]. The list is partly taken from the PACOMA study [Marivot 91].

The following arguments for inclusion of a radioisotope on such a list can be put forward:

- Long half-life (arbitrarily fixed as >15 years)
- High mobility in barrier systems
- High radiotoxicity (low ALI values)
- High abundance in the waste
- Easy to measure, might serve as key nuclide
- External radiation
- Occurs in gas form (migration and handling problems)

Only radionuclides with a considerable half-life need to be considered in connection with waste disposal, but the decay rate should always be considered in connection with the risk of transport through the actual barrier systems. Only radioisotopes with a high mobility are likely to result in doses from reasonably well constructed repositories. Fortunately, the isotopes with high radiotoxicity and high abundance in the waste are also those with low mobility in engineered or geological barrier systems. The question of isotope abundance in the waste is dependent on the type of waste.

2.1.2 Chemical and physical stability of the waste

The following features or mechanisms related to waste properties may be important:

- **Leaching and stability of cemented waste products**
Some isotopes - notably the α -emitters - are normally retained very efficiently in the high pH environment in cementitious waste. This is not the case for Cs isotopes or (when CO₂ is absent) for ⁹⁰Sr. Permissible leach rates may have to be specified. Soluble salts are leached quite easily from cemented waste and may produce a transient pulse of contaminated solution with high ionic strength promoting the migration of radioisotopes. Reactions between cement and SO₄²⁻ and, under cold conditions, also Cl⁻ produces voluminous minerals and results in the swelling of some products. Combined with freeze/thaw or dry/wet cycling e.g. under intermediate storage,

the coherence of the product may be lost. This can also occur with cemented ion exchange resins.

- **Leaching and stability of bituminised waste products**

Leaching of radioisotopes from bituminised materials is dependent on waste loading, type of embedded waste (e.g. ion exchange resins, evaporator concentrates and whether co-precipitation is used to immobilise for example Cs and Sr), the remaining water content in the product, organic impurities, type of bitumen and type of leach solution.

The leaching phenomenon is at least partly due to water uptake and the formation of a relatively coarse interconnected system of cracks or voids filled with a water-based salt solution. Models of water uptake and the associated swelling of bituminized evaporator concentrates are available [Brunel 94].

- **Hygroscopic properties and swelling due to water uptake**

Water uptake into bituminized waste products leads to swelling and, if this is prevented, to generation of very considerable osmotic pressures. Some room for swelling inside the units should therefore be provided to diminish the risk of the external containers cracking

If a disposal facility is situated in the unsaturated zone above the groundwater table, the units will not be directly exposed to liquid water, but the surrounding air will be of a high humidity.

It has been shown [Brodersen 92] that cementitious materials containing soluble salts under such circumstances will absorb moisture and generate droplets or layers of contaminated strong salt solution on the surfaces. The same is the case for bituminized evaporator concentrates [Brunel 94] and, to some degree, also for bituminized ion exchange resins. The water condensation phenomena constitute a release mechanism in near- or above-surface repositories.

The volume changes associated with water uptake into conditioned waste materials may result in the cracking of external barriers.

- **Corrosion**

Corrosion of steel under anaerobic conditions and general corrosion of other metals, in particular aluminium at high pH, can produce quite large amounts of H₂ in disposal facilities.

- **Bio-degradation**
Gas generation (CO₂, CH₄ or H₂) by bio-degradation could be important, particularly for compressed solid waste with a content of organic materials. The gases may contribute to various transport mechanisms and especially carbon dioxide could be important from a chemical point of view.

Degradation by micro-organisms is not expected to contribute significantly to releases from bituminized waste although this is still an uncertain area. Micro-organisms may also contribute to the degradation of cement.

- **Irradiation and gas production**
External or internal irradiation is normally of minor concern with cemented low- and intermediate-level waste, but certain products containing shredded organic materials might swell and crumble. Some gases are produced and pH may decrease.

- **Organic complexants**
Elements, which normally have a low solubility and a high tendency of sorption e.g. Pu and Am, can change behaviour on contact with organic complexants.

The complexants originate from:

- Organic components in the waste
- Additives in the concrete used in the repository or in the waste solidification matrix
- Products from chemical, irradiation or biological degradation of components in the waste or in the surroundings

Organic complexants in the waste can typically be chemicals used for the decontamination of equipment or detergents. Degradation of cellulose in the waste can produce strong complexants at high pH.

2.1.3 Chemically toxic components

Radioactive waste may contain both organic and inorganic chemically toxic materials. The contents are highly dependent on the type of waste. In nuclear research waste, the most likely toxic materials are the metals Be and Cd, used for various purposes in nuclear physics, and Pb, widely used as shielding material. Hg may occur in research waste.

Most of the inorganic materials (e.g. heavy metals) can be considered as a minor problem because they tend to be converted into insoluble form. Behaviour of the organic components is more difficult to predict, but the content of organic chemotoxic components in the radioactive waste in the Nordic countries is normally small. Examples of these could be chlorinated aromatics, fungicides, pesticides and some organic metal-components. An effective countermeasure is to burn the waste, but that is not always possible.

2.2 Waste types, waste containers and conditioning methods

2.2.1 Denmark

Three types of radioactive waste are stored in interim facilities at Risø:

- Evaporator concentrate
- Compacted solid waste
- Non-conditioned solid waste

The evaporator plant has operated since 1959 treating low-level radioactive waste water from the nuclear facilities at Risø. A bituminisation plant has been used as final evaporator since 1970. Here the rest of the water is boiled off and the dry materials, mostly Na_2SO_4 and NaCl , are embedded in a bitumen matrix. The suspension is cast in standard units (see below). About 40 units with bituminized waste are produced annually. Sometimes ion-exchange resin is added during the bituminisation step. The drums used for casting bituminized evaporator concentrate may occasionally contain activated or contaminated metallic scrap.

The compaction system uses a low pressure hydraulic press installed in a glove-box. The waste consists of slightly contaminated material (paper, plastic, rubber, glass, aluminium, smoke detectors, etc.) received from laboratories at Risø and from hospitals, laboratories and other users of radioactive isotopes outside Risø. The waste is compacted in standard units. About 60-80 units with compacted waste are produced annually.

Long lived low- and intermediate- level radioactive solid waste are stored as non-conditioned in galvanised standard units or small containers of stainless steel. The waste is mainly aluminium items from the research reactor or waste from investigations of spent fuel at the Risø hot cells (closed in 1994).

The standard units used for bituminized waste and compacted solid waste consist of a 100 litre drum inside a 200 litre drum with a 5 cm layer of cement mortar in the annular space between the two drums. When the unit is filled the top is cast

with concrete and the drum is closed with a steel lid. This unit is also made from 200 litre drums of galvanised steel for intermediate-level waste.

Other waste containers used at Risø are:

- 280-litre drums of mild steel used as overpack for old standard units in bad condition
- Small containers of stainless steel used for long-lived low- and intermediate-level waste
- Stainless steel containers used for control rods from the DR3 reactor
- 200 or 280 litre drums of steel used for contaminated soil
- 200 or 280 litre drum of steel used for contaminated or activated metal scrap
- Steel boxes with a thick concrete layer used for filters from the hot cell
- Steel boxes used for mixed solid waste

Voids in some units may be filled with cement mortar to improve the stability.

2.2.2 Finland

Long-lived low-and intermediate-level radioactive waste in Finland is mainly produced at the two nuclear power plants in Loviisa and Olkiluoto, but a minor part is also produced in other industries, in hospitals and by universities, etc.

The nuclear power industry produces low-and intermediate-waste partly as reactor waste, which is produced during the normal operation of the plant, and decommissioning waste, which is mainly produced during the demolition of a plant.

The reactor waste is divided into two categories: process waste, consisting of intermediate-level active filters and ion exchangers, and maintenance waste, consisting of low-level materials like working clothes, tools, machine parts, and building materials.

Decommissioning waste is created during the dismantling of nuclear power plants. It is divided into 1) activated, 2) contaminated, and 3) very low-level material. The activated material consists of core shrouds, steam dryers, etc., while the contaminated waste consists of, for example, pipes, valves, etc. The very low-level material is mainly concrete.

The low-and intermediate-levelwaste produced outside the nuclear power plants, i.e. the waste produced by industry, and at universities, hospitals etc., consists after treatment, only of dry material and emanates mainly from various instruments for measuring thickness, density, etc. in industry. A minor part of the waste consists of

contaminated metal scrap, radioactive standards, and minor objects containing radioactive paint (e.g. compasses and emergency exit signs). Liquid waste is evaporated before being transported for storage. Minor objects, like compasses, are packed into 200-litre steel drums, or small steel or lead packages. Large metal objects, for example parts from cobalt therapy instruments used at hospitals and contaminated metals from industry, are stored without packing on floors or shelves. The annual increase in waste volume is about 2 m³.

Radioactive waste is also produced at VTT during the operation of the 250 kW Triga MK II research reactor. Spent ion-exchange resins are transferred into the plastic drums that are used for the delivery. The drums containing resins are placed in interim storage at VTT. Other types of radioactive waste produced at VTT are mainly packed into 200-litre steel drums.

The container types used for the storage of low-and intermediate-level radioactive waste are mainly:

- plastic sacks
- bales
- 200-litre steel drums
- 1.3 m³ or 1.4 m³ steel boxes
- concrete boxes for 12 or 16 steel drums

Combinations of the various container types are also used, like steel drums in steel boxes or plastic sacks in steel drums. Recently, a bailing device and a corresponding measuring system have been introduced. The use of large bales, consisting of several plastic sacks, reduces the need for storage space as well as the time needed for measurements. Gamma spectroscopic measurement for each individual sack is not needed anymore.

All low- and intermediate-level active waste that is transported to the Olkiluoto VLJ final storage is finally placed in concrete boxes.

Waste from the Loviisa NPP is packed in 200-litre steel drums. It is divided into four categories depending on whether it is burnable and/or compressible. Soft material is compressed. Large metal components are handled individually. In some cases they are cleaned in order to render them exempt from regulatory control if possible. Small radioactive objects are packed in steel drums, while larger radioactive metal components are currently stored at the plant.

Ion-exchange resins and other intermediate-level waste in Loviisa are placed in storage tanks. The volume of ion-exchange resins used for the purification of the primary coolant water was 284 m³ at the end of 1994. It is stored under water in

steel tanks with a volume of 300 m³. There has not yet been any final treatment of the resins. One possibility is to solidify the ion-exchange resin with cement.

All active water in Loviisa NPP is evaporated. The annual production of evaporator waste is about 100 m³. However, the stored volume of evaporator waste, has not increased during 10 years because about 100 m³ of old decay concentrate has been released each year. The released activity is about 10 GBq (ca. 1% of the release limits).

In the Olkiluoto NPP, the low- and intermediate-level active waste is produced during maintenance and repair work and in the filtering system of the process water. The low-level waste is compacted in 200-litre steel drums. Metal scrap is packed without treatment into concrete boxes. The intermediate-level ion-exchange resins are bituminized into 200-litre steel drums. After packing, the waste after packing is stored in storage buildings for CL1 and CL2.

2.2.3 Iceland

There are nonuclear installations in Iceland, and radioactive waste is thus generated in very small quantities in medicine, research and industry. Except for the sources from a few discharged smoke detectors stored at the Icelandic Radiation Protection Institute, no long-lived radioactive material can now be classified as waste, since all of the material is still in the possession of its owners. A few sources are not in use any longer, and some instruments containing long-lived radioactive sources have been returned to their foreign producers.

In terms of radioactivity, nearly all material (79 TBq) is contained in one source, a ⁶⁰Co teletherapy unit stored in the basement of the Radiation Therapy Department at the Icelandic State Hospital. The rest (0.19 TBq), is distributed over approximately 200 different sources, e. g. 18 ²²⁶Ra needles with a total activity of 0.02 TBq. All sources are either metal or metal encapsulated.

Since actually no long-lived radioactive waste exists in Iceland, no conditioning methods or waste containers have been developed.

2.2.4 Norway

Low- and intermediate-level radioactive waste from nuclear installations at the Institute for Energy Technology in Norway (IFE) include waste from plant operations such as ion-exchange resins from the two Norwegian research reactors, contaminated metal scrap, liquid and semi-liquid waste and solid laboratory waste, such as plastic, glass and paper. Radioactive waste from other sources include liquid waste from medical use and research activities, encapsulated sources from research activities and industrial processes, signs and instruments containing trit-

ium, scale from oil production and smoke detectors and other consumer goods containing radioactive sources. The total production of radioactive waste excluding scale is approximately 110 - 120 waste containers per year.

Solid waste is shredded and compressed or only compressed. Larger items are dismantled or cut into pieces. Inflammable materials containing nuclides of half-lives shorter than one year are stored for 1 - 2 years before incineration. After incineration ash and soot are treated as radioactive waste. Encapsulated sources are normally not dismantled from their shielding houses but wasted as a unit. Smoke detectors are dismantled and the small ^{241}Am source is treated as radioactive waste. Other metal waste is cut into pieces and imbedded in concrete inside waste containers.

The most commonly used waste containers for solid and metal waste are 200 litre steel drums with concrete and sometimes lead-shielded walls. Depending on the dose rate, the waste is stored in inner drums of 30 litre, 60 litre or 100 litres, giving a shielding thickness of the walls of the 200 litre container. Recently, rectangular steel cases or concrete cases have been used for larger metal parts. Steel cases are used for low radioactive waste while concrete cases with 120 cm or 20 cm walls are used for waste giving higher dose rates.

Mud from waste storage tanks is absorbed in vermiculite and stored in the above mentioned drums. Low-level radioactive ion-exchange resins are placed in similar units after excess water has been removed. Intermediate level radioactive ion-exchange resins are mixed with concrete inside steel containers with 2 - 3 cm lead shielding and placed inside 200 litre steel drums. This process is performed remotely.

Liquid waste concentrate is obtained from the evaporation treatment of low-level liquid waste. Organic liquids are mixed with emulsifiers and mixed in portions of 10 litres in 120 - 130 litres of aqueous waste. Liquid waste in portions of 120 - 130 litres is put into polyethylene-lined barrels of 200 litres. The liquid is mixed with cement and other additives or by an absorbent and, thereby, transformed into solid form.

Scale is stored in steel cases of 860 litres with an airtight lid.

2.2.5 Sweden

Long-lived low- and intermediate-level radioactive waste is produced in relatively minor quantities in Sweden [SKB 97]. The main sources are power reactor components and waste from research.

The used reactor components produced so far are kept at the power plants and at CLAB (Central Interim Storage Facility for Spent Fuel). The packaging intended for the waste comprises containers made of reinforced concrete (outside dimensions, 1.2 m x 1.2 m x 4.8 m) with inner stainless steel cassettes. The storage volume for waste in each cassette is 2 m³.

The research waste is collected, conditioned and stored at Studsvik. It consists of waste from research carried out at Studsvik and of waste collected from other producers of radioactive materials in Sweden e. g. industry, universities and hospitals. The waste packages produced at Studsvik can be classified into the following main categories:

- Concrete containers with intermediate level waste
- Concrete containers with plutonium waste
- Steel drums with ashes
- Steel drums with refuse and scrap
- Old concrete boxes with plutonium waste

The intermediate-level waste in the first category comes from a concrete plug hole store at Studsvik and also directly from different facilities at Studsvik. It consists of solid material such as activated components, contaminated components and radiation sources. The waste is sorted, cut and compacted and then placed into special 80-litre steel drums with double lids. The drums are placed inside reinforced cubical concrete containers (outside dimensions 1.2 m x 1.2 m x 1.2 m). Each container has five holes prepared for five drums. For the waste with the highest dose rates, lead inner containers are used inside the drums. By the end of 1996, the number of concrete containers with long-lived waste was about 240.

Concrete containers with inner double lid drums are also used for solid waste containing plutonium. Two types of inner containers inside the double lid drums are used for the waste: stainless steel inner containers with thick walls and stainless steel inner containers with thin walls. The raw waste material consists of plutonium contaminated solid material such as glove boxes, tools, instruments, consumable supplies and laboratory outfits. The number of concrete containers with plutonium waste produced was by the end of 1996 about 70.

Liquid waste from the facilities at Studsvik (mainly from the test reactor, R2) is treated in a treatment plant. A widely adopted technique is used for the separation of Cs-isotopes for the liquid. The technique is based on co-precipitation with ferrocyanides. After separation in a centrifuge, the sludge is set in cement to produce solid blocks in steel drums with mixers. By year-end 1996, the number of steel drums with solidified long-lived waste produced was about 440.

Part of the ashes from the Studsvik incineration plant has to be considered as long-lived waste. The ashes are put in 100 litre steel drums. These drums are placed in 200-litre steel drums and the space between the inner and outer drums is filled with concrete. By the end of 1996, the number of drums with long-lived ashes produced was about 270.

The same packaging system is used for refuse and scrap. Part of the packages with this type of waste also have to be considered as long-lived. The raw waste material consists e. g. of contaminated tools, instruments, consumable supplies, laboratory outfits, clothes, plastic bags, metal parts, construction materials and radiation sources. The number of drums with long-lived refuse and scrap produced was about 220 by year-end 1996.

In total, about 30 old concrete boxes with plutonium waste are stored at Studsvik. Each concrete box consists of a filled glove box surrounded by a layer of reinforced concrete.

2.3 Measuring and estimation of activity content

2.3.1 Methods used in Denmark

All standard units containing lowlevel compacted solid waste or bituminized evaporator waste are measured by a NaI-detector before being stored in the interim facility for low-level waste. The activity of ^{60}Co and ^{137}Cs is calculated from four measurements on each unit, assuming a homogeneous distribution supplemented by three point sources near the wall in the midplane of the unit.

In case of disused sources, external γ -radiation is used together with exposure rate constants to estimate the activity.

Beta-counting of solution samples evaporated on Al-discs and γ spectrometry using a Ge-detector are used for control analyses during the operation of the waste treatment plant. The spectrometer is calibrated to measure samples of different geometry assuming that the activity in the samples is homogeneously distributed.

The plutonium content is measured in samples of evaporator concentrate by using α -spectrometry. Contents of $^{239}\text{Pu}+^{240}\text{Pu}$ and ^{238}Pu are obtained using ^{242}Pu as the internal standard. The measurement is time-consuming because of the extensive preparation before the sample can be counted. Correlations between contents of Pu isotopes and ^{137}Cs are calculated and compared with expected values for ^{137}Cs used as the key nuclide for waste of the type considered.

2.3.2 Methods used in Finland

Sufficient information about the activity in radioactive waste can often be obtained from dose rate measurements. They are usually calibrated with a ^{60}Co or ^{137}Cs standard. However, dose rate measurements only yield preliminary information for handling the waste, and in most cases, γ spectroscopic measurements are needed in order to obtain more exact information for individual nuclides. Measurement campaigns of waste packages are made at the nuclear power plants a few times per year.

The scanning geometries used in the measurements of the activities in steel drums are, in principle, the same used at Loviisa, Olkiluoto and VTT. The main differences are found in the calibration procedures and in the use of standards.

At Loviisa, a drum is used in which the calibration sources can be placed in various places. The drum contains representative waste, i.e. paper, clothing, plastic etc. Three sources are used: ^{133}Ba , ^{137}Cs and ^{60}Co .

At Olkiluoto, there are several calibration sources used for different types of waste packages:

- The scanning measurements of plastic sacks are calibrated with an ^{152}Eu point source.
- The ^{152}Eu standard is mixed with inactive waste and is used for measuring low-level steel drums.
- The activity content in a drum with intermediate-level active ion-exchange resin is measured in small portions in the laboratory. The measured resin is then bituminized and used as a standard.

At VTT, point sources are used which are fixed to the wall of the drum.

Rotating drums and other packed waste is scanned in order to obtain representative values for the total activities. In cases when the dose rate is too high, scanning is not possible because the dead time for the measuring equipment increases (the equipment gets blocked). The measurements are then made from some distance, in the highest dose direction, which is a conservative alternative to scanning.

The power plants have extensive nuclear waste accounting systems, which can be used for updating scaling factors for various nuclides and waste categories at different times. This makes it possible to adjust total activity estimates for waste in storage if, for example, it is found that the previously used scaling factors were erroneous during a certain period.

The activity in the decommissioning waste is estimated by using both numerical calculations and experimental data from contamination measurements made at VTT.

2.3.3 Methods used in Iceland

A prerequisite for an estimate of the activity content of a radioactive source is a knowledge of its isotope composition. This is well documented for nearly all existing sources in Iceland. In the case of an unknown composition, it will most probably emit γ -radiation and can be identified by γ -spectrometry. In the improbable case of an exclusive α - or β -emitting isotope, it can be discovered with special detectors, and other methods can be used to identify and measure the activity content.

The activity of most radioactive sources is documented, and measurements are therefore unnecessary. If the activity of a known γ -emitting isotope needs to be measured, either of the following two methods is used. If the material consists of only a single isotope or the relative composition of the different isotopes is known, the exposure rate at a specific distance from the source is measured, and the activity estimated using the isotope's γ -ray constant. In the case of several isotopes with unknown relative composition, measurements are made with a Ge-spectrometer at a specific distance. A ^{152}Eu point standard is used for this calibration of the spectrometer.

2.3.4 Methods used in Norway

Radioactive waste delivered to the Waste Treatment Plant at IFE, Kjeller, must be accompanied by information on the nuclide content and the activity levels. Because this information is often incomplete or missing, it is necessary to perform analysis and measurements.

After the evaporation of small samples of liquid waste on metal disks, the total α - and β -activity in liquid waste is measured. The nuclide content and activity levels are calculated based on available information from waste producers or from radiochemical analysis or γ -spectroscopy.

In radiochemical analysis of liquid or semi-liquid waste separation and plating/filtering is followed by α -spectroscopy or β -spectroscopy. The quality of the radiochemical methods is tested in Nordic and international calibration tests. The measuring equipment is calibrated by using standard calibration sources.

Samples of waste are often analysed by γ -spectroscopy in order to obtain information of the nuclide composition and activity levels. The measuring equipment is based on Ge-detector systems and NaI-detector systems. Calibration of these sys-

tems is performed by using standard calibration sources for different measuring geometries in use.

After waste processing and conditioning in waste containers, dose rate measurements are performed outside the containers. Based on available information on the nuclide composition supplied by the waste producer or gained by radiochemical analysis and/or γ -spectroscopy, the activity level for each nuclide inside the container is calculated. For ion-exchange resins from the two research reactors in Norway, the nuclide composition is well known. Calculation of the activity levels for nuclides are thus based on the measurements of ^{60}Co .

2.3.5 Methods used in Sweden

Gamma spectrometric measurements are performed on 80-litre double lid steel drums with intermediate-level waste at Studsvik (see Section 2.2.5) before the drums are placed in concrete containers [Sundgren 89]. The results from the measurements are registered in a waste register. The measuring method gives generally less than a $\pm 40\%$ deviation or, in the worst case, for very inhomogeneous waste $\pm 70\%$ deviation [Sundgren 94].

The plutonium activity in concrete containers with plutonium waste are also analysed during the waste treatment procedure. Two different methods are used. Small sources and plastic bags with soft waste are analysed by γ spectrometry (measurement of 129 keV from ^{239}Pu). Hard waste components are analysed indirectly by total α analyses of smear samples. Estimations of plutonium contents, based on the analyses, are stored in the waste register.

Representative samples from each batch with precipitation sludge are taken out before the sludge is set in cement. The samples are γ spectrometric and α spectrometric analyses. The results from the measurements are stored in the waste register.

For drums with ashes and refuse/scrap produced in 1980 or later, γ spectrometric measurements are performed on the 100 litre inner drums. The results from the measurements are stored in the waste register. For practical reasons, the plutonium and americium quantities in the waste are estimated based on the ^{137}Cs activity. 1 % and 4 % of the Cs-activity at the time of measuring, are assumed to correspond to activity values for $^{239+240}\text{Pu}$ and ^{241}Am , respectively [Andersson 90].

The total plutonium content in the boxes with plutonium waste are estimated based on information from transport documents.

2.3.6 Inventory of new available methods

Measuring systems and equipment for measurements and analyses of nuclide content and activity levels in unprocessed and processed waste are commercially available from several producers and suppliers. A summary of various available measuring systems and equipment from companies, suppliers and organisations are given in Appendix 2 in the AFA-1.1-report. A series of systems based on non-invasive methods for measurements of the nuclide content and activity levels of unprocessed waste and of waste inside waste containers are presented. The systems can be mobile or stationary and use γ spectroscopy, active or passive neutron-based systems for measurements of the amount of α -emitting transuranic nuclides ($Z > 92$) in waste packages or a combination of these methods.

Systems based on γ -spectroscopy for the assessment of nuclide composition and contamination levels and activity levels in unprocessed waste and in waste drums are available. The detection limits depend on the density of the material but typically range from 100 Bq to 500 Bq for γ -emitting nuclides with energies from 300 keV to 1500 keV. Matrix correction is available for most of the systems.

Systems using high resolution γ spectroscopy are available for plutonium and uranium measurements. The detection limit is typically $< 3\%$ ^{235}U .

Systems for the assessment of ^{235}U and plutonium content in waste or inside waste drums using passive or active neutron coincidence technique are available. Some systems combine the active and passive method by the measurement of delayed neutrons after the active source has been removed. Some systems also combine neutron coincidence techniques with high resolution γ -spectroscopy. Calculations of the total content of plutonium in waste are made from measurements of one plutonium isotope and knowledge of the isotope composition. Typical detection limits for drum monitoring systems using a 1000 second count time are:

- Active systems measuring thermal neutrons: $< 1\text{ mg } ^{239}\text{Pu}$
- Passive systems: $< 5\text{ mg } ^{240}\text{Pu}$ or $< 20\text{ mg Pu-total}$

Systems for remotely operated measurements of spatial distribution of contamination and radioactive materials in unknown or high dose fields are available. Imaging systems based on radiography or tomographic principles are currently being developed. They give information about density distribution and location of radiation sources inside units with conditioned waste, and are particularly of interest with respect inhomogeneous waste.

2.3.7 Discussion and recommendations

Measurement of activity contents in radioactive waste is primarily carried out to establish a credible inventory of radioisotopes for use in safety assessments during the handling, storage and disposal of the waste. Comprehensive measurements of all relevant radioisotopes are costly, difficult and may even be counterproductive as far as safety is concerned, resulting in a radioactive dose to operators and the generation of secondary waste. The selection of a suitable level of documentation for activity contents should therefore take aspects like the following into account:

- Analyses of solid samples from waste streams before conditioning or obtained by destructive methods from selected waste packages are only meaningful if they can be correlated in a statistically significant manner with the activity inventory in a large number of waste packages.
- Information about isotopes likely to be present in the waste is often given by the origin of the material.
- Analyses and other information about the waste should be stored in such a manner that isotope inventories (with the associated uncertainties) can be generated and retrieved as needed for safety assessments.
- Obviously the quality of the inventory information does not need to be the same for small and large amounts of a radioisotope or for short-lived compared with long-lived isotopes.

In the Nordic countries, like elsewhere, the most important measurement method is γ scanning of the final conditioned waste packages. This primarily provides information about contents of ^{60}Co and ^{137}Cs with uncertainties which depend upon the degree of homogeneity of the waste. Correct calibration of the equipment and correct data evaluation are of course necessary. Intercalibrations between the various scanning systems are useful in this context.

In Denmark, a system based on a NaI-detector was selected due to the reasonably low cost and ease of operation. However, the low degree of resolution precludes the detection of small amounts of other isotopes besides the two mentioned above, and the system is best suited to relatively old waste with no interference from short-lived γ -emitters.

The systems used in the other Nordic countries are based on high resolution Ge detectors as is also the case in commercially available drum scanning equipment.

This eliminates most interference problems but not uncertainties due to the inhomogeneous distribution of activity and shielding within the drums.

Improvements in this area are possible ranging from mean density estimates obtained from the attenuation of photons passing through the drum from an external ^{60}Co source (as practised in Finland) to complex radiographic and tomographic imaging techniques combining multi-detector systems with complicated movements of the waste package.

The radiographic and tomographic equipment is costly, time-consuming to operate and partly still on the development stage. It is therefore best suited for the characterisation of 'difficult' units or as a quality check on a limited number of units selected on a statistical basis for detailed control. Systems of this type are not available in the Nordic countries, and it is questionable whether improved information about relatively few waste packages can motivate the necessary investments if they were to be used in the future.

Laboratory α -, β - and γ -analyses of samples from unconditioned waste streams are carried out routinely in most of the Nordic countries at nuclear power plants as well as research centres. The results can be and are, to some degree, utilised to estimate isotope inventories in final waste products. The method works well with homogeneous waste, preferably such waste that is produced by reasonably standardised methods so that representative samples can be obtained. Conversion of the analyses to isotope inventories will be waste stream specific, may require rather careful considerations and will, in itself, introduce uncertainties.

The contents of ^{137}Cs or ^{60}Co measured by γ -scanning are often correlated with contents of long-lived radioisotopes which are difficult to measure because they are pure α -, β - or weak γ -emitters. The correlations are strongly dependent on waste origin and should be checked using the above-mentioned methods of laboratory analyses. Correction for decay or in-growth of relatively short-lived isotopes will be necessary. Taking this into account, correlation or scaling factors can give reasonable estimates of the contents of β -emitters, such as ^{63}Ni , ^{90}Sr , ^{99}Tc and of α -emitters, such as the plutonium isotopes, etc. However, increased uncertainty with respect to inventories determined in this indirect manner must be accepted. The method is used in most of the Nordic countries but with different correlation factors and different key nuclides according to the particular waste under consideration. Further work on the documentation of the correlations may be required.

Some waste may neither be homogeneous nor contain materials with well-established correlations between external γ radiation and the isotope contents. If from the origin or other administrative information it appears likely that the waste contains significant amounts of α -emitters in form of transuranic or fissile mate-

rial, it may be necessary to check this using one of the non-destructive methods based on neutron measurements. Equipment for this purpose is not available in the Nordic countries. Where needed, estimates of transuranic contents are obtained in other ways (safeguards information, measurement of weak photons from loosely packed waste (Sweden), measurements of surface contamination before packaging). The sensitivity of the present neutron-based methods is not particularly high and this limits their applicability in separating waste which has to go to deep geological disposal and waste which may be disposed of in near-surface facilities.

In general, it is concluded that the question about the documentation of radioisotope inventories and requirements with respect to the underlying analytical and calculation methods must be seen together with the intended use of the information. Collecting unnecessary information might be a serious waste of resources.

2.4 Estimation of general composition

2.4.1 Methods used in Denmark

A rough estimate of the general composition of the waste stored at Risø is available but no detailed analyses are carried out.

A sample from each period with the bituminisation plant is kept for possible examinations of the product in the future.

Origin and other information about solid waste is stored in a database in connection with the compaction of such waste in drums.

2.4.2 Methods used in Finland

General requirements for waste package acceptance in Finland are included in a regulatory guide. This guide includes, among other things, requirements on potential adverse characteristics (e.g. flammability, swelling capacity, gas generation potential, concentrations of chemically aggressive substances) [Ruokola 97].

Chemical analyses carried out on ion-exchange resin at the nuclear power plants aim mainly at determining the pH and the boric acid concentration.

The waste in the STUK interim storage facility consists of solid materials, with the exception of a small amount of Kr gas. The material is well documented and, for every nuclide, there is information about the chemical form in which it occurs.

2.4.3 Methods used in Iceland

Information about chemical contents is based on information from the producers of the radioactive source. Since nearly all of the radioactive material is in the form of sealed sources, chemical analysis will not be carried out.

2.4.4 Methods used in Norway

Radioactive waste delivered to the Waste Treatment Plant at IFE, Kjeller, shall be accompanied by information on the chemical composition. No chemical analysis is performed at IFE.

2.4.5 Methods used in Sweden

The chemical composition of the waste in the concrete containers with intermediate level waste at Studsvik is estimated by sorting and weighing during the treatment procedure. The same method is applied for plutonium waste in concrete containers.

The chemical composition of the content in the drums with solidified sludge can be estimated based on the quantities of chemicals used in the precipitation process.

2.4.6 Discussion and recommendations

Some information on the general waste composition is required in a safety analysis for a repository. However, it is normally sufficient with rough estimations on:

- metals (give rise to gas evolution from corrosion)
- complexing agents (influence the transport of radionuclides from a repository)
- toxic substances

The estimations should include information about the whole waste packages, i.e. information about both the waste and the packaging. Information from the primary waste producers can then be used in combination with simple analyses (e.g. weighting) and observation during the waste treatment processes.

3 Performance assessment (AFA-1.2)

3.1 Waste disposal systems in the Nordic countries

The practice and planning for disposal of radioactive waste varies widely within 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.

3.1.1 Denmark

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

Most of the stored waste is LLW and ILW but some is α -contaminated. Risø plans to store the waste for some 30 to 50 years and disposal will first take place in connection with the future complete decommissioning of the nuclear facilities at the research centre.

Short-lived LLW and ILW can be disposed of using relatively uncomplicated methods in near-surface facilities. Long-lived α -emitting elements should not be present in more than trace quantities. In Denmark the nuclear authorities have not yet specified limits for the content of long-lived α -emitters in waste which is to be disposed of in a near-surface facility.

Some preliminary design work for disposal systems for LLW and ILW in Denmark has been carried out [Brodersen 86]. The facilities considered can all be regarded as examples of advanced types of near-surface burial systems, and a smaller version of a facility like "Centre de l'Aube" in France could be a possible concept for the future disposal of Danish LLW and ILW.

The facility consisting of square boxes with walls and a bottom of about 1 m-thick reinforced concrete is supposed to be constructed in unconsolidated geological formations such as clay or sand. The facility should probably be positioned above groundwater level. Questions concerning hydrology are discussed in a generic way in [Brodersen 96].

Standard units containing LLW- and ILW (200 litre steel drums) are placed in position by travelling crane. The units are stacked on top of each other in 5 layers. After a square box is filled, the crevices between the units are backfilled with a suitable injection concrete. A 1 m concrete lid is then cast and the soil layer distributed on top of the construction.

It is estimated that the total volume needed for the disposal of LLW and ILW in Denmark will be less than 10.000 m³ including waste from the dismantling of the nuclear research facilities.

3.1.2 Finland

About 27 % of all electricity produced in Finland was generated by nuclear power in 1996. 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), initially made contractual arrangements for the entire fuel cycle service from the former USSR, including the 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. One consequence of the amendment of the Nuclear Energy Act, inter alia, is that IVO has to implement the same principles and time schedule as TVO in the management of spent fuel after 1996, when it will no longer be allowed to return spent fuel to Russia. The 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 1850 tU of BWR fuel from Olkiluoto and 760 tU of PWR fuel from Loviisa. The mission of the new company is the disposal of spent fuel.

Conditioning and storage of low- and intermediate-level waste 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.

Presently - according to the amended Nuclear Energy Act - the management of all nuclear waste relies on a domestic solution. Most of the wastes arise from the operation and decommissioning of the four power reactors in Finland. Limited amounts of radioactive waste arising from research activities as well from hospitals and industry were previously stored in an interim storage facility operated by the Radiation and Nuclear Safety Authority (STUK) on the island of Santahamina in Helsinki. These wastes have recently been transferred to Olkiluoto and they are further stored and ultimately disposed of in the VLJ Repository at Olkiluoto.

The construction of the repository for the low- and intermediate-level 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 the operation of the low-level part of the repository is planned to be started in 1998.

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

The plans for the decommissioning of the Finnish NPPs are updated every five years. The latest plans were published at the end of 1993 and the updating included the disposal plans and preliminary safety analyses for the decommissioning waste of the Olkiluoto nuclear power plant. According to the new plan, the existing VLJ repository for low- and intermediate-level operating waste will be extended with three new silos at the depth of 60 - 100 m. Besides dismantling waste, activated metal components, except fuel boxes, accumulated during the operation of the reactors will also 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 the repository will be sealed in around the year 2055. Similar plans also exist for the Loviisa repository for the expansion of the facility to enable the disposal of decommissioning wastes as well.

3.1.3 Iceland

No specific radioactive waste disposal plans have so far been developed in Iceland. The country is situated on the Mid-Atlantic Ridge, one of the most active volcanic and seismic regions in the world, and it is anticipated that a satisfactory disposal concept may be difficult to find.

The prospect of finding a suitable place for situating an engineered repository in hard rock, taking among others also the very high groundwater flow and temperature gradient into account, has been studied [Helgason 96]. Two possible regions were identified, a palagonite rock formation in the northeast, and some large intrusions in the southeast. The former consists of zeolitized palagonite tuff, hydrothermally altered with low permeability. A favourable characteristic of zeolites is their ability to absorb large ions such as cesium and strontium.

3.1.4 Norway

The Institute for Energy Technology (IFE) is the operator of the only facility for the treatment and storage of low- and intermediate-level radioactive waste in Norway and therefore all such waste is collected, treated and stored by IFE. Up to the end of 1996, 2260 drums and other containers were stored in storage buildings at the premises at IFE, Kjeller. In addition, approximately 1000 drums of low- and intermediate-level radioactive waste are deposited in a near-surface repository on these premises.

In accordance with a decision made by the Norwegian government in 1994, a combined storage and disposal facility for low- and intermediate-level radioactive waste is under construction in Himdalen in Aurskog-Høland municipality in Norway. The Directorate of Public Construction and Property in Norway is responsible for the construction and will be the owner of the facility. The Institute for Energy Technology (IFE) will be the operator of the storage/repository.

The facility is built in hard rock as a near-surface rock cavern facility, with 50 metres of rock covering, located in a small hill. It will be accessible through a tunnel. According to the parliamentary resolution, buried plutonium-bearing waste containers shall be placed in the storage part awaiting a future decision regarding how this waste must be disposed of. The waste that can be disposed of will be placed in concrete structures (sarcophagii). The Norwegian government has decided that the waste drums in the near-surface repository at IFE, Kjeller shall be retrieved and placed in the new storage/repository in Himdalen.

The Himdalen facility will consist of four rock caverns, three for the disposal of waste and one for storage of plutonium bearing waste. The caverns are installed at right angles at the end of a 138 metre-long entrance tunnel. The outermost cavern will be used as a storage area. The total capacity of each cavern will be 2 500 drums giving a total capacity of 10 000 drums for this facility. In the repository part, the waste containers will be imbedded in concrete sarcophagii with watertight roofs. In the storage area, waste containers will be placed inside concrete bunks.

Building of the new facility started on 4 March 1997 and will, according to the plans, be completed in February/March 1998. The facility will be in operation up to the year 2030 when it is anticipated that the facility will be filled to capacity. In the year 2030, a decision will be made regarding whether the storage part containing plutonium bearing waste should be transformed into a repository or whether this waste should be retrieved. The repository will then be closed but will be submitted to institutional control for a period of 300 - 500 years.

High-level waste in Norway consists of spent reactor fuel from the two research reactors. Spent fuel elements are placed in storage pits at IFE, Kjeller and IFE,

Halden. There is still no shortage of storage capacity for spent fuel in Norway and no decision has been taken on the disposal or future management of spent reactor fuel.

3.1.5 Sweden

The nuclear power programme of Sweden consists of 12 nuclear reactors with a combined capacity of 10 000 MW net electric power. The nuclear power plants generated about 52 % of the total Swedish electric energy produced in 1996.

The Swedish nuclear fuel and waste management company, SKB (SKB - Svensk kärnbränslehantering AB) operates systems and facilities for the management and final disposal of spent nuclear fuel and other radioactive waste from the Swedish nuclear power plants. A complete system has been planned for the management of radioactive waste from the Swedish nuclear power plants and from Studsvik.

The Swedish final repository for radioactive operational waste, SFR, was taken into operation in 1988. It is a repository for low- and intermediate-level radioactive waste, built in the bedrock under the Baltic Sea close to Forsmark nuclear power plant. A 50-metre layer of rock covers the repository caverns under the seabed. The first stage of SFR, which is in operation, includes buildings on the ground level, tunnels, operating buildings and disposal caverns for 60 000 m³ of waste. A second stage for approximately 30 000 m³ is planned to be built and commissioned after the year 2000. By the end of 1996, the amount of waste disposed of was about 21 000 m³ [SKB 97]. The waste materials are conditioned at the power plants, at the Central Interim Storage Facility for Spent Nuclear Fuel, CLAB, or at Studsvik.

SFL is planned to be situated at a depth of about 500 m in crystalline bedrock. It will consist of a repository intended for encapsulated spent fuel SFL 2, and a repository intended for other long-lived waste, SFL 3-5. The two repositories will be built at the same depth, but separated horizontally by about one kilometre. The original plans for SFL also included a repository, SFL 1, intended for vitrified spent fuel. However, SFL 1 will not be built.

The SFL 3 will be designed for long-lived low and intermediate-level waste from Studsvik and operational waste from CLAB and an encapsulation plant. The SFL 4 will be designed for decommissioning waste from CLAB and the encapsulation plant. The SFL 5 will be designed for the disposal of concrete containers containing reactor core components and internal parts. Strictly taken, not all of the waste forms destined for SFL 3-5 fall into the category of long-lived waste. In fact, only the waste that come from Studsvik, the core components and the reactor internals are long-lived. Operational waste and later decommissioning waste from CLAB and the encapsulation plant could, in principle, be disposed of in SFR. However,

SFL 3-5 is intended to receive all low-and intermediate-level waste that arises during the post-closure period of SFR [Lindgren 94].

3.2 Performance assessment methodologies

3.2.1 Role and scope of performance and safety analyses

Performance and safety analyses are required during various phases of a project to develop and construct a facility for the disposal of radioactive wastes [NEA 96]. 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 the extent of the potential hazards involved determine the role and scope of analyses required. In the 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 intermediate-level wastes and particularly those including a significant amount of longer-lived radionuclides - increasingly detailed and concept- and site-specific performance assessments 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 of the authorities and to receive a permit for the 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 in order for the application to receive a licence to commission and operate the repository.

Furthermore, the extent and type of performance and safety analyses is dependent on the purpose for which they are carried out and on the organisation conducting the studies. For example, the regulatory body may consider that an independent performance assessment 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, in order 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.

3.2.2 Choice of employed methodology

How a performance assessment needs to be undertaken and what type of methodology is required are dependent on the regulatory requirements, safety or performance indicators (i.e. release rates, individual/collective doses, fluxes to the biosphere etc.) considered, the target audience and the timescales necessary to be covered for the considered repository [Savage 95]. The safety requirements vary from country to country, but within the Nordic countries and in the framework of the NKS safety research programme, the aim is to employ, 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 decisionmakers and general public.

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

3.2.3 Methodologies for different stages

There are various approaches and techniques for carrying out the analysis of the performance of repository sub-systems. Regardless of the detailed methodologies employed, it is important to first carefully review the 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 assessment would then be the prediction of consequences in selected scenarios that take into account the identified key safety issues. The major components of a full-scope performance assessment may include the following aspects:

- Development and choice of important scenarios
- Development of conceptual models for the sub-systems 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 sub-systems 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 engineered 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.

Scenario development

The performance assessment requires, as a starting point, a number of assumed courses of events or scenarios through which one wishes to analyse the performance of the considered sub-systems 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 [NEA 96]. 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 intermediate-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 [Stephens 95] 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 their benefits and drawbacks has been presented in [Eng 94]. The safety and performance assessments 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 their subsequent migration and transport in geosphere and biosphere.

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 the Process System (PS), which according to the definition in [Eng 94], 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 [Eng 94].

In system analysis of e.g. nuclear power plants, the event and fault tree analyses were extensively applied. In waste disposal performance assessment, the fault tree method was found to be less suitable as it is primarily intended to be employed in cases where the events and processes are well-known and supported by extensive statistical data. The performance assessment of repositories can be supported by a comprehensive understanding of the processes and it has been concluded that a reversed event-tree structure can better be used to visualise the Process System. This 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: (1) Definition of the system, (2) Selection of FEPs relevant for the defined system, and (3) Identification of influences between the FEPs.

All of these steps need to be well documented and compiled. In addition to the influence diagrams, an extensive database with descriptions of all FEPs and inter-

actions between them have to be prepared with links from each FEP and influence-arrow to the pertinent detailed description or definition in the database. The influence diagram approach has been applied in Sweden e.g. for the performance assessment exercise, SITE-94 [SITE 94] and for the SFL 3-5 repository [Skagius 94]. Based on the reduced Influence Diagram, containing about 900 influences, the Reference Scenario and a further Reference Case was formulated to carry out the subsequent quantitative consequence estimation of releases from the repository.

The third method for scenario analysis as described in [Eng 94] is the 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. However, the approach is 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 of 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 clockwise 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 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 have a direct or a rather complicated indirect effect through a pathway M_{4132} involving three subinteractions I_{41} , I_{13} and I_{32} .

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. The RES interaction matrix can describe the whole repository system with all of the technical and natural release barriers at the same time or sub-divided into several individual interaction matrices for a group of parameters describing a subsystem. For example, within

the sub-project AFA-1.2, a restricted demonstrative exercise on the near-field sub-system of an idealised repository system was organised in connection with one working group meeting. This simple application is summarised in the final report of AFA-1.2 [Vuori 97a]

What has been described above regarding the content of the RES approach can be categorised as a 'soft' application aiming at defining the scenarios of performance assessment. The consequences of these scenarios are then evaluated with relatively simple models. Alternatively, in the 'hard' application of the RES approach, a fully-coupled model is established for the system to be analysed involving the explicit equations for all the interactions. However, it is usually 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 section.

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 to the desired degree of detail. 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 an appropriate conceptual model is dependent on the purpose and aims of the pertinent study. For example, the estimation of the total groundwater flux through a repository requires a less detailed conceptualisation as compared to the case where a detailed distribution of fluxes among the sub-structures 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, in any case, all of the modelling assumptions have to be carefully documented and justified.

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.

Mathematical models are required as the primary tool of performance assessment. Together with the appropriate system- and site-specific model parameters, they present 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, semi-analytically 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 are a whole spectrum of models available with 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 in the capabilities of the models to describe the processes in a way that is compatible with experimental results.

At the other end of the spectrum of models lies the assessment models for sub-systems, such as the near-field, and the whole repository system. These models usually have a simplified geometry and an 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 bound. To obtain a quantitative view of the safety margins, analyses with realistic or best estimate parameter values and for the most probable courses of events are useful. With the increasing power of computer hardware, simplifications are not so necessary, but increasingly complex phenomena can, in principle, also be accounted for in the applications of assessment models. Nevertheless, simplified modelling procedures are needed to increase the transparency of performance assessments in view of the needs of broader audiences involved in the decision-process for nuclear waste management projects.

Analysis of consequences and their uncertainties for key scenarios

The most simple ways of describing the consequences of the chosen key scenarios of performance assessment can be accomplished by straightforward scoping and bounding analyses in order to obtain 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 be employed. In the deterministic approach, each individual calculation scenario or case is analysed using a single set of fixed parameter values. Quite sophisticated models can also be applied. In this approach, a base case can represent either the best estimate or a 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 deterministic analyses, no attempt is made to differentiate the scenarios considered by assigning a certain probability of 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 sub-system or the whole repository system. The final result of consequence analysis is then also expressed in a form of a 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 also cover the uncertainties related to the choice of scenarios and conceptual models.

Another aspect related to uncertainties arises from the fact that performance assessments 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.

3.3 Phenomena and interactions

This Nordic study focuses on safety in the final disposal of long-lived low- and intermediate- level waste. In the preceding section, general methodological aspects of the analyses of the performance of a repository system were discussed. The topic of this section is devoted to the consideration of those physical and chemical phenomena - as well as their mutual interactions - that are expected to play a decisive role 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.

3.3.1 Typical repository concepts for low- and intermediate- level wastes

A succinct summary of the planned and existing repository concepts for low- and intermediate-level waste repositories has been presented in Section 3.1. There are obvious differences with respect to 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. With respect to 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 wastes, 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 mixtures, or in some cases, cementitious products) 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 the groundwater table or assumed to be saturated in some performance assessment scenarios)

A simplified repository concept (Figure 3.2) containing some of these common features was considered as a topic of the demonstrative applications of the interaction matrices according to the RES approach [Vuori 97a].

3.3.2 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 a 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). In the case of metal wastes, a galvanic reaction might occur between the waste form and the 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. By comparison, inorganic complexing agents are, in most cases, less important. The strength of non-desirable interactions and direct contacts can be avoided or suppressed by the presence of engineered barriers.

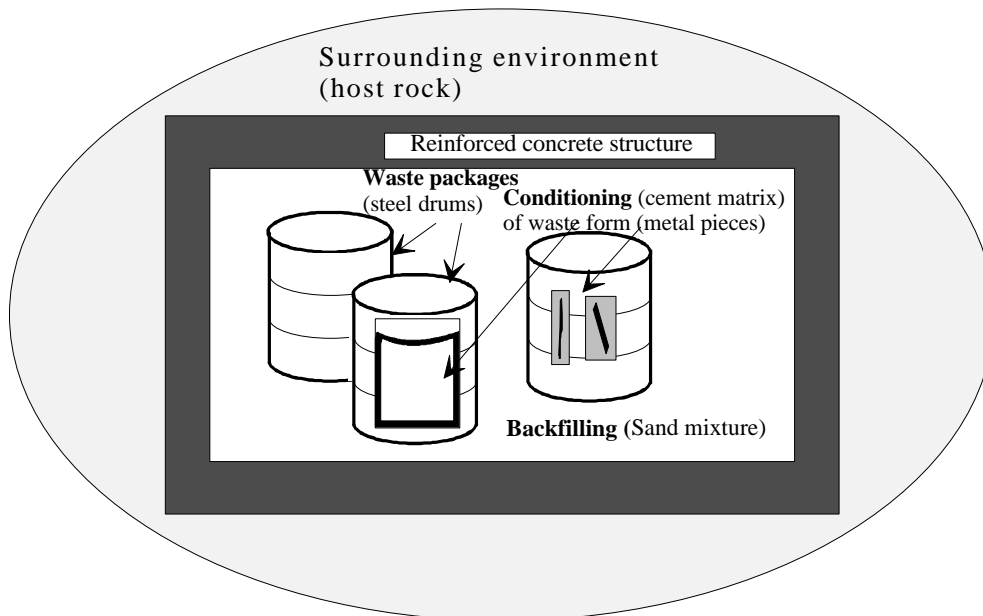


Figure 3.2 A simplified system of engineered safety barriers used as the basis for a RES interaction matrix workshop session within the AFA-1.2 sub-project.

3.3.3 Effects originating from the behaviour of engineered barriers

One function of the engineered barriers is to avoid or defer, for as long as possible, the direct contact of flowing groundwater percolating 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 is 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/co-precipitation of radionuclides with corrosion products and co-precipitation with calcite. Re-precipitation 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, ultimately the concrete barriers might be altered by the 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 also take place 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 impair 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 to be 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.

3.3.4 Effects caused by groundwater flow

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 the 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 the removal of some important chemical components from these barrier materials during their gradual degradation, hence the reason for increased permeability and the reduction of sorption capacity.

In the case of repositories for long-lived low- and intermediate-level wastes, the heat generation rate is low when 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, the potential of heat generation for enhancing the flow of groundwater is also negligible.

3.3.5 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 the 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 the condensation of water inside the waste packages, swelling and the 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. The salinity of groundwater also 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.

3.3.6 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 in the case of low-level wastes, the package is often assumed to be degraded fast and subsequently, it is assumed that the radionuclides will become instantaneously and evenly distributed 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 the amount of water available, the sorption properties of structures and the backfill barriers regarded as a homogeneous mass. For certain elements, the solubility limitations also 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 re-

pository 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 the total and simultaneous loss of function of all engineered barriers and hence complete reliance on the functioning of the natural barriers.

3.4 Examples from the Nordic countries

Some safety assessment studies and related work are briefly described below.

3.4.1 Examples from Denmark

No disposal facilities have been constructed in Denmark and comprehensive safety assessments have, therefore, not been carried out. However, safety-related processes relevant for the disposal of bituminized and cemented waste containing soluble salts have been studied experimentally. It has been demonstrated that hygroscopicity might be an important property for the disposal of such materials under unsaturated conditions [Brodersen 92]. The precipitation of calcite from the groundwater inside cracks in defect concrete barriers has also been studied and shown to have a strong influence on pH and, thereby, on the retention of many radionuclides [Harris 96]. Research models for the phenomena are available.

3.4.2 Examples from Finland

Safety studies for Olkiluoto Repository

Construction of the VLJ Repository was finalised in the summer of 1991. The VLJ Repository is an underground disposal facility for low- and intermediate-level operational waste generated by the Olkiluoto nuclear plant. A summary of the main features of this repository has been presented in Section 3.1. The repository consists of two silos excavated at a depth of 60-100 meters in the bedrock. The silo for low-level waste is a shotcreted rock silo. For intermediate-level waste a reinforced concrete silo has been constructed inside the rock silo.

The Finnish regulations for disposal of low- and intermediate-level radioactive waste 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).

The Final Safety Analysis Report including the post-closure performance assessment and the application for the operation license were submitted to the authorities until May 1991. The post-closure performance assessment is comprehensively summarised in [Vieno 93]. The safety analysis is based on detailed site investigations performed before and during the construction of the repository. Properties of waste products and engineered barriers have also been comprehensively studied

for 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 the 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.

Preliminary analyses for safety in connection with the 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 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 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 effectively prevents releases into the biosphere. The most important barriers and phenomena restricting releases are good corrosion resistance of metals in a 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 m³ per year and the amount of well water ingested is about 1 m³/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.

3.4.3 Examples from Norway

After a process of selecting a suitable site for a low- and intermediate-level radioactive waste repository in Himdalen in Aurskog-Høland municipality (see Section 4.4) a detailed study of the geological and hydrological properties of the location was performed and agriculture in the area, plant and animal life, hunting activities, tourism, ancient monuments and population in the area were considered. Special emphasis was put on the transport distance from IFE, Kjeller, to Himdalen.

As inflow of water is unavoidable in caverns and tunnels built into rock formations. The tunnels decline slightly (1:50) from the caverns to the tunnel entrance giving the facility a self-draining property though a drainage system. The inflow of

water is calculated to be in the range of 135 - 1350 litres per hour. After excavation, the groundwater level will fall as the rock is drained and the water level will reach the floor of the caverns after about 15 years. The groundwater level is calculated to be stabilised after 55 years. This means that all water gradients are directed towards the caverns and the transport of radionuclides through the rock is nearly impossible. The only pathway of contaminated water is through the tunnel due to the self-draining property of the facility.

Two drainage systems are to be installed. One system will collect water leaking from the surrounding rock formation into the caverns. This water will not contain radioactive pollution but must be drained in order to keep the facility dry. The second drainage system will collect water leaking from the concrete structures if this should happen. During normal operation, this drainage system should be dry but, if by some reason, water leaks through sarcophagii or storage bunks, this water may contain radioactivity and must therefore be collected and managed as radioactive waste. Water in this system will also indicate some malfunctions in the facility. Two separate drainage sumps will be installed in the service area, one for each drainage system. Plans for monitoring the drainage water for radionuclides will be described in the safety assessment for the operation of the facility.

The Directorate for Public Construction and Property is the owner and the constructor of the new storage and repository in Himdalen and has therefore applied for a construction licence. This licence is based on a safety analysis of the facility. This safety analysis, describing the probability and consequences of releases of radionuclides is not restricted to the near-field alone but extends to the surrounding environment and the population in the area. Scenarios were developed for four main release mechanisms. These are:

- Water release scenarios:
 - Diffusive release
 - Release from flooded repository
 - Water abstraction from well intersecting the drain
- Gas release pathways
- Human intrusion
- Natural disturbances

The consequences of releases were calculated as doses to critical groups living in the area. The calculated doses were compared to the basic requirements that doses to individuals shall not exceed 1 μSv per year in scenarios judged to be likely to occur and shall not exceed 100 μSv per year for scenarios dependent on low-probability events and hence unlikely to occur.

From the safety analysis, it can be concluded that doses to individuals in critical groups will be several orders of magnitude below the required dose rate limits. The only exception is water extraction and consumption from a well intersecting the drain from a flooded repository. The probability of this scenario is extremely small and during the period of institutional control of 300 - 500 years, the water drained out of the repository will be monitored for its radionuclide content.

3.4.4 Examples from Sweden

SKB has performed a pre-study 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 [Wiborg 95]:

- 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 pre-study 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 reactor decommissioning 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 expected to be allocated to SFL 3.

Steel will be present in all repository parts, as waste, waste packaging 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, Zircaloy, 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 initial 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 graphically structure 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, which are effective in preventing the release of radionuclides, such as sorption in concrete, are also effective in the case of 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. Experimental studies are performed on:

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

The degradation of cellulose in concrete may have an influence on the chemistry of radionuclides. Experimental studies on cellulose degradation are therefore performed [Allard 95].

3.5 Discussion

While the whole project AFA-1 focused on safety aspects in general of the final disposal of long-lived low and medium level radioactive waste, the sub-project AFA-1.2 dealt more specifically with the performance assessment of the engineered barrier system (near-field) of the repositories for low- and intermediate-level wastes. The topic intentionally excluded the discussion of the characteristics of the geological host medium. Therefore, the work emphasised a more generic discussion of the features of performance assessment, independent of the fact that different host media are considered in the Nordic countries.

The different waste management systems which exist and which are planned in the Nordic countries have been briefly described. The main emphasis has been on the general discussion of methodologies developed and employed for performance assessments of waste repositories. Some of the phenomena and interactions relevant for a generic type of repository have been discussed as well. Among the different approaches for the development of scenarios for safety and performance assessments one particular method - the Rock Engineering System (RES) - was chosen to be tested by demonstration 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 has concluded that the use of the RES approach is very easy to learn even during a short discussion session. The use of different ways of indicating 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 makes it easy to link the cell-specific comments so that they are readily available for the 'reader' of the obtained results.

A short review of the performance assessments carried out in the Nordic countries for actual projects concerning repositories for low- and intermediate-level waste has also been included in the final report [Vuori 97a] of the sub-project.

4 Environmental impact assessment (AFA-1.3)

4.1 Examples from Denmark

The purpose of environmental impact assessment (EIA) in Denmark is to assess the projects that are likely to have significant effects on the environment at an early stage. The EIA follows the provisions of the Planning Act on supplements to regional plans. The rules are prepared in accordance with an EC directive..

Annex 1 in the Danish Ministerial Order No. 847 of 30 September 1994 on EIA is a list of projects that are likely to have significant effects on the environment and are subject to EIA. Projects of type 3 covers: "Installations solely designed for the production or enrichment of nuclear fuel, for the reprocessing of irradiated nuclear fuel or for the permanent storage, final disposal or processing of radioactive waste". As no nuclear installations have been constructed since the EIA rules were established, type 3 in annex 1 has not yet been used.

The EIA in Denmark gives the public the opportunity to influence a project through two periods of public consultation. Ideas and proposals from the developer are first discussed at an early stage in the process. A regional plan supplement is then proposed from the regional planning authorities who carry out the EIA. Public, authorities and developer can then assess the details of the project in this second public consultation. The regional planning authority makes the final decision on adopting the regional plan supplement and the accompanying environmental impact statement [Johansen 95].

In the following an example of a non-nuclear project which has to undergo an EIA is briefly described:

The developer "Dansk Naturgas A/S" (Dangas) wants to build an underground storage facility for natural gas (1 billion m³) near the village of Tønder in the south of Jutland. Dangas therefore requests the regional authority to carry out an EIA on the project. Three other solutions are considered at the same time.

The regional authority made a proposal for a supplement to the regional plans 1993 - 2004 in which an EIA of the project is included. The EIA includes: purpose and placing of the storage, principle and performance, environmental impacts (noise and vibrations, light, air, soil and groundwater, refuse and waste water) during construction and operation of the storage. Visual and socioeconomic impacts together with safety assessment were included. Finally alternatives were discussed.

The first public consultation was arranged, where 34 ideas, proposals and remarks to the project were received. The regional authority then made the proposal for a supplement to the regional plans. Another public consultation was to be arranged, but meanwhile Dangas stopped the project because they want to have another pipeline from gas deposits to Jutland instead of a storage facility. This is a new project where a new EIA will be necessary.

4.2 Examples from Finland

4.2.1 The legal background

The main provisions concerning the final disposal of radioactive waste in Finland are included in the Environmental Impact Assessment Act and the Nuclear Energy Act.

The Environmental Impact Assessment Act [YVA 94] requires an assessment covering the significant indirect and direct effects on man, nature and the built-up environment that will result from the different alternative implementation modes of the project. Consequently, the impacts considered include among other things the following effects:

- health, living conditions and satisfaction of people
- soil, water, air, climate, flora, fauna and interaction between them as well as biodiversity
- urban structure, buildings, landscape, townscape, cultural heritage
- use of natural resources

The environmental impact assessment procedure includes hearings of citizens in local and neighbouring municipalities and authorities and other groups whose interests may be affected by the project.

In the EIA for the final disposal facility of the spent fuel, the company Posiva Oy is responsible for the project and the Ministry of Trade and Industry (MTI) acts as the co-ordinating authority. The other parties involved in the procedure are citizens whose circumstances and interests may be affected and other authorities as well as researchers whose expert opinion has been requested. The Act emphasises interaction between the public and experts.

The EIA involves two phases. The first step for the developer is to draw up the EIA programme and submit it to the co-ordinating authority. The programme contains information on how the environmental impacts of the project and its preliminary alternatives will be studied and assessed. After submission of the programme, the co-ordinating authority announces that the project is under preparation and the parties involved can comment upon the programme.

The co-ordinating authority compiles the comments by the citizens and experts. These will be noted in the programme through the statement by the co-ordinating authority. Once the programme has been approved, the second phase of the EIA will begin. The aim of the EIA is to compile the environmental impact statement (EIS). The EIS shall be attached to the application for the decision in principle according to the Nuclear Energy Act. The hearing of the EIS can be held jointly with the hearing of that application.

According to the amendment of the Nuclear Energy Act [YEL 87] in 1994, the nuclear wastes produced in Finland shall be disposed of in the Finnish bedrock. The use of nuclear energy must be safe and it may not cause damage to people, environment and property. Nuclear energy production as a whole should also be in the total interest of the society. A positive statement from the safety authority is a binding prerequisite for the acceptance of the decision in principle by the government. Furthermore, the municipalities also have an absolute veto right with respect to the siting of nuclear waste management and disposal facilities in their area. Local authorities and citizens may express their comments and opinions in the public hearing process can be submitted either in written or in oral form.

4.2.2 Environmental Impact Assessment in progress

Possible municipalities and sites for a spent fuel encapsulation and disposal plant in Finland are Olkiluoto in Eurajoki, Romuvaara in Kuhmo, Hästholmen in Loviisa and Kivetty in Äänekoski . The Finnish nuclear power plants are located on the islands of Olkiluoto (Eurajoki) and Hästholmen (Loviisa).

According to current plans, the site for the disposal facility will be selected no later than by the year 2000. The project will finally require positive statements from the safety authority (STUK) and the host community as prerequisites for the decision in principle by the Government. The decision has to be endorsed by the Parliament. According to the present schedule, construction of the repository will start in the 2010's, and the operation of the facility will begin in the year 2020.

Posiva Oy started the planning of the EIA Programme in spring 1997 and the EIA officially began in early February 1998, when Posiva Oy submitted the EIA programme to the Ministry of Trade and Industry (MTI). The MTI publicly announced the beginning of the EIA procedure and asked for statements on the EIA programme from the possible host communities, their neighbouring communities and from a number of central and local authorities. All of the potentially affected parties can also express their opinions on the EIA programme. Sweden, Estonia, and Russia have also been informed about the start of the EIA in accordance with the Espoo Treaty. The co-ordinating authority will submit its statement with possible changes and additions to the EIA programme in June 1998. The EIS is likely to be completed and submitted to the co-ordinating authority in 1999. After that, MTI

asks for statements and opinions on the EIS. The EIA procedure will be concluded when MTI as the co-ordination authority presents its statement on the EIS.

The basic case for the environmental impact assessment is the final disposal of the spent fuel accumulated during the 40 years of operation of the present nuclear power plant units in Olkiluoto and Loviisa. In addition, the final disposal of the fuel to be accumulated during a longer operational period is to be studied.

The final disposal facility will comprise an encapsulation plant for the fuel and a deep repository in bedrock. The spent fuel will be transported to the facility from the interim storage facilities at Olkiluoto and Loviisa. The environmental impacts of all the phases of the project - investigation, construction, final disposal, decommissioning and sealing - shall be assessed in the EIA procedure.

The final disposal facility is described in the EIA programme according to the basic concept outlined by Posiva. For the facility, there are four alternative sites, which have been selected as the result of a multi-phase process, at first covering the entire area of Finland. The transport of the fuel, road, railroad and sea transport modes will be studied as well as alternative routes as examples relating to each possible final disposal site.

The environmental impacts to be assessed by the EIA cover the significant indirect and direct effects on man, nature and the built-up environment that will be caused by the different alternative implementation modes of the project. The EIA work by Posiva Oy has been divided into four assessment sectors:

- impacts on health
- impacts on nature
- economic impacts
- social impacts

The impact areas will depend on the categories of the effects and will be further refined during the assessment phase. The impact on the non-fulfilment of the project will not be assessed since the spent fuel is, in accordance with the amended Nuclear Energy Act and the MTI decisions, to be finally disposed of in the Finnish bedrock in compliance with the aforementioned implementation schedule. In the planning of the assessment programme, the participation of the citizens of the impact area has been sought.

The radiological health effects relating to the final disposal caused by the radioactive materials have been assessed since the beginning of the 1980's. The regional and urban structure of the alternative final disposal sites as well as their soil and bedrock, surface and groundwater and organic nature was also examined before the beginning of the EIA procedure.

The land use plans for the final disposal site must include a reservation allowing final disposal of nuclear waste. The construction and operation of the final disposal facility also requires other permits or decisions and notifications according to several laws and treaties.

The EIA programme also includes an information and interaction plan. Its goal is to inform the local inhabitants about the project, the EIA procedure, the interaction possibilities and to maintain interaction between the local people and Posiva. The means for reaching the goal include bulletins delivered to each household in the region, exhibitions, use of the mass media, co-operation and follow-up groups, public events, discussion groups as well as discussion events for local associations and other small groups.

A Decision in Principle (PAP) by the Government, a construction permit and an operating licence are required by the Nuclear Energy Act for the construction and operation of the final disposal facility. The EIS has to be submitted along with the application for the Decision in Principle. Preparatory work for this decision has also been commenced by Posiva Oy. The environmental impact assessment (EIA) is interlinked to the PAP-process in a number of ways. The public hearings required by the EIA Act are also stipulated by the Nuclear Energy Act. The Ministry of Trade and Industry is the responsible authority controlling both procedures (PAP & EIA).

4.2.3 Public sector's research

The public sector's research programme has included research on environmental impact assessment topics from the year 1994. The authorities need information for defining the requirements for the environmental impact assessment, especially for social impact assessment, and guidelines for preparing the environmental impact statement (EIS). Information is needed concerning citizens' opinions about siting the nuclear waste facility, also because the Nuclear Energy Act requires acceptance of the municipality for the siting of a plant.

In the municipalities, which are considered to be possible for the siting of a repository, many inhabitants have protested against the siting of the plant in their home community. This is why it is important to identify the possibilities and means for delivering interactively objective information and enabling open and many-sided discussion.

The aim of the studies in this area has been to define the societal and sociopolitical questions and issues, which should be taken into account when assessing impacts of nuclear waste management in the possible siting municipalities. In the studies, information has been collected for setting the requirements for environmental im-

pact assessment. Information has been gathered for the authorities from local organisations and inhabitants and from the experts and authorities for the inhabitants. The aim of the study is to identify channels and to create networks for co-operation and discussion between local and government authorities, the citizens and researchers as well as to carry out impartial studies about the impact of nuclear waste management. The focus of the study is primarily on local and regional issues. A summary of the survey results as well as an analysis of the results is included in [Vuori 97b].

4.3 Examples from Iceland

4.3.1 General

The law on Environmental Impact Assessment came in to force in Iceland in May 1994. The approach follows the 85/337/EC directive with some differences. For example, the list of projects, which require a mandatory EIA goes one step further than the EC directive with regards to the type and scale of projects.

The National Physical Planning Agency (NPPA) is responsible for the implementation and is the decision maker in the process but the Minister for the Environment is the principal authority.

The main aim of the Act is that if a project is likely to have significant effects on the environment, natural resources and community they are assessed prior to a decision. It also includes that an assessment is a part of the planning process.

Projects subject to EIA are divided into two categories: Projects that are always subject to EIA and projects that maybe considered to have significant effects on the environment natural resources and community (Annex II of the EC directive). The Minister for the Environment determines whether a project from the second category is subject to an EIA.

The following projects are always subject to environmental impact assessment:

- Hydroelectric power stations with an capacity of 10 MW or more, or water reservoirs where more than 3 km² of land is covered by water on account of dams and/or changes in river beds.
- Geothermal power stations with a capacity of 25 MW or more in raw energy or installed capacity of 10 MW or more and other thermal power stations with an installed capacity of 10 MW or more.

- Construction of high voltage power lines with 33 kV voltage or more.
- Gravel mines on land of 50 000 km² or larger area or where the planned removal of gravel exceeds 150 000 m³.
- Tourist centres in uninhabited areas.
- Disposal installations for poisonous and hazardous waste, and waste disposal installations where waste is disposed of, or where land is filled with waste, in an organised manner.
- Plants for the initial melting or remelting of cast iron, steel and aluminium.
- Chemical plants.
- Construction of new roads, railways and airports.
- Ports which permit the passage of vessels over 1 350 tonnes.

Furthermore, all projects listed in Annex I of Directive 85/337/EEC and not listed above are always subject to an Environmental Impact Assessment. Annex II contains a further list of projects and operations that may be subject to an EIA.

4.3.2 EIA for Seyðishólar scoria excavation

Seyðishólar are craters built up of scoria in an eruption, which occurred 6500-7000 years ago. These cover about 100 metres of the surrounding area, which is mainly covered with lava flows from the same eruption. The lava and the slopes of the craters are partly covered with vegetation and low birch woodland is in the vicinity. Close to the site are several summer houses. The scoria in the craters is estimated at 50 million m³ and it is also estimated that the eruption created a total of about 120 million m³ of scoria.

During the past 50 years, scoria have been excavated from the craters on two sites owned by different parties and mainly used for building roads, in the last years mainly for summer houses, but also in the construction industry. The plan is to take 8-10 million m³ from one side of the craters in 12 years for export and domestic use. Using up to 75 trucks per day, five days a week.

One prerequisite is to restore the wounds of the previous excavation which is difficult as there are up to 70 metres from the bottom to the top of the mine and an area of about 10 ha has been disturbed. No land use plan exists for the area.

The EIS included a good description of the project and special studies on the geology, flora and fauna, groundwater and noise calculations. The EIS concluded that the main negative impacts were on vegetation, noise disturbance in the summer house area and increased traffic of big trucks on the main road. It also concluded that the change of landscape in the hills of the craters was positive as it healed the wounds. In long term, the impacts were estimated to be minor or positive.

The EIS was advertised, the NPPA consulted the local authorities, the Nature Conservation Council and the Health and Pollution control institution. During the five weeks which the public had to comment on the EIS, the NPPA received nine comments from owners of summer houses in the neighbourhood, the Icelandic summer house owners' organisation, an NGO nature conservation group and two geologists.

In the consultants' and public's comments, the main points raised were that the geological value and change in the landscape was underestimated, the site should be restored but without disturbing as large an area as planned. The benefit of exporting the scoria is uncertain, effects on tourism and local properties were valued as negative, air and noise pollution would be significant, public consultation was insufficient and the risk of accidents on the main road would increase. The NPPA consulted a private consultant, a specialist in noise measurements, who confirmed that the noise calculations were accurate, the local road department and police were also consulted about the risk of growing traffic and confirmed that no special operations were needed.

The NPPA's decision was that the project was approved with conditions. They were that the excavation area was decreased and the Nature Conservation Council had to be consulted on forming the final landscape and re-vegetation. Special measures were to be taken to prevent groundwater pollution, among them monitoring. The number of trucks leaving the site on Fridays was limited, when the traffic doubles (summer house-tourism) compared with the extent of traffic from Monday to Thursday. The daily operating hours were limited, the road to the site should have asphalt in the first year of operation to decrease noise and dust disturbance in the summer house area. And finally, the local authorities have the power to take action if the impacts turn out to be different from what was foreseen.

The NPPA's decision was appealed to the Minister for the Environment. He verified the NPPA's decision with a stringent limitation on traffic and operating hours on Fridays and stated that the road into the site should be asphalted from the beginning.

The main lesson from this, as it is in many other cases, is that public consultation and distribution of information, explaining the projects, receiving comments and responding in the early stages of preparing the EIS (scoping) is most valuable. This

process was finished in August 1996, but the operation on the site did not start in October 1997.

4.4 Examples from Norway

The waste facilities which will be built as a combined storage facility and repository for LLW and ILW in Himdalen in Aurskog-Høland municipality, has undergone a long development process.

The siting, construction, operation and further work involving nuclear installations are mainly regulated in Norway by three acts:

- Planning and Building Act
- Radiation Protection Act
- Atomic Energy Act

The process of selecting a site for the disposal of LLW and ILW in Norway has been underway since 1989. An official committee was established by Royal Decree on 13 October 1989 with a mandate to prepare a report that described the waste and discussed safety criteria and principles for all of the LLW and ILW in Norway. The committee's report was published as NOU 1991:9.

The report gives an overview of the management of radioactive waste in Norway. In its search for possible solutions for the disposal of Norwegian waste, the committee assessed existing rock cavities (mines, road tunnels, railway tunnels a.s.o.). The construction of a new installation specially designed for the purpose of disposal was also considered. The main conclusion was to construct a repository in the abandoned mines at Killingdal near Røros, 430 km from Kjeller, as the primary solution. Secondary, the committee recommended a further investigation of the possibility of establishing an engineered repository near Kjeller. Based on this document, an investigation of possible locations for a repository in rock caverns within a 25 km radius from Kjeller started with 52 possible sites. A systematic site screening process was carried out and 13 technically comparable sites were identified. Further meetings with municipalities and local authorities to discuss future land use plans and potential environmental impacts, reduced the alternatives to two sites: Kukollen and Himdalen. These two sites were then assessed with respect to safety as well as social, economic and environmental consequences.

According to the National Code for Planning and Construction, an environmental impact assessment had to be performed. The full impact assessment for the three alternative sites Killingdal, Kukollen and Himdalen was conducted and aspects such as area planning, industry, natural resources, historical sites, recreational areas and wildlife were covered.

The Steering Committee for the impact assessment recommended that the repository should be sited in Himdalen.

A public hearing of the EIS together with the handling by the Steering Committee on Energy and Environment resulted in comments and suggestions for changes in the proposal. The Ministry of Industry and Energy then forwarded an adjusted proposal to the Parliament, in which it was suggested that the waste, which contained plutonium should be stored instead of being disposed of. It would then be possible to monitor and, if necessary, remove or re-condition the waste at a later date. All short-lived waste, in accordance with the recommendations for waste classification given by the IAEA, is to be disposed of.

In 1994, the Parliament decided that the facility should be a combined storage and disposal facility and that the investigation at the Himdalen site should continue.

The builder and owner of the facility is the Directorate of Public Building and Property (Statsbygg). The Directorate builds and owns most of the public buildings in Norway. Statsbygg had to apply for a building licence, conducted a detailed description of the building, together with safety report. Statsbygg forwarded the application to build the combined storage facility and repository in Himdalen for the Ministry of Health and Social Affairs. The Ministry sent the application to the Norwegian Radiation Protection Authority (NRPA) for consideration and recommendation. According to the Atomic Energy Act, the NRPA reviewed the safety report and provided advice to the Ministry. The final decision was taken by the Government and the licence was granted to start construction of the combined storage and disposal facility.

IFE will be the operator of the facility and applied for an operating licence. Their safety report is actually under review by NRPA. The Government will issue the license to the operator based on recommendations from the NRPA.

In response to a request from the NRPA, the IAEA convened a team of five international experts to review the Norwegian work on establishing a combined storage and disposal facility for low- and intermediate-level waste within the auspices of the IAEA's Waste Management Assessment and Technical Review Programme (WATRP). The scope of the review included the legal framework, the approach to the selection of the site, the technical concept (combined storage/repository in a rock cavern) and the long-term safety of the facility.

The team comprising experts from Canada, France, Germany, Switzerland and the USA, reviewed a large number of documents pertinent to the project which were provided in English by the Norwegian organisations involved.

The members of the review team published a report and the main conclusions were that, based on the presented existing information, the team believes that the Himdalen site, in combination with the engineering concept, can be suitable for the storage and disposal of the relatively small amount of Norwegian LLW and ILW. Comments were taken into account during the application treatment.

The review team concluded that the legal system and the licensing process as they are applied to the projected facility correspond to international standards. The criteria which have been applied with respect to the selection of a site are comprehensive and consider the important factors for both environmental protection and long-term safety. The safety assessment of the repository should be updated, periodically on the basis of the final design, the refined activity inventory and scenario development, together with the increased site knowledge obtained during construction and operation of the facility.

4.5 Examples from Sweden

4.5.1 EIA for a deep repository

The purpose of an Environmental Impact Assessment (EIA) in Sweden is to ensure that the basis for decision-making is as adequate as possible for all parties involved. It should be possible for different interested parties to reach a consensus on the content of an Environmental Impact Statement (EIS). The EIA should be an important part of an open and credible decision-making process. The Swedish Government has stated that it is “important that transparent forms for the EIA should be established at an early stage of the siting process” [Eng 97].

An EIS is formally required for certain facilities in accordance with the Swedish Act on the Management of Natural Resources etc. and the Act on Nuclear Activities. The deep repository, SFL, is such a facility [SKB 97]. A number of parties are involved in the current siting activities and in the EIA for SFL [Eng 97]:

- **The Swedish Nuclear Fuel and Waste Management Co (SKB).** SKB has the direct responsible for handling and safe disposal of the waste as well as the responsibility for submitting an EIS at the time of application to construct a repository. SKB has also the responsibility for ensuring that the process leading up to the EIS (EIA) is carried out in such a way that applicable laws and regulations are followed.
- **The Swedish Nuclear Power Inspectorate (SKI).** SKI is the regulatory authority for the safety of the waste facilities as well as safety issues in the SKB research program.

- **The Swedish Radiation Protection Institute (SSI).** SSI is the regulatory authority regarding radiation protection issues.
- **The National Council for Nuclear Waste (KASAM).** KASAM is an advisory group to the Swedish government concerning nuclear waste issues.
- **The National Co-ordinator for nuclear waste disposal.** The co-ordinator is responsible for national co-ordination in the area of nuclear waste disposal. This involves promoting the co-ordination of information and investigation work, which is considered necessary by the municipalities involved in SKB's siting studies.
- **County Administrative Boards in regions affected by siting studies**
- **Municipalities affected by SKB feasibility studies**
- **Land-owners**
- **Environmental and nature conservation organisations**
- **Other societies**
- **Researchers**
- **The interested public**

To date, feasibility studies have been performed (or are currently being performed) for five different municipalities. The feasibility studies aim at identifying interesting areas for site investigations and illustrating the possible consequences of a deep repository siting in the municipality and in the region. SKB has an information office in each municipality where feasibility studies are being performed. Numerous meetings are held with the local community and different societies and groups. SKB also performs study visits for interested groups and individuals to the already existing nuclear waste facilities in Forsmark and Oskarshamn and to the underground research laboratory at Oskarshamn [Eng 97].

Site investigations will be performed for at least two sites. The investigations will aim at providing background data for designing a deep repository with respect to the properties of the site and for carrying out an environmental impact assessment including an assessment of long-term safety [Eng 97].

4.5.2 EIA for an encapsulation plant

SKB has suggested that an encapsulation facility should be constructed adjacent to the Central Interim Storage for Spent Nuclear Fuel, CLAB, in the municipality of Oskarshamn. In 1994, a National EIA Forum regarding this issue was established. It is chaired by the county authorities and has representatives from the Oskarshamn municipal council, SKI, SSI and SKB. The various aspects of construction and operation of an encapsulation plant at CLAB have been discussed in the Forum [Eng 97].

4.6 Discussion and recommendations

The system for environmental impact assessment (EIA) in a country is dependent on the legislative structure, the legislation application, administrative practice and general social objectives. It is therefore natural that the EIA systems differ from country to country, even if the directives of the European Community (document 85/337/EEC) and internationally accepted principles are adopted. There are e. g. differences in the objectives for the EIA systems in the Nordic countries [Hildén 96]:

- The EIA system in Denmark shall guarantee the specific assessment of environmental consequences for certain projects. Emphasis should be placed on public participation and on an open decision process.
- The EIA system in Finland shall guarantee a special assessment of the environmental consequences of certain projects. Emphasis should be placed on project planning and public participation.
- The EIA system in Iceland shall guarantee a special assessment of the environmental consequences of certain projects. Emphasis should be placed on project planning and public participation.
- The EIA system in Norway shall guarantee a special assessment of the environmental consequences of certain projects. Emphasis should be placed on project planning and public participation.
- The EIA-system in Sweden shall give the authorities a basis for assessment of the effect on environment, health, safety and general interests in accordance with the Swedish Act on the Management of Natural Resources etc. for a broad spectrum of projects.

There are also differences between the Nordic countries regarding the responsibility for the Environmental Impact Statement (EIS). The commissioner of the build-

ing project is responsible for the EIS in Finland, Iceland, Norway and Sweden. The regional planning authority is responsible for the EIS in Denmark [Hildén 96].

It is valuable to know about these differences when comparisons are made between different projects in the countries.

The AFA-1.3 project mainly concentrated on information exchange and discussions. This was found to be a highly appreciated form of co-operation that if possible should continue. Both experts and representatives from municipalities and county administrative boards participated in the arranged seminars.

5 Concluding remarks

The cultural similarities between the Nordic countries give a very good base for efficient and stimulating co-operation. This has been obvious within the AFA-1 work on safety in final disposal of radioactive waste. The participants in the project have generously contributed with their experiences and the discussions during the project meetings have in many cases been even more valuable than the results presented in this report.

Representatives from all Nordic countries have participated in each of the sub-projects.

The results from the AFA-1.1 study includes recommendations regarding the characterisation of waste under treatment and the characterisation of existing and old waste packages. It is advisable to, if possible, obtain information on waste under treatment. Classification of the waste according to physical and chemical composition is also most simply achieved during treatment.

The main emphasis of the AFA-1.2 study was placed on a general discussion of methodologies developed and employed for performance assessments of waste repositories.

Within the sub-project AFA-1.3, three theme meetings about environmental impact assessments were held. The first was held in Iceland (1995), the second in Finland (1996) and the third in Sweden (1997). The participants at the last meeting expressed a wish to continue with this type of meeting and that the next meeting will be held in Norway in the autumn of 1998. This means that the work on environmental impact assessment should, if possible, continue during the next NKS programme period.

Other areas that could be of interest for the next programme period are:

- Decommissioning (plans, quantities, treatment)
- Quality assurance at waste treatment
- Environmental management at waste treatment
- Life Cycle Analysis
- Risk philosophy
- Clearance levels and principles for clearance of buildings and contaminated areas
- Follow-up activities related to storage and burial of waste drums:
Buried drums at Kjeller. Storage in bunkers at Risø. Storage in storage buildings at Kjeller, Risø, VTT and Studsvik

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Appendix 1: Abbreviations

Abbreviation	Explanation
AFA	Avfallsprogram inom NKS (Waste program within NKS)
ALI	Annual Limit on Intake
BWR	Boiling Water Reactor
CLAB	Centrallager för använt bränsle (Central Interim Storage Facility for Spent Nuclear Fuel)
EIA	Environmental Impact Assessment
EIS	Environmental Impact Statement
ENS	European Nuclear Society
EC	European Commission
FEP	Features, Events and Processes
FSAR	Final Safety Analysis Report
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IFE	Institutt for energiteknikk (Institute for Energy Technology)
ILW	Intermediate Level Waste
IVO	Imatran Voima Oy (IVO Group includes IVO Power Engineering and Loviisa NPP)
KASAM	Statens råd för kärnavfallsfrågor (National Council for Nuclear Waste)
KTM	Kauppa-ja teollisuusministeriö (Ministry of Trade and Industry)
KU	Konsekvensutredning
LLW	Low Level Waste
MÀU	Mat á umhverfisáhrif (Environmental Impact Statement)

MKB	In Sweden: Miljökonsekvensbeskrivning. MKB-dokument (Environmental Impact Statement, EIS), MKB-process (Environmental Impact Assessment, EIA) In Finland: Miljökonsekvensbedömning (Environmental Impact Assessment)
NKS	Nordisk kärnsäkerhetsforskning (Nordic Nuclear Safety Research)
NOU	Norges offentlige utredninger
NPP	Nuclear Power Plant
NPPA	National Physical Planning Agency (in Iceland)
NRPA	Norwegian Radiation Protection Authority
PACOMA	Performance Assessment of the Geological Disposal Medium-level and Alpha Waste in a Clay Formation in Belgium
PAP	Decision in Principle Process
PS	Process System
PSAR	Preliminary Safety Analysis Report
PWR	Pressurized Water Reactor
R2	Test reactor at Studsvik
RES	Rock Engineering System
SFL	Slutförvar För Långlivat avfall (Final Repository for Long-Lived Waste)
SFR	Slutförvar För Reaktoravfall (Final Repository for Radioactive Operational Waste)
SKB	Svensk kärnbränslehantering AB (Swedish Nuclear Fuel and Waste Management Co)
SKI	Statens kärnkraftinspektion (Swedish Nuclear Power Inspectorate)
SSI	Statens strålskyddsinstitut (Swedish Radiation Protection Institute)
STUK	Säteilyturvakeskus (Radiation and Nuclear Safety Authority)
Topseal	International topical meeting demonstrating the practical achievements of nuclear waste management and disposal

TRIGA	Training and research reactor
TVO	Teollisuuden Voima Oy (Industrial Power Ltd.)
VLJ	Voimalaitosjäte (Nuclear power plant operational waste)
VLJ Repository	Disposal facility for low and medium level operational waste arising at the Olkiluoto nuclear plant.
VTT	Valtion teknillinen tutkimuskeskus (Technical Research Centre of Finland VTT Chemical Technology VTT Energy VTT Communities and Infrastructure)
VVER	Russian power reactor
VVM	Vurderinger af virkninger på miljøet (Environmental Impact Statement)
WATRP	Waste management Assessment and Technical Review Programme
YVA	Ympäristövaikutusten arviointi (Environmental Impact Assessment)

Appendix 2: Participants

The table below shows the main project participants. Many other persons have participated in the project. This applies particularly to the theme meetings within the AFA-1.3 sub-project.

Participants		AFA-1.1				AFA-1.2				AFA-1.3 ¹⁾			
		94	95	96	97	94	95	96	97	94	95	96	97
Denmark													
Steen Carugati	Risø	X	X	X	X	X	X	X	X		X	X	X
Knud Brodersen	Risø	X	X	X	X	X	X	X	X				
Finland													
Jussi Palmu	Posiva	X				X					X		
Torbjörn Carlsson	VTT	X	X	X	X								
Pekka Viitanen	VTT			X	X								
Seppo Vuori	VTT						X	X	X		X	X	X
Irmeli Harmaajärvi	VTT											X	X
Iceland													
Tord Walderhaug	Geislavarnir	X	X			X	X	X			X		
Sigurður Emil Pálsson	Geislavarnir	X	X								X		
Þóroddur Þóroddsson	Skipulag										X		X
Norway													
Anita Sörlie	Strålevernet	X	X			X	X				X		
Malgorzata Sneve	Strålevernet			X	X			X	X			X	X
Sverre Hornkjøl	Strålevernet			X	X			X	X	X			
Gordon Christensen	IFE	X				X							
Steinar Backe	IFE		X	X	X		X	X	X		X	X	X
Sweden													
Karin Brodén	Studsvik	X	X	X	X	X	X	X	X	X	X	X	X

1) During 1994, the AFA-1.3 work only comprised project planning. The real project work started in 1995.

Appendix 3: Financing

Financing of the studies within AFA-1.

Country	Participation organisations	NKS-financing kDKK	National financing	
			Financier	kDKK
Denmark	Risø	500	Risø	500
Finland	VTT, Posiva	600	IVO, TVO, KTM, STUK	600
Iceland	Icelandic Radiation Protection Institute National Physical Planning Agency	425	Icelandic Radiation Protection Institute	425
Norway	Norwegian Radiation Protection Authority, Institute for Energy Technology	500	Norwegian Radiation Protection Authority, Institute for Energy Technology	775
Sweden	Studsvik RadWaste, SKI, SKB, SSI	1325	SKI, SKB, SSI	1775
Joint		225		
Total		3575		4075