NKA NORDIC LIAISON COMMITTEE FOR ATOMIC ENERGY

NKA/AO (81) 5
August 1981

ENLARGED NORDIC COOPERATIVE PROGRAM ON NUCLEAR SAFETY

NORDIC STUDY ON REACTOR WASTE

Main Report

A Joint Scandinavian Research Project
Sponsored by

The Nordic Council of Ministers

NOTE

This study was carried out during the years 1979-1981 as a joint Nordic project, sponsored by the Nordic Council of Ministers.

The views and results contained in the report are not necessarily those of the institutions who participated in the study.

Working documents belonging to the study (NKA/AO-documents) are available upon request in each of the four participating countries, see list of addresses at the end of this report.

The results described in the present report have served as background information for volumes 1-3 of the Nordic Study on Reactor Waste, NKA/AO (81) 5.

NKA NORDIC LIAISON COMMITTEE
FOR ATOMIC ENERGY

NKA/AO (81) 5
August 1981

ENLARGED NORDIC COOPERATIVE PROGRAM ON NUCLEAR SAFETY

NORDIC STUDY ON REACTOR WASTE

Main Report

A Joint Scandinavian Research Project
Sponsored by

The Nordic Council of Ministers

Organization of the Nordic Study on Reactor Waste

Project Board

Chairman: F. Marcus, Nordic Liaison Committee for Atomic Energy

Members:
- R. Boge, State Radiation Protection Institute, Sweden
- M. Bonnevie-Svendsen, Institute for Energy Technology, Norway
- K. Brodersen, Risø National Laboratory, Denmark
- H. Jensen, Nuclear Installation Inspectory, Denmark
- M. Kuuskoski*, Imatran Power Company, Finland
- O. Paakkola, Radiation Protection Institute, Finland
- N. Rydell, Swedish National Council for Radioactive Waste

Project coordinator: H. Reijonen**, Technical Research Centre of Finland

Project leaders:
- J.P. Aittola, Studsvik Energiteknik AB, Sweden: Product Characteristics
- P. Linder, Studsvik Energiteknik AB, Sweden: Disposal
- P.O. Nielsen, Scandpower, Norway: System- and safety analysis

---

Author/editor of final report: U. Tveten, Institute for Energy Technology, Norway

---

Author of Technical Part I: P.O. Nielsen

Author of Technical Part II: U. Tveten

Author of Technical Part III: M. Bonnevie-Svendsen

---

* replaced Mr. J. Heinonen in 1980

** replaced Mr. B. Berlin in 1979

SUMMARY

In 1981, 14 nuclear power reactors are in operation and 2 under construction in the Nordic countries. So far, the reactor waste originating from day-to-day operation of these plants has been stored in solidified form at the reactor sites. Within a few years a satisfactory disposal procedure needs to be established. While the main R & D efforts in the waste field have earlier been devoted to the question of irradiated fuel and waste from reprocessing, there is therefore now an increased interest in reactor waste with its much lower radioactivity but somewhat larger volumes.

Many of the methods developed to evaluate the safety of high level waste can also be applied to reactor waste. Furthermore, the principles used in nuclear reactor safety analysis are essentially applicable. The main difference is the difficulty in defining possible accident scenarios in the case of reactor waste management systems.

Since 1977, efforts have been made in a joint Nordic study to examine which facts need to be known in order to perform a comprehensive safety assessment of a reactor waste management system, and how the results can be interpreted. It turns out that even in the absence of well-defined requirements from the regulating authorities it is quite feasible to evaluate the safety of a proposed waste management arrangement and to ensure a balanced and reasonable safety level in all sub-systems.

In the present study a Reference system related to the waste generated over 30 years from six 500 MW-reactors is examined. A considerable amount of specific information is needed about the waste itself, about the intermediate storage and transportation systems, and about the repository. Fairly adequate methods are

available to calculate possible releases of radioactivity and the resulting doses to man. With a few exceptions, no mechanisms were identified in the study that could give non-occupational doses during normal operation conditions. These exceptions are connected to long-term releases from repositories, but the resulting individua as well as collective doses will be quite small. A number of abnormal events were postulated such as a fire in waste incorporated in bitumen, or construction work on top of a repository once the administrative control period has elapsed. The dominating radionuclides during storage and transportation accident scenarios are Cs-134, Cs-137 and Co-60. For most of the release scenarios from repositories Cs-137 and Sr-90 are dominating. Some scenarios are, however, dominated by the very longlived nuclides I-129 and C-14. A closer examination of the concentration in the waste of these nuclides and of their leaching properties indicates that their small - but significant - influence, as calculated, is probably grossly overestimated.

Traditionally, stringent requirements have been attached to the characteristics of solidified reactor waste products. In the study it is shown that mainly leachability and water resistance are critical properties. The mechanical stability obtained in routine solidification processes, in conjunction with the outer container (steel drum, transport container, etc.) turns out to be sufficient. Many of the test methods available for product control in the laboratory are of limited applicability when it comes to judge to what extent the full-scale products have adequate characteristics for their further handling.

In the report it is demonstrated how a safety assessment can be carried out in detail and how the resulting doses can be presented together with their probabilities of occurence. In some cases, especially in connection with the repository, probabilities are so low or so difficult to quantify that a probabilistic risk assessment is not meaningful.

Difficulties were encountered in applying ICRP methodology and available dose calculation methods to calculation of population doses due to small activity releases, and effects extending into the far future. A simplified approach must be adopted, and it appears necessary to define a reasonable time span over which to integrate hypothetical population doses. In this study it was chosen to limit the period to 500 years.

As a conclusion from the Nordic study on reactor waste it can be stated that the major tools are available to perform a careful safety analysis of specific management systems for reactor waste. Some further refinement would be useful mainly concerning the pathways for nuclide migration from repositories. Added knowledge will most certainly still further reduce the already low calculated radiation doses to man.

NORDIC STUDY ON REACTOR WASTE

MAIN REPORT

LIST OF CONTENTS

1 INTRODUCTION

Major efforts have been devoted to the development and evaluation of methods for safe handling and disposal of irradiated nuclear fuel and waste from reprocessing of nuclear fuel. Important contributions in this area have come from the Nordic countries, particularly from Sweden /1,2/.

The management systems for reactor waste have for good reasons been considered to have less risk potential, and have received less attention. Reactor waste, which is now being produced at all operating nuclear reactors, has, however, to be taken care of now, and to be disposed of without much further delay. In this study "reactor waste" comprises radioactive waste arising from the daily operation of nuclear power plants i.e. "low" and "intermediate" waste. Disposal operations and the related transport systems have come into operation in several countries in recent years, and are now being planned in the Nordic countries. Studies of waste management systems for the steps beyond waste treatment and storage at the nuclear power plant are therefore now necessary.

Ideally, a waste management system should be optimized so that the waste products match the requirements of the selected methods of storage, transport and disposal. In the past, the lack of an overall analytical approach has not led to optimization in waste management. One of the reasons is that, in the absence of disposal facilities no well-founded requirements could be formulated concerning the waste products. These facts may have led to unreasonable specifications for example concerning the quality of the waste products.

The present Nordic study should be seen as an attempt to perform a systematic examination of important factors to be considered in risk assessment of reactor waste management systems. The applied methodology could also be used to check the relative importance of various parameters and subsystems, to match them to each other, and to evaluate the sensitivity to the assumptions made.

Analytical work on this topic has indeed already been performed in the Nordic countries and elsewhere. A Swedish conceptual study has recently been completed and is expected to lead to definite plans for a central disposal facility /3/.

The present study treats waste characteristics, storage, transportation and disposal. It is important to note that the waste characteristics are obtained from studies of waste matrices and forms that exist in the Nordic countries; while the disposal alternatives are reference models, based upon existing technology, and assuming hypothetical conditions representative of geological environments in the Nordic countries.

The major part has been devoted to the performance of a comprehensive safety assessment. The main purpose is to demonstrate how existing methods may be used, and to determine in which parts of the system significant risks may be encountered. The results of such assessments may eventually be used to alter product specifications. System requirements may be relaxed in parts where the risks are insignificant; while other parts of system may turn out to be unacceptable in their present form.

The parameter values used in the reference calculations are partly realistic, and partly conservative values. A few of the models used for analysis of different parts of the system are of a realistic type, but most of them are based upon very conservative assumptions. This must be kept in mind when comparing results calculated using the different methods. Some of the parameters needed in the analysis are of a local character, e.g. population distribution. One such set of local parameters of a specific site (in the Oslo Fjord area) was selected mainly because it was available, for the project. It must thus be kept in mind that the results calculated using local parameter values are valid only for that specific site.

On the other hand, the description of the geological environment surrounding the reference disposal facilities is very schematic, and can not serve as a substitute for site specific information about local geology if a safety analysis of a real disposal facility with known siting is to be performed.

Part III of the study is concerned with results of laboratory tests performed to check important product characteristics, and also to check the analytical methods by which these characteristics have been determined.

The study has been designed to serve as a background for those who are engaged in and making judgements on various assessments of waste management. Part of the study is of a more general nature, and will hopefully be useful for a number of regulatory bodies.

2 REACTOR WASTE MANAGEMENT SYSTEM

The existing and planned waste management systems in the Nordic countries are consistent with requirements and practices in most Western countries with operating nuclear power reactors. The waste is collected in various purification systems, is later solidified in suitable matrices, packaged, stored at, or near, the power plant, and finally transported to a repository for disposal.

The management of reactor waste is planned so that radiation doses and releases of radioactive materials are kept below defined limits consistent with radiation protection regulations. Preparation of detailed criteria for reactor waste management systems have been initiated in some countries, but so far they are not ready for application.

The waste management system dealt with in this study is limited to the following steps: temporary storage at or near the power plant, transportation, and disposal. The analysis thus does not include any of the previous steps, such as treatment, solidification and packaging. It was decided to examine a few different alternatives representative of the possibilities available in the Nordic countries.

The following is a short review of the reference system used, and this is also shown schematically in figure 2.1. A more detailed description is found in Technical Part I, and in chapters 4, 5, 6 and 7 of the main report.

WASTE TYPES. At nuclear power plants many waste streams are collected and treated, each with its typical nuclide composition and chemical and physical properties. Two specific waste types were selected for analysis in the study: granular ion exchange resin from the Reactor Water

Clean-Up System (RWCS), and powdered ion exchange resin from the Spent Fuel Pool Clean-Up System (SFPCS); both from a Boiling Water Reactor (BWR). This is the most widely used reactor type in the Nordic countries. These two waste types normally account for more than 90% of the radioactive process waste resulting from operation of nuclear power plants of the Light Water Reactor (LWR) type, but may constitute less than 10% of the total waste volume.

SOLIDIFICATION. Both waste types are assumed to be solidified by mixing with a matrix material. Concrete and bitumen are chosen for this purpose, as these materials are used at nuclear power stations in Sweden and Finland.

WASTE PACKAGES. Two types of packages are considered, both in use in the Nordic countries: standard 200 liter steel drums, and specially made cubic reinforced concrete moulds. While steel drums are used for both waste in bitumen and cement, moulds are used for waste in cement only.

STORAGE. Temporary storage of solidified waste at or near the nuclear power station is a requirement in many countries with operating nuclear power reactors. Three different types of storage facilities are examined in the reference systems:

- A storage building for cementized waste, where the waste is stored in an open storage hall.
- A storage building for bituminized waste, where the waste drums are stored in shielded pits under the floor.
- A rock cavern storage for all types of waste. A cavern might be designed in such a way that it could later serve as a disposal facility. It would initially serve as a temporary storage, from where waste could easily be removed for alternative disposal. If it was decided to use it for disposal, it would then be permanently sealed.

TRANSPORTATION. Transportation of waste from the power plant to the disposal site is assumed to take place by road or sea. Transportation by rail is not considered in this study, but many of the conclusions and results for road transportation are valid for transportation by rail. Transportation by road will be by truck. For transportation by sea, a specially designed vessel is envisaged, as proposed in a Swedish study /4/. Transport containers may be used where convenient, if necessary with shielding capability.

DISPOSAL. Three different types of disposal facilities are considered:

- Shallow land burial, where the waste is placed in trenches, which are covered with the soil originally located in the trenches.
- Near-surface concrete bunker, divided into smaller cells where the waste drums or moulds are piled. The cells are filled up with concrete, bitumen or clay before closure.
- Rock cavern, with about 30 meter rock cover. The space between waste containers is filled with concrete, and all empty spaces are filled with clay prior to closure.

These repository concepts are considered in various combinations with three types of geological formations; sandy till, clayey till and crystalline rock.

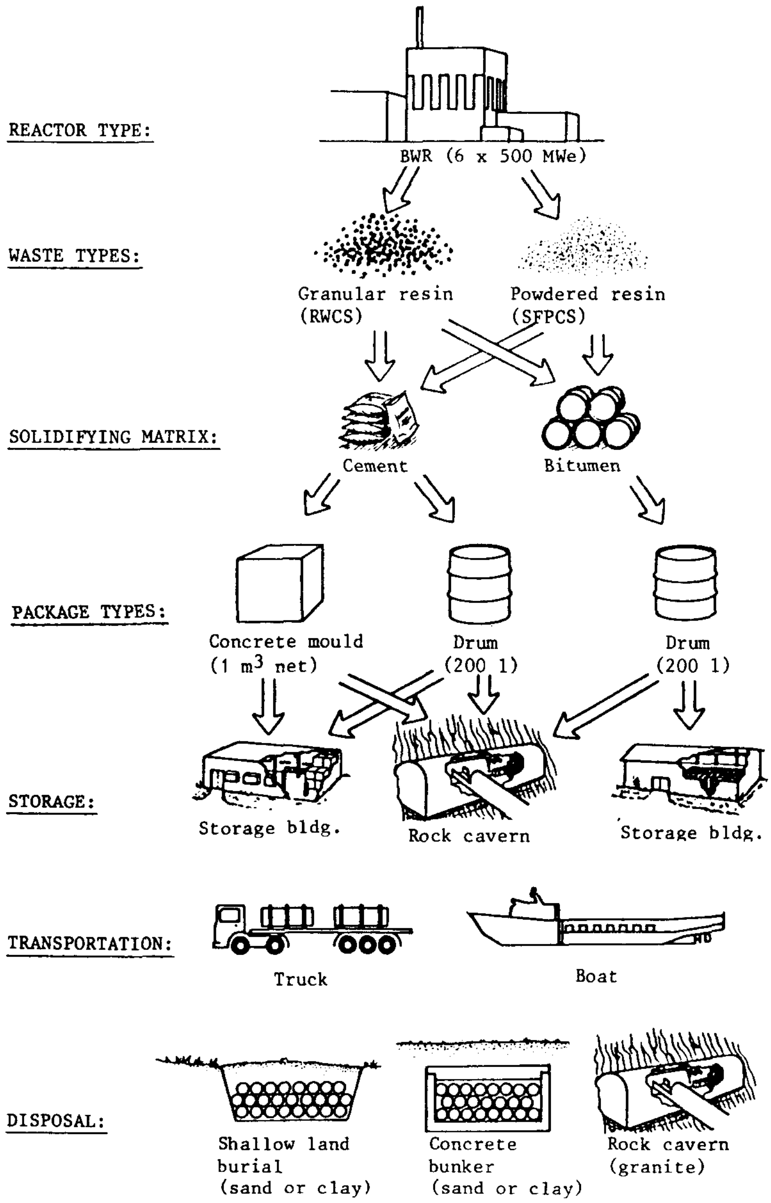


Figure 2.1 The reference waste management system.

3 GENERAL ASPECTS OF SAFETY ANALYSIS

A safety analysis may be performed in various ways and on different levels of sophistication, but typically an analysis may contain the following steps:

- Specification of the system
- Identification of e.g. radiation exposure modes
- Calculation of consequences
- Evaluation of probabilities for these consequences
- Discussion of important factors and the influence of variation of the assumed values of these parameters

The safety analysis should encompass normal events as well as accident conditions.

A sequence of events and/or conditions leading to a release of radioactivity to the environment is often referred to as a scenario. In some cases part of the exposure pathway is seen as part of the scenario, e.g. leakage from a repository giving exposure via farm products.

In relation to a release of radioactive materials the consequences both to the individual and to the population as a whole should be examined.

The consequences may as a rule be expressed as doses, and these could be used by the authorities for judging the acceptability of a proposed waste handling system. The doses could also be translated into casualities and/or economic consequences, for comparison with risks from other activities in society, and for economic considerations.

Risk is generally expressed by the relationship between consequense and probability. it has frequently been expressed by the "average risk", which is e.g. the average number of fatalities per year. The risk related to reactor accidents has for instance often in the past been expressed in this manner. If the consequences of an accident could be very large, this representation of risk is not satisfactory. Expressed as average risk, the risk related to small accidents occurring relatively frequently, will be the same as the risk related to extremely infrequent accidents with catastrophic consequences; but when risk is expressed as one single number (the average risk) this quality difference will not be observed. This is the reason why it has become usual to present the risk as so-called risk spectra. The need for this is however much less obvious when the consequences from all possible scenarios are moderate.

In the present study the risks are presented as a type of risk spectra, in a few cases where this has been possible. They are not complete risk spectra, as it has not been possible to identify the release spectra needed for a complete analysis, linking magnitude of release and probability of release. For all the analysed scenarios, the releases specified are always given as one single release for each scenario, and this is usually a release resulting from quite unfavorable assumption about the conditions leading to the release.

Though the calculated risk spectra are based upon a single value for the release magnitude and probability for each of the analysed scenarios, they give valuable information about how the consequences will vary with weather conditions, combined with population distribution around the site of the release.

In the present study it was chosen to present the consequences of releases as radiation doses, individual and collective. It is important to point out, however, that the definition of the doses may differ somewhat between the different scenarios. One main reason is that the doses in some scenarios are dominated by nuclides of moderate half-life, while in other scenarios they are dominated by very long-lived nuclides.

Calculated individual doses may primarily be used for judging the acceptability of acertain process or subsystem of a waste management system. Modifications of the products or system may be required in order to satisfy individual dose criteria.

Calculated collective doses can in principle be used in comparisons of the level of safety in the various parts of the management system. In some countries collective dose criteria are specified, and the calculated collective doses may be directly compared to these. If these are satisfied, it might still be pertinent to compare the risk related to these collective doses in various parts of the management system. For such a comparison to be undertaken, it is however also necessary to know the related probabilities. If this can be achieved, two different types of conclusions may be drawn from the comparisons:

- That the level of safety in a particular part of the management system is so much higher than in the other parts of the system, that an alternative and cheaper solution may be employed.

- That the level of safety in a particular part of the management system is so much lower than in the other parts of the system, that this part alone determines certain requirements, e.g. to quality, and that a better solution to this subsystem ought to be found; and that the cost of this solution should be balanced against the possible savings due to relaxation of product requirements.

The collective doses calculated in this study are in some cases the total collective effective (wholebody) dose equivalent commitments (dose commitments), and in some cases the collective annual committed effective (wholebody) dose equivalents (committed dose).

- "Committed dose" means the dose resulting from a body burden of radioactive materials, and is received distributed over the time from intake to radioactive decay, expulsion from the body, or a combination of these two factors. In the case of a long-lived nuclide which stays in the body, it is usual to integrate over 50 years to obtain the life-time dose.

- "Dose commitment" means the dose resulting from an "environmental burden" of radioactive materials, and is received distributed over the time from release to radioactive decay, transfer to parts of the environment where there is no exposure of humans (e.g. deep ocean), or a combination of these two factors.

4 WASTE

4.1 INTRODUCTION

Reactor waste can be defined as all solid, valueless, radioactive materials which are removed from a nuclear power reactor during normal operation. Spent fuel is not valueless, and is by this definition not reactor waste. Two other types of waste directly associated with the reactors are also exluded by this definition. One is waste from decommissioning, the other is waste which may arise in connection with remotely possible larger accidents. Both these types of waste should be taken into account in the planning of waste handling facilities. They will not be commented further upon here.

For a reactor waste safety analysis information must be available about:

- the amount of waste

- the type of waste, i.e. the chemical and physical properties of the material

- the activity level distributed on nuclides with special emphasis on any longlived nuclides present

- chemical states affecting the release and migration of such nuclides

In the case of reactor waste the available information will mostly be incomplete. Some mixing of wastes having different properties will occur. Radioactivity measurements are generally limited to a few gamma emitting nuclides. It can be difficult to specify chemical states. A substitution of experimental data by too conservative estimates can lead to unrealistically pessimistic results.

In the following some general comments on various types of reactor waste are given together with somewhat more detailed information about the waste types used as example in the reference system. Further information about this waste is available in Technical Part I, chapter 2.

4.2 TYPES OF REACTOR WASTE

This study is mainly dealing with waste from BWR's. The origin of the activity is either corrosion products or fission products and actinides. Transport of activity from the core of a light water reactor to the outer systems, where it is collected as waste or released, may take place in four different ways:

- The first is the intentional replacement of components from the core, eiher scheduled or due to necessary repairs. This gives rise to relatively small amounts of solid waste in form of activated construction materials and similar objects. The activity and the nuclide composition are determined by the composition of the construction material, the exposure time in the core and the neutron flux.

- Another, though less important, way is the transport of gaseous activity from the core. Compounds containing I-129 and C-14 tend to concentrate on filter materials in the gas retention system. Some Cs-135 may also be present from decay of a noble gas fission product. Waste from the offgas systems could therefore have a nuclide composition different from ordinary reactor waste.

- The main transport mechanism is the circulating water in various systems of the reactor. By far the largest amount of radioactive waste produced by a LWR arises from the purification of water. "Wet waste" from the water purification systems consists of spent ion exchange resins, filter sludges and evaporator concentrates. In PWR's filter cartridges are widely used, giving rise to a special type of solid waste.

- A fourth category comprises waste from decontamination operations.

Beside the relatively modest volumes of wastes which contain significant radioactivity, larger volumes of low- and inactive wastes are also produced. They consist mainly of solid materials, for example protective materials of various kinds, which must be taken care of because they can have been contaminated by contact with reactor water or other radioactive sources. The majority is burnable, and considerable volume reduction can be achieved by incineration. Otherwise, the management of such partly inactive wastes is largely a question of segregation and administrative routines.

4.3 ION EXCHANGE RESIN WASTE

Most of the activity collected in reactor waste is associated with the wet wastes, especially with the granular ion exchange resin from purification of reactor water in the primary system (RWCS-wastes). Considerable activity may also be present in powered resin from the purification system for water from the spent fuel storage pool, (SFPCS-waste). The other main type of wet waste is evaporator concentrate, but the amount produced at Nordic power reactors is generally low.

This study concentrates upon the spent ion exchange resins (RWCS and SFPCS) from boiling light water reactors. The same methodology could be extended to cover other types of waste if the basic information about the properties of the waste is available.

The RWCS- and SFPCS-waste produced at the power reactors is normally a mixture of strong cation and anion exchange resins with only partly used capacity. Some thermal and radiation degradation may have taken place. The properties of used resins can differ considerably from fresh ones. The resin may contain a considerable amount of corrosion product sludge. Even after complete drainage 50% or more water remain in the resin. This can only be removed by drying. Mixing of various batches of resins, often with widely different properties, can usually not be avoided at the power stations. The majority of radionuclides collected on the ion exchangers will be present in an ionic state, C-14 and I-129 as anions such as carbonate, hydrocarbonate and iodide ions.

In table 4.1 a summary is given of the production rates assumed for ion exchange resin waste from a BWR. Table 4.2 presents estimated mean values of activities in these wastes as derived from an analysis of experimental data from US and Nordic power reactors. To give an impression of relative potential radiotoxicities, even the number of ALI´s (the new ICRP values for annual limit of intake, occupational exposure) /5/ for the two waste types combined have been calculated. The decay of activity and the resulting decline in toxicity is illustrated in figure 4.1.

Table 4.1 Estimated annual amounts of reactor waste from a 500 MW_eBWR. Only values for RWCS and SFPCS are given. Other types of waste mostly with much less activity will also be produced. The amounts can be converted to waste produced per GW_e-year by multiplication with 2.5.

		RWCS	SFPCS	Total
Wet waste decanted	m^3/year	12	10	22
Dry resin	kg/year	2,400	1,000	3,400
Solidified in bitumen	m^3/year	6	2.5	8.5
or concrete	m^3/year	12	4	16

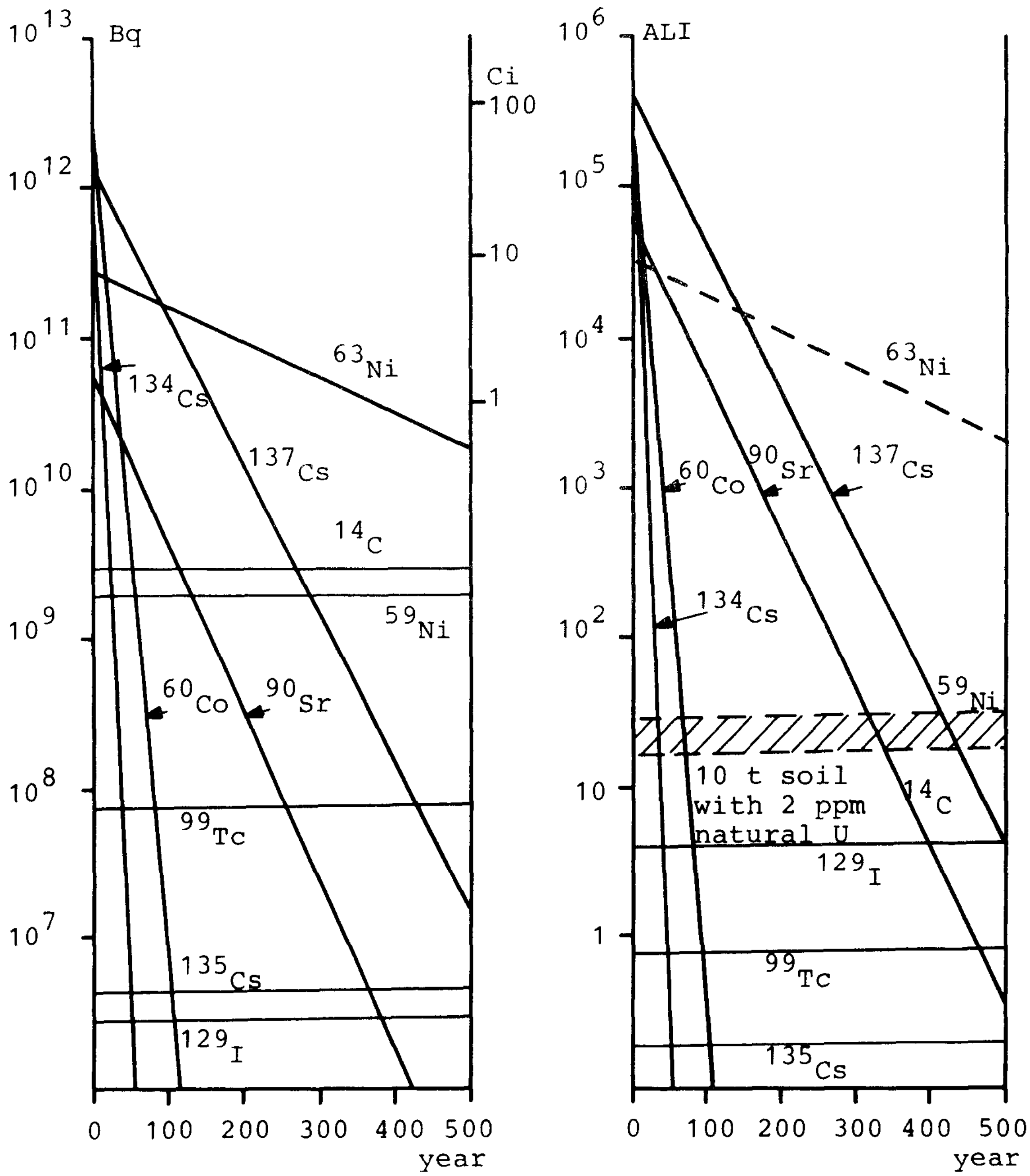


Figure 4.1 Decay of activities and relative toxicities for an annual amount of 3.4 tonnes dry resin waste from a 500 MW_eBWR.

The activity of wet waste can only be due to nuclides which are present in significant amounts in reactor water. As long as actinides can not be found in the reactor water, the waste can be regarded as not containing actinides.

The content of fission products depends very much on the number of defect fuel pins in the core. Cs-isotopes are the dominating fission products. The release rate for Sr-90 turns out to be considerably lower than for Cs-137.

The production and release rate of activation products is also different from reactor to reactor. The most important activated corrosion products are Co-60 and Ni-63.

Using correlation factors derived from analyses of reactor water it may be possible to estimate the content of pure beta emitters such as Sr-90 and Ni-63 from gamma spectrometric analyses of the Cs-137 and Co-60 content.

In most cases practically the total amount of nuclides released to reactor water is collected in the waste. A noteable exception is C-14 where less than 0.1% of the amount produced by activation of reactor water has been found by analysis /6/ to remain in the waste while most of the rest is probably released to the atmosphere. This seems to make a detailed safety analysis of the remaining 0.1% in the waste somewhat irrelevant. Similar objections apply for I-129 considering the very low activity in the reference waste. Both are, however, interesting in connection with methodology development since under unfavourable conditions they represent the worst cases of longlived nuclides with high radiotoxicity (I-129) and minimal retention in waste matrix and surrounding barriers.

During the first period where wet waste is conditioned and converted to solid waste units the main problem is external radiation due to relatively short-lived nuclides such as Co-60 and Cs-134. After a prolonged period of storage a considerable part of these nuclides will have decayed and the remaining external radiation will be due to Cs-137. As indicated in figure 4.1 the activity for the first hundreds of years will be determined by Cs-137 and Sr-90 and somewhat later by Ni-63. It follows that even after a long period of storage, operations like transport and disposal involve a certain risk potential. In the long term it is only the small amount of long-lived nuclides such as C-14 and I-129 which can give some radiation doses. Some knowledge of the content and behaviour of the long-lived nuclides in the waste is therefore needed. It is, however, indicated by the comparison shown in figure 4.1 with toxicity due to natural activity from an amount of ordinary soil (or inactive concrete) similar to the waste volume that the long term problems can hardly be very important for these types of waste.

4.4 WASTE MATRICES

Various possibilities exist for the conditioning of ion exchange resin waste. The simplest procedure is to pack drained but otherwise untreated resin into suitable containers. The transfer of as much as possible of the activity into a ceramic matrix system may result in improved fixation of the activity in a small volume /7/. Incineration of resins is also a possibility and may result in some volume reduction /8/.

Solidification by mixing with a suitable matrix material is at present regarded as the most practical manner to secure improved safety during handling, storage and disposal of reactor waste. Release of activity from the monolithic blocks produced by solidification can be expected to be much less and especially much slower than from unconditioned waste.

Various matrix materials can be employed in the solidification of ion exchange resins. The main types are: cement, bitumen, and various polymers. Various bituminization and cementation processes are used at Nordic power reactors and research facilities. Experience with polystyrene, polyester and modified polyester which permit solidification of undried waste, is available from other European countries.

It is difficult to predict which system will be preferred in the future. The low temperature systems with a cement or polymer matrix certainly have some advantage over bitumen as far as conditioning process is concerned. If the properties of the resulting products should prove to be of minor importance, this may indicate that bitumen will not be used extensively for the conditioning of reactor waste.

4.5 CONTAINERS

Some kind of container must be used for monolithic waste units. Easy availability is the main advantage of the commonly used 200 liter steel drum. Placed in a suitable environment such as thin steel plates may have considerable life-time as a protective barrier. This has generally not been taken into account in safety analyses.

In connection with cementized waste large concrete moulds of cubic or other shapes are often used. Their relatively thick walls provide some shielding and will also serve as an additonal migration barrier against leaching of activity from the contained waste.

The safety of unconditioned waste placed in a sufficiently strong and durable container may be quite as good as the safety of the same waste conditioned by solidification in a matrix material and placed in a less durable container.

This could in principle be demonstrated by the safety analysis methodology presented in this study. The information needed about the containers are their durability and in case of porous materials (concrete), diffusion coefficients for the nuclides in the materials.

4.6 WASTE VOLUMES

The capacity of a storage, transport, and disposal system for radioactive waste is as a first approximation determined by the volume which must be handled. This means that it should be an economic advantage to concentrate the waste as much as possible in the conditioning processes and to use thin-walled waste containers which can be packed closely. Problems with external radiation during handling, and perhaps, with decreased chemical stability and internal radiation damage during storage and disposal will place some limits on how far it is practical to proceed in volume reduction. The optimal solution of this problem depends on an overall cost-benefit analysis of the system including conditioning of the waste. This has been outside the scope of this study, but the safety analysis presented is an important part of such an overall evaluation.

The example calculations given in the following are based on the expected waste volumes from conditioning of ion exchange resins used at six 500 MWe BWR's in 30 years. The waste volume is about 1500 m^3 as bituminized material or 2900 m^3 as cementized material. This volume represents only the most active resins from the reactors. Other types of reactor waste give rise to significant additional volumes, but it is unlikely that the low activity wastes need to be handled and disposed of in the same careful manner.

Table 4.2 Estimated activity in reactor waste from a 500 MWe BWR.

Only values for RWCS and SFPCS are given. Other types of waste mostly with much less activity will also be produced.

GBq's are converted to Ci by multiplication with 0.027.

ALI's are the new ICRP values for annual limit of intake occupational exposure. A few values are not yet available. In this case the old MPCw values multiplied by the water intake of a "standard man": 0.8 m^3/year are used. The values are probably somewhat too low. The number of ALI's are a measure of the relative potential radiotoxicity of the waste.

	Nuclide	Half-life	Activities in resins RWCS	SFPCS	ALI	Number of ALI's in both wastes
		year	GBq/year		MBq/year	
Activation products	C-14	5,730	3	0.1	200	16
	Co-60	5.3	3,000	200	20	160,000
	Ni-63	120	300	20	9	36,000
	Ni-59	80,000	2	0.1	60	35
Fission products	Sr-90	28	50	5	1	55,000
	Tc-99	210,000	0.05	0.03	100	0.8
	I-129	17,000,000	0.002	0.001	0.7	4
	Cs-134	2.1	1,000	800	3	600,000
	Cs-135	2,000,000	0.003	0.002	30	0.2
	Cs-137	30	1,000	500	4	375,000
					Total:	1,230,000

5 STORAGE

5.1 PURPOSE OF STORAGE

The temporary storage of radioactive waste at nuclear power plants serves as a buffer, collecting the waste until it can be transported to a facility for disposal. If a disposal facility is available, the storage period needed is fairly short, and the capacity of the storage can be kept small.

Prolonged storage of the waste does, however, also offer an advantage as it allows decay of the short-lived radionuclides before the waste packages are eventually removed from the controlled area of the power plant. This reduction in activity lessens the shielding requirements during transportation, and also reduces the consequences, in the unlikely case that an accident should happen during transportation.

5.2 FUNCTIONAL REQUIREMENTS

The temporary storage must satisfy a number of functional requirements. Specific requirements may vary with national regulations, site, climatic conditions, property of waste packages, etc. In the Nordic countries the storage must generally

- have sufficient capacity
- provide necessary radiation shielding
- provide weather protection for the packages (rain, snow, frost)
- provide reasonable protection against influence from outside (missiles, fire). (Valid only for a small fraction of the waste)

Finally, the storage facility must be equipped with suitable equipment for internal transport and handling of the waste packages, such as fork lifts with shielded cabins, remotely operated overhead cranes, etc.

5.3 FACTORS TO BE CONSIDERED

5.3.1 WASTE TYPES

The prime factors to be considered in the design of a waste storage facility are the waste characteristics:
nuclide composition and concentration, matrix material, package type, radiation level, and product properties, in particular stability and fire hazard. Only a small percentage of the waste does have a content of radionuclides sufficiently high to cause a significant radiation from the package, while the majority of the waste is characterized by a very low radiation level. It may therefore be economical to divide the storage into two parts: one part for the waste that requires heavy shielding and one part for the waste that requires none or only moderate shielding. The first will normally include heavy concrete walls and roof for shielding, the second could be a much lighter construction. If the waste - or part of it - is inflammable the fire hazards should also be considered in the design, for instance by dividing the storage into compartments, to prevent spreading of fire. Installation of fire- or smoke detectors and fire extinguishing systems will mostly be in addition to the passive precautions.

5.3.2 STORAGE CAPACITY

The dimensions of the storage will depend on the number of packages produced per year and the storage period. Both factors may be difficult to estimate: the amount of waste could vary considerably from plant to plant even for plants of similar design. The storage period will depend on when a final repository will be available, this in turn depending on uncertain political decisions. A long storage period may require some precautions to limit degradation of the waste packages, for instance through corrosion due to high humidity, or, in the Nordic countries, through cracking and crumbling due to many freeze-thaw cycles.

5.3.3 LOCATION

The location of the storage may have some influence on the design. In most cases the storage will be located within the controlled area of the nuclear power plant. Such co-location will significantly simplify the handling procedures, the survey of the storage and the radiological survey of operators.

5.3.4 SITE DATA

The climatic conditions will, in general, influence the design. In the Nordic countries the main question is whether the site has a wet, coastal climate or a drier, more inland type climate. A high humidity may cause a more rapid corrosion of steel drums, probably further increased by a high salt content in the air, hence some precautions to protect the waste packages may be necessary. Low winter temperatures may require other countermeasures to limit the number of freeze-thaw cycles which in some cases could cause deterioration of the waste packages.

In order to perform consequence- and risk-analysis, statistical data on wind direction and atmospheric stability are also required.

Other site data that should be evaluated include natural phenomena such as

- seismic activity
- soil stability
- possibility of flooding

and the possibility of certain man-caused events such as:

- explosions in the immediate vicinity
- airplane crash
- sabotage

All these factors are generally evaluated for all nuclear power plant sites. A site approved for a nuclear power reactor will, as a rule, be suitable also for a waste storage facility.

The population density is, according to the safety analysis, apparently not critical in itself, although for psychological reasons one will certainly prefer to locate a storage facility outside heavily populated areas.

5.3.5 HANDLING EQUIPMENT

The equipment for handling the waste depends on the types of packages, the arrangements and lay-out of the waste treatment plant and the design of the storage facility. Two main types are available: equipment with integral shielding for protection of the operator and remotely operated equipment. Often both types of equipment will be used, for instance a fork lift with a shielded cabin for transfer of the waste from the treatment plant to the storage facility while handling within the storage could be by means of a remotely operated overhead crane. Shielded containers for handling the most radioactive packages may also be required to protect the operators.

5.3.6 EXISTING STORAGE TYPES

Most storage facilities have so far been designed as surface structures. In certain locations underground storage may be economically attractive, in particular where the rocks permit direct construction of a cavern. Such a rock cavern storage was recently put into operation at the Oskarshamn power plant in Sweden. For an underground storage, the possibility of flooding should be evaluated.

5.4 SIGNIFICANT SAFETY-RELATED EVENTS

The safety analysis described in chapter 8 shows that the only event in a storage facility that could lead to significant releases of radioactive materials, and consequently to radiation doses to the population, is a fire in bituminized waste. Although the probability of a fire is assumed to be very low, it does seem to be important to prevent a fire involving large amounts of radionuclides. Compartmentation of the storage facility is a reliable method to limit the amount of waste that could be involved in a fire. Alarm systems detecting smoke or fire must be installed, and fire fighting equipment must be readily available.

On the other hand, recent fire experiments with ion exchange resins incorporated in bitumen indicate that the release of radionuclides may be significantly lower than previously assumed. Further the combustion properties of different bitumen types vary considerably, some types will not even sustain a fire when the external heat source is removed. It is thus possible that the consequences of a bitumen fire are significantly less than will be calculated in this study. The probability of a fire can apparently to some extent be controlled through the choice of bitumen type.

6 TRANSPORTATION

6.1 PURPOSE OF TRANSPORTATION

Sooner or later the radioactive waste must be moved outside the controlled area surrounding the nuclear power plant where the waste was produced, except for the few cases where the final disposal takes place at the site of the power plant.

6.2 FUNCTIONAL REQUIREMENTS

The transportation system will need to satisfy a number of functional requirements. They specifically depend on authority requirements, property of waste packages, mode of transportation, transport routes, and to some extent, site, climatic conditions, etc. In the Nordic countries the following requirements will generally be applicable to the transportation system and to its operation:

- have sufficient capacity
- provide necessary radiation shielding
- provide adequate mechanical protection
- provide protection against excessive heat (primarily for waste incorporated in bitumen).

6.3 FACTORS TO BE CONSIDERED

6.3.1 WASTE TYPES

The prime factors are waste related: nuclide content, matrix material, package type, radiation level, and product properties, besides the required transport rate and the total transport volume.

As mentioned above, it has been assumed that the waste is incorporated in a matrix material to form a solid block. It is not unlikely that in the future more advanced and costly methods for treatment and conditioning of the wastes will be introduced, and that such process systems, in order to obtain an optimum economy, should be large enough to process the waste, or rather certain types of waste, from several nuclear power reactors. It is probable that the best siting of such "central" process facilities would be at or near the repository. The introduction of a central waste treatment plant could raise some new questions with respect to the transportation system that have not been discussed within this study.

The majority of the solid waste from nuclear power stations is of low activity, partly of so low acticity that no particular safety precautions are required. In a large scale transport system it seems practical to use different types of containers for transportation of the waste packages. Some of the waste packages could be transported in standard containers, others would require containers corresponding to "strong industrial packages", and a minor part will require containers with shielding capability. A few packages may require special precautions.

6.3.2 TRANSPORT CONTAINERS

Use of containers with various properties, depending primarily on the radiation level of the waste packages, provides a kind of inherent safety in transportation: the higher the activity content, the heavier the container, and thus also the mechanical protection of the waste packages. In this discussion it should not be forgotten that the packages with waste incorporated in cement or in bitumen possess a significant resistance to damage due to mechanical forces.

Most transport containers with shielding capability are made from reinforced concrete and steel. Shielded concrete containers could typically have a wall thickness of 250-300 mm, a steel container with equivalent shielding capacity would have a wall thickness of 80-90 mm. Unshielded containers equivalent to "strong industrial packages" would typically be made of 8-10 mm steel plates with ribs for additional stiffness.

6.3.3 MODES OF TRANSPORTATION

The capacity, and hence the size, of the container depends on the mode of transportation.

The following carriers are all suitable for transportation of radioactive waste:

- ship
- railroad
- truck

The choice of carrier or mode of transportation will depend on several factors. All operating and planned nuclear power plants in the Nordic countries are located on the coast and have their own harbour facilities. If the respository is located close to the coastline, transport by ship may appear the most economic solution, but other factors such as transport length at sea compared to distances over land, water depth and ice conditions during the winter are factors that should be considered. Coasters and other small freighters may be suitable for transport of radioactive waste, but it can be worthwhile to evaluate design and construction of a special vessel for this purpose as has been done within the Swedish ALMA-study (see Technical Part I, chapter 5). This vessel is designed for roll-on/roll-off, permitting loading and unloading of the vessel without crane handling. At the same time, the vessel is suitable for transportation of other types of radioactive material such as spent fuel elements and high level waste from reprocessing. It is presently being evaluated whether that type of vessel would be suitable for transportation of other kinds of hazardous industrial materials. Such alternative use of the vessel may contribute to make a ship-based transport system more economical.

While ship transport can benefit from use of large containers with loaded weights of close to 100 tonnes, much smaller containers must be used if road transport is chosen. Typically, the over-the-road weight is limited to 30-40 tonnes (weight limits depend on actual transport route). Due to the short distance between the waste and the driver's cabin, it may be necessary to reserve 2-3 tonnes of the payload for a shielding arrangement behind the cabin. For road transport, normal trucks - semitrailers - are used. This provides a significant flexibility as there is no need to establish a special transport system, except for the containers.

Transportation by railroad offers the same flexibility as road transportation. An additional advantage is that containers could be significantly larger, perhaps as large as those foreseen for ship transport. Only few of the nuclear power stations in the Nordic countries have a railtrack connection, and so far only in Finland railroad transportation is considered.

7 DISPOSAL

7.1 INTRODUCTION

Disposal can be defined as release or placing of waste material in suitable long term storage conditions without the intention of retrieval /9/. The long-term hazard of certain types of radioactive wastes makes it impossible to rely on continued human supervision to maintain a containment and isolation of the waste. Therefore, there is a need for emplacing the waste in such a way that the requirement for human supervision is limited in time.

Even if the establishment of a disposal facility is delayed for different reasons, it is of great value to know at an early stage the requirements the repository will place on the waste products. Such a knowledge facilitates planning and operation of the waste handling system, and should ensure that the waste will be produced in its final form in proper time.

The factors important for evaluation of the safety of the disposal facility are discussed in this chapter.

7.2 OPTIONS FOR FINAL DISPOSAL OF REACTOR WASTES

The following options for disposal of reactor wastes are, at least technically, available in the Nordic countries.

1. Declassification
2. Shallow land burial
3. Man-made structures
4. Near-surface geological formations
5. Deep geological formations
6. Sea dumping

Some of the waste is hardly contaminated with radioactivity. The extra effort with administration, mesurement and sorting out of this waste will limit the use of the option of declassification of waste. But in special cases, for instance at decommissioning of a nuclear plant, a routine for a systematic exclusion of non-radioactive and slightly contaminated material would be worth while.

For shortlived and only slightly active waste the use of shallow land burial is common practice in many countries. In each particular case the climatic, hydrological and geological conditions need to be evaluated. A safety analysis will indicate the limits for permissible contents of various nuclides. For reactor waste a man-made structure in the surface layer of soil might be suitable. Such systems are used in the USSR and in France /10,11/.

None of the Nordic countries have so far established repositories for radioactive waste, but considerable investigations have been made through the recent years, in particular on disposal of high level waste.

In Denmark deep disposal of high level waste in salt domes has been investigated /12/. Although such a geological formation is also suitable for disposal of reactor waste, this type of repository has not been evaluated within the Nordic study.

In the other Nordic countries crystalline rock is available at or near the surface of the ground. The disposal of high level waste and spent fuel in deep rock formations has in particular been investigated in Sweden /1,2/. The understanding of the mechanism of water movement and nuclide migration obtained by these studies can also be applied to safety analyses of near-surface repositories.

The Nordic countries in general refrain from sea-dumping of radioactive waste and hazardous chemicals as part of a general policy to conserve the sea. This does not imply that sea-dumping of reactor waste is considered hazardous, but it does exclude sea-dumping as a realistic alternative for disposal of reactor waste at present.

Considering what has been mentioned above, the following three options are of prime interest in the Nordic countries:

- shallow land burial
- man-made structure (concrete bunker)
- near-surface geological formations (rock cavern)

These are all treated in the reference study.

7.3 BASIC CONDITIONS FOR SAFETY ANALYSIS

7.3.1 WASTE

The waste characteristics and their relevance for system and safety analyses are discussed in chapters 4 and 9.

As the waste in most cases is stored at the reactor site for several years before transportation to disposal the most shortlived nuclides have decayed. For the reference waste the external radiation level is determined by the Co-60, Cs-134, and Cs-137 contents.

For handling, storage and disposal operations, the radiation level of the waste packages is of importance. After the waste has been disposed, external radiation is not important, but longlived nuclides, which do not significantly contribute to the external radiation level, are of major importance for the analysis of the risk related to migration of radionuclides from the waste to the biosphere. In this connection a theoretical knowledge of the leaching rates, mobility and biological behaviour of the different nuclides is necessary.

Some practical aspects concerning the waste packages are also of importance:

- The sizes and the weight of the packages should be suitable for handling and placement at the repository
- The radiation level should be within the limits acceptable for the disposal site. If the shielding cask is needed the size and weight of the waste package should be adapted to this requirement.
- The strength and geometrical form of the packages should allow a safe stacking of the packages.
- The concentration and the total amount of the different radioactive nuclides should be within certain limits specified according to the safety assessment for the specific disposal facility.
- The leakage rate of the different nuclides from the packages should be within acceptable limits according to the analysis for the specific disposal facility.

7.3.2 VOLUME NEEDS

For the design of facilities for storage and disposal the volume of the waste that has to be taken care of is a key factor. For planning the handling and transportation system the maximum weight and dimensions of the packages are also important. The data must be defined in an unequivocal manner.

The volume of the waste can be given in different ways:

- untreated volume (as produced at the plant)
- volume after treatment for volume reduction
- conditioned volume (after solidification or fixation)
- packaged volume
- stacked volume, including empty space between packages
- storage volume, including transportation and handling areas, empty spaces between waste piles, roof and walls.
- plant volume, including access tunnels, service facilities etc.

An example can be given from the reference study.

For granular ion exchange RWCS resin from a BWR the respective volumes are:

	Annual production m^3/a
Untreated volume	250
Dewatered volume	12
Solidified volume (cementized)	12
Packaged volume (concrete mould)	21
Stacked volume	25
Storage volume (proportional)	43
Plant volume (proportional)	60

This example indicates how important it is to specify exactly the type of volume involved when talking about storage and disposal volumes.

7.3.3 MAN-MADE BARRIERS

Isolation from the environment of the radionuclides in the repository is provided by the combined resistance to leaching of the waste material and to migration through waste containers and other man-made barriers, and finally to migration through the natural barriers surrounding the repository. The general safety principles of redundancy and diversification could be applied here. That means that several simple barriers may be better than one very advanced barrier of a sophisticated type. The barriers should have different mechanical, physical and chemical properties in order to avoid that one single mechanism destroys all the barriers at the same time (so called common mode failure).

The barrier materials most frequently used are concrete, bitumen and clay. Steel plate is commonly used, but often not regarded as a barrier due to lack of information about rate of corrosion. For reasons of diversification a suitable combination of the materials might be useful.

The packages can be simply stacked in the repository. In that case the area exposed for leaching will be the sum of the surfaces of all the disposed waste packages. If, however, the spaces between the packages are filled with a material with as good resistance against migration as the waste package itself, the effective leaching area will be significantly reduced and comprise only the outer surface of the repository.

Concrete and different types of clay have been suggested as filling material and both materials will provide an additional migration barrier. While concrete will provide the best physical protection of the waste packages, clay has the advantage of being a plastic substance, at least when wet. A clay barrier can thus adapt to movements which fracture the repository and still maintain its role as a migration barrier. Clay also has the ability to seal crevices that may occur in the surrounding concrete structure or rock formation.

For both materials sufficient space between the piles must be provided to facilitate placement of the filling material. Horizontal layers of filling material at suitable intervals will provide further compartmentation and isolation.

For the safety evaluation a calculation of the release of radionuclides from the repository is required. For that purpose the type and the thickness of the barriers must be known, as well as the diffusion constants for relevant nuclides in the barrier material.

7.3.4 NATURAL BARRIERS

In selection of a site for a near-surface disposal facility a suitable geological formation is an important factor. In the Nordic countries different kinds of till, sedimentary clay and rock formations are available.

As long as access to the disposal area is prevented by supervision and fences, the only potential pathway of radionuclides to man is by ground water to a well or to a lake or to some other surface waterbody outside the fence. The activity in the reactor waste is dominated by nuclides having halflives of 30 years or shorter. As can be seen from the safety analysis, there should be no difficulty in choosing a site with an area large enough to allow these nuclides to decay before they could reach any point outside the controlled area.

To make such an evaluation the flowrate of the ground water or the ground water gradient and the permeability in the soil must be known.

Other information which is needed about the formation includes its porosity, homogeneity, and the retention properties of the geological media for the different radionuclides.

7.3.5 ENVIRONMENT

Different conditions with regard to the interaction between environment and the disposal facility are of relevance before and after the site has been left for unrestricted use.

During the operational period various events can cause the release of airborne activity and then the same factors that have been dealt with in previous chapters on storage and transportation are relevant for the safety analysis.

After the repository has been closed, but is still under supervision, only the hydrological and geological conditions are of interest. These have already been mentioned above in connection with natural barriers. When the site has been left for unrestricted use the possible human activities within the declassified area are of importance for the safety analysis.

8 SAFETY ANALYSIS

8.1 INTRODUCTION

Several alternate reference waste management systems have been evaluated in the present study. They are shown schematically in chapter 2, figure 2.1.

The purposes of the present safety analysis include an attempt at comparing alternate systems, and also at demonstrating how product characteristics can be related to parameters and scenarios considered.

The various components of the reference system chosen for analysis in the study are described in section 8.2. For further details on the reference system one should consult Technical Part I. For details on the analysis, beyond what is presented in the present chapter, see Technical Part II.

8.2 ACCIDENT SCENARIOS CONSIDERED

For storage, transporation and final disposal, a wide range of accident scenarios can be postulated.

The waste packages inherently have a certain mechanical resistance. They can withstand the impact from drops, and related mechnical vibrations occuring during normal storage and transportation conditions. The same applies to climatic conditions. Only repeated freeze-thaw cycles could cause problems, but these are or can be eliminated by short transportation times or by proper heating and ventilation of the storage facility.

War and sabotage related scenarios are omitted from further considerations. The related probabilities are difficult to assess, and these scenarios would furthermore be related to social or international disorders of much more serious consequences.

It has been demostrated in the Technical Parts that proper design and fabrication, including quality assurance procedures, can eliminate problems related to processes in the waste within the time periods envisaged for temporary storage.

Improper waste placements have been reported in literature: in one case waste has been dumped outside the disposal area. Rather than analysing such scenarios, which are not likely to occur in the Nordic countries, it is pertinent to stress the importance of proper administrative procedures, so that similar events may be avoided.

The stable geological conditions in the Nordic countries make most of the scenarios involving natural phenomena, such as earth quakes, tidal waves etc., irrelevant.

As a result the scenarios listed in table 8.1 ought to be considered. Some of these can be shown to be of minor importance, either by simple evaluations, or as result of experiments carried out.

Table 8.1 Events and processes relevant to release scenarios considered in the safety analysis

Table 8.1. a Storage

Event	Basic assumptions/comments
Bitumen fire	Fire in one of the storage pits where the cover is removed. Two layers, 8 drums, are involved. Fire fighting systems assumed to be not working.
Fire cementized waste	Experiments and simple analyses show that no release would be expected even if fire lasted for several hours.

Table 8.1. b Truck transportation

Collision, impact only	Impact velocity 80 km/h. Release fraction in the form of fine particles less than 1/1000 of total content, even when protection by heavy transport containers neglected.
Fall-in-water accident	One concrete mould lost in a river, with a flow of 10 m^3/s. The package is recovered after 30 days, or not recovered at all.
Collision, bitumen fire	The contents of one severely damaged container involved (8 drums). Fuel fire ignites the bitumen, and no effort is made to extinguish the fire.

Table 8.1. c Ship transportation

Event	Basic assumptions/comments
Wreckage	One concrete mould is lost and left undamaged on the sea floor.
"	One concrete mould is lost and left severely damaged on the sea floor.
Bitumen fire	Collision with tanker and subsequent fire.

Table 8.1. d Disposal

Migration in ground within institutional land use control period (0-200 years)	All radioactivity migrated from the repository reaches a well 100 meters away.
Man induced events after 200 years	Scenarios include a well drilled close to or in the repository, intrusion (dwelling or excavation), and farming (inhalation of dust and uptake in plants).

8.3 METHODS USED FOR SAFETY ANALYSIS

The general approach used for safety analysis is the same that is usually applied in reactor safety studies. Essentially this means that a chain of events is calculated or evaluated in the following sequence:

- magnitude and nuclide composition of release of radioactivity, as well as time-dependance of release and probability of release.
- transport of radioactivity from release point to humans, via numerous exposure pathways.
- calculation or estimation of doses and probabilities under many different conditions (e.g. different meteorological conditions) in order to calculate a risk spectrum.

No new analytical methods or computer codes were specifically developed within this study. Existing methods were used and if necessary modified.

The methods used for the different parts of the waste handling system vary in precision and flexibility. For some parts of the analysis well established methods were available, while for other parts it was necessary to perform extensive preparatory calculations in order to determine what methods were satisfactory.

Methods for calculation of the consequences of atmospheric releases were readily available, and using these methods it was also possible to calculate risk spectra. However, because the information on probabilities and magnitudes of the atmospheric releases is incomplete, these spectra do not give the complete risk picture, and the limitations should be kept in mind.

For some of the other parts of the safety analysis, particularly some of the disposal scenarios, it was not possible to estimate any probabilities at all; and since some of the parameter values were rather uncertain, it was necessary to perform conservative dose calculations.

Sensitivity analyses were performed to a limited extent in this study. Primarily, the extensive information needed for performance of a full sensitivity analysis was not time available within the study. What has been performed is a set of parameter variations, on parameters for which alternative values were readily available, and on parameters to which the results were felt to be sensitive. For one scenario sensitivity analysis using response surface technique was done.

Performance of a true sensitivity analysis is rarely possible. Such an analysis requires extensive information on how the uncertain parameters may vary. If this information is not available, it is possible to perform parameter variations. By choosing extreme values for some parameters, one may obtain hypothetical maximum and minimum doses. In other cases a parameter variation may be used to show that the results are relatively insensitive (or extremely sensitive) to the actual value of the parameter or parameters varied.

8.4 ANALYSES AND RESULTS

A short description of the various methods utilized is presented here, together with some discussion of data used, as well as some of the results of the calculations.

8.4.1 STORAGE

Normal events and processes in connection with storage do not contribute significantly to the risk. It has been demonstrated with the drop tests performed within the study, also with full scale drums with inactive ionexchange resin incorporated in concrete and bitumen, that the packages can resist even very strong impacts. Thus it is unlikely that normal handling of waste packages could lead to release of radioactive materials or to damage of packages that may influence the safety of waste management in later stages.

The environment in the storage may have some influence on the safety of later handling of the packages; particularly low temperatures and high humidity. While a few periods of sub-zero temperatures will have no influence on waste packages satisfying moderate quality requirements, a large number of freeze-thaw cycles may cause deterioration (cracking), in particular to cementized waste. High water content in the waste packages will tend to increase this effect. Hence, either the waste packages should be able to withstand such influence, or the number of freeze-thaw cycles should be limited.

For waste incorporated in concrete, corrosion of the steel drum is not critical to safety, although heavy corrosion should be avoided in view of ultimate handling sequences. Corrosion of steel drums containing bituminized waste may however be important, as bitumen is not formstable. Leaking drums may cause troublesome contamination of the storage facility, and later handling and transportation will be difficult and more hazardous.

Certain processes may take place in the waste packages themselves. Radiation and microorganisms may have detrimental influence on bitumen, but have little effect on concrete. However, even under unfavourable conditions, such processes have been shown not to have any significant influence within the time periods considered in this connection (5-50 years).

Another process that could be of importance is swelling of the waste, as that could cause breaching of the waste package. For bitumen some swelling (5% by volume) can probably be tolerated, as the drums normally are only about 90% filled. Significant swelling of simulated bituminized waste has been observed when the waste was submerged in water. Under storage conditions no scenario can be devised where swelling of bitumen would pose a safety problem.

Abnormal events during storage related handling operations that can have at least some potential for release of radioactive materials include collision or drop of a waste package during handling, and exposure to fire. These events have been analysed in some detail (see Technical Part II).

As mentioned before, drop tests have demonstrated that packages with solidified waste are extremely resistant to the forces resulting from fall. Tests with steel drums containing simulated waste incorporated in concrete showed only very small releases even when dropped from a height of 43 meters (corresponding to an impact velocity of 27 m/s). Drums with bitumen did not show any release at all. The tests also showed the fraction of dust size particles to be extremely small. The maximum impact velocity should, of course, be evaluated for each particular plant, but in most cases the impact velocity - in collisions as well as drops - will be considerably lower than 27 m/s (97 km/h). Hence it is unlikely that a handling accident could cause more than a negligible release of airborne radioactive particles. Since such an incident would take place in an area with controlled access, plant personnel and equipment should be well prepared to deal with such events.

However, packages that have been exposed to heavy impact in a drop or a collision may have suffered damages that could represent an increased hazard during later handling, transportation and disposal. Although these effects may not be very serious, such packages should be carefully inspected and, if necessary, be subjected to special treatment, for instance additional encapsulation or resolidification.

A fire in bituminized waste could lead to significant releases of airborne radioactive materials, while this is shown in the study not to be the case for cementized waste. The former case is described in more detail in the following.

Although experiments have shown that bitumen is very difficult to ignite, it is assumed that fire takes place in one of the storage pits where the cover is removed. The fire could accordingly involve 8 drums. It is further assumed that the building has been damaged, and that the airborne release has immediate access to the surroundings. The heat generated by the fire is assumed to be absorbed by the building, so that there will be a release of radioactive materials directly to the atmosphere, and with negligible thermal buoyancy. Other assumptions are mentioned in section 8.4.2.2.

Risk spectra as well as individual and collective doses for some selected weather conditions were calculated using the computer program CRAC, originally developed for use in the American Reactor Safety Study /13/. The risk spectra are based upon one specific release and one specific population distribution, and accordingly show the variation in consequence caused by weather conditions and differences in population distribution in different directions from the assumed release point. The spectra are also conditional; meaning that it is assumed that the release has taken place. If alternative release magnitudes, with corresponding probabilities, were taken into account, a broadening of the spectra would result. As the release used in the calculations is the highest release conceivable under the conditions used, the upper end of the risk spectra would change very little or not at all.

The meteorological and population statistics used to generate the risk spectra, apply to a specific site in the Oslo Fjord area, and the results are accordingly not generally applicable, nor necessarily "typical of Nordic conditions".

It was found that maximum individual doses in the range 0.01-0.1 Sv (1-10 rem) could be expected. These doses occur at ca. 500 meters from release point, and the doses diminish rapidly with increasing distance. At 5 km they are about a factor 50 lower. The actual doses depend upon numerous factors; e.g. weather conditions.

The very longlived nuclides that will dominate the doses from a disposal facility, are not included in the calculations for the storage fire or fire during transportation, as they do not contribute to the dose within the time frame of these calculations. With the reference waste composition it was found that of the important nuclides Cs-134, Cs-137 and Co-60 contributed about equally to the total doses while Sr-90 contributed only ca. 1%.

It is interesting to notice the calculated relative importance of the exposure pathways, as shown in table 8.2. Exposure during the time period immediately following the release is quite insignificant compared to the so-called chronic pathways; exposure to radioactive materials deposited upon ground, and exposure via nutrition. This information is especially important, since these doses may be influenced in a variety of different ways, if felt necessary. The mitigating actions may be e.g. decontamination of houses, limited condemnation of agricultural products, change of agricultural production to less critical products.

It must however be stressed that the maximum doses calculated for these abnormal events are all quite moderate, and should not require such actions.

Table 8.2 Percentage of dose received via each exposure pathway

Exposure via nutrition	56%
External exposure from materials deposited upon ground	44%
Inhalation of resuspended activity	0.2%
Inhalation from passing cloud	0.01%
External exposure from passing cloud	0.005%

The calculated risk curve is shown in figure 8.1. The probabilities in the risk curve are conditional; i.e. it is assumed that the release has taken place. Furthermore they are complementary cumulative, which means that the probabilities shown are probabilities that the related consequence or a larger consequence will result. We may e.g. read from figure 8.1 that there is a probability 0.01 that the collective effective dose would be ca. 25 manSv (2500 manrem) or larger. It should be stressed once more that using the collective dose to express the risk, is just one of many possible ways, of which the most correct would probably be to use the health effects as a basis.

Although the calculated doses are quite moderate, they also show the need to ensure that a fire cannot involve a large number of packages with bituminized waste. This justifies the common practice for design of a storage facility for bituminized waste: the storage is divided into fireproof cells each containing only a limited number of waste packages. It should be pointed out here that drums with bituminized waste must be exposed to high temperatures over an extensive period of time (of the order of 20 minutes or more) to start a fire.

The acceptability of the calculated doses will, of course, depend upon the probability of such an accident. Attempts were made to estimate the causes of a fire in the stored bituminized waste and the related probability. The only conceivable causes that have been identified are:

- sabotage
- airplane crash followed by fuel fire
- large forest fire
- overhead crane failure

These causes are discussed in detail in Technical Part II, and are summarized below.

Sabotage is not considered here, as an evaluation of sufficient depth to be meaningful is outside the scope of the present study.

The probability of a plane crashing on a reactor building has been estimated to be less than one in ten millions in the Nordic countries.

The risk represented by a forest fire could easily be controlled by establishing a deforested fire belt.

No way can be seen how even a severe failure of the overhead crane could cause ignition of the bituminized waste.

The general conclusion of the reference case safety analysis is that the risk related to storage of solid or solidified radioactive waste is very small. Except for a fire in bituminized waste no process or event has been identified that could lead to release of significant amounts of radioactive materials to the environment. All other incidents may have consequences only to the operators and other personnel employed at the power plant. Although calculation of doses to the operators has not been performed in this study, it seems unlikely that significant doses may be contracted.

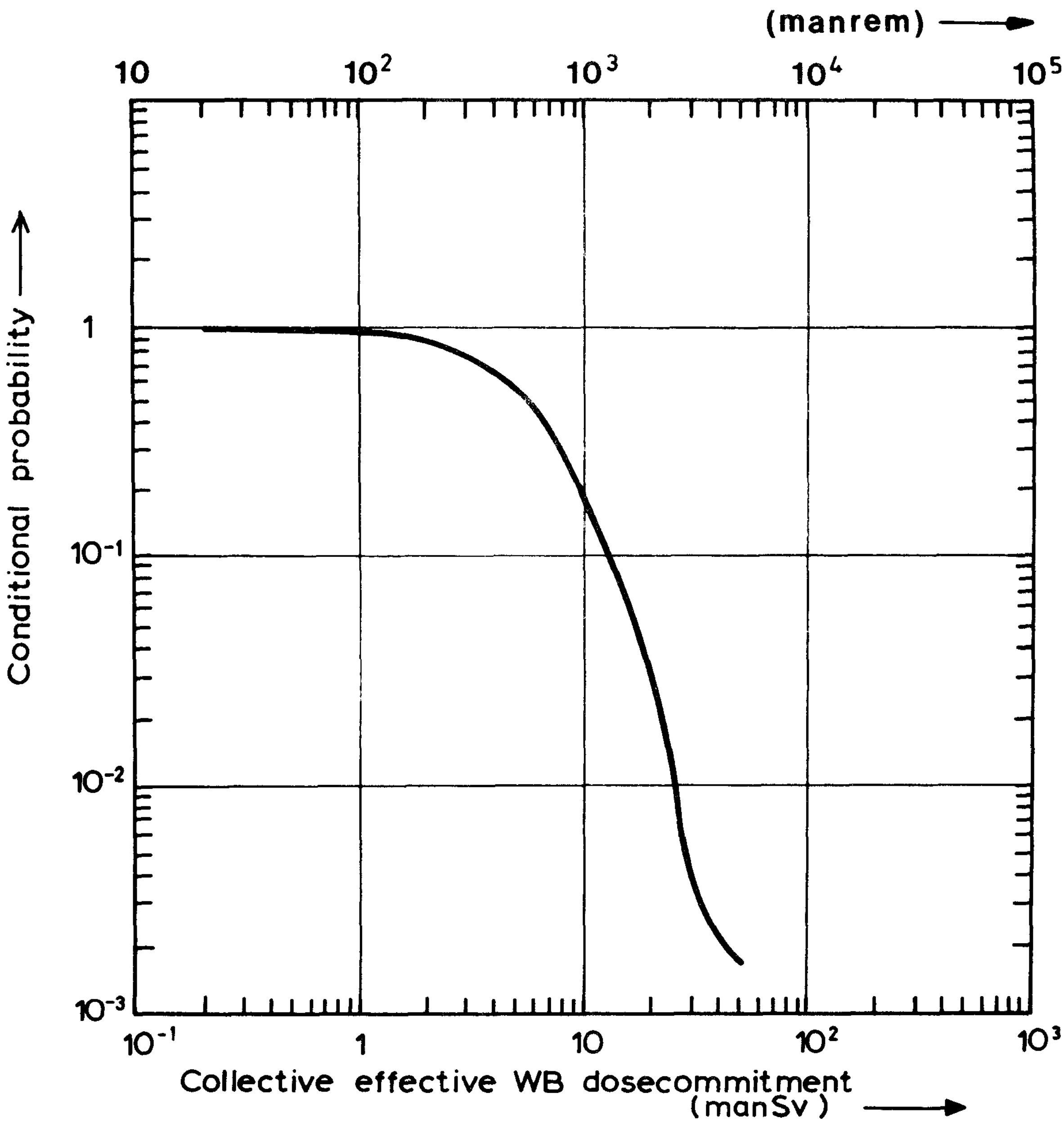


Figure 8.1 Calculated risk curve. Bitumen fire in storage.

8.4.2 TRANSPORTATION

Most of the normal events and processes to which the waste packages may be exposed during transportation are rather similar to those relevant to storage, and those discussions are not repeated here. Abnormal events such as drops and impacts were in principle described in the pervious chapter, and do not justify analytical considerations. Among the normal events are also climatic influences. Rain and sub-zero temperatures will not damage the waste packages. Only repeated freeze-thaw cycles could possible cause damage, but experiments carried out have shown the packages to be resistant also under these conditions. The cases most important for the transportation stage were identified as the following.

- fall-in-water accident
- bitumen fire
- ship wreckage

The probability of a transportation accident during transportation with truck was found to be of the order of 10^{-9} per transport kilometer. The probability of a release of radioactive materials will of course be lower, and is discussed in the following sub-chapters.

8.4.2.1 FALL-IN-WATER ACCIDENT

Truck accidents may involve loss of the payload in a river or lake. Immersed in water, radioactive materials may be released from the waste and ultimately cause exposure of humans, either via drinking water or nutrition. Severe damage to the waste packages will tend to increase the release rate, while an intact container will protect the waste packages against contact with water and delay release for a considerable period of time.

The previously mentioned drop tests show that it is very unlikely that a steel drum may be punctured, even when hitting a pointed object. A drop could however cause a concrete mould to crack. Only the concrete mould is accordingly analysed in relation to the fall-in-water accident.

Three different cases have been analysed:

- Case 1. Mould damaged so that the complete, but intact waste package is in contact with water. Recovered after 30 days.
- Case 2. As above, but not recovered.
- Case 3. As above, but in addition extensive damage to waste package. It is assumed in this case that there is a pulse release of 10% of the cesium content, and this pulse is found to dominate the doses.

The last case might be the result of drop from a high bridge or down a steep slope.

Individual and collective doses are calculated using the computer programs BIOPATH /14/.

The river in which the mould is assumed to fall is quite small, with a flow of 10 cubicmeters per second. Individual doses will be roughly inversely proportional to the water flow, while collective doses as calculated by BIOPATH are almost independent of the water flow. The reason for this lies in the way the calculation is performed. In the present calculations it is assumed that 2% of the activity flow is transferred to agricultural area via irrigation. Furthermore the terrestrial nutrition pathways are dominating when the collective doses are concerned. Accordingly the water flow in the river will have little influence on the collective doses.

A number of exposure pathways are included in the calculations. It is found that fish, milk and other agricultural products are the most important pathways.

Site specific data used in the calculations are valid for an "average area" in Sweden, and e.g. the fresh water surface, agricultural area and population density are given average values. The collective doses, as calculated, are dominated by the population on the Baltic Sea, rather than the population in the local or regional area. If the population distribution used in these calculations had been the same as the one used in the bitumen fire cases, there is however a possibility that the collective doses would be dominated by the local or regional population. The reason is that the average population density, used in the fall-in-water accident calculation, is quite low (20 persons per squarekilometer), compared to the inhomogeneous population distribution used in the bitumen fire calculations.

All calculated annual individual committed doses are lower than 10^{-5} Sv (10^{-3} manrem), and all collective dose commitments are lower than 0.05 manSv (5 manrem).

The probability of a transportation accident resulting in the immersion of the payload is estimated to be 1/10, and the probability of the immersion resulting in exposure of the waste to water is estimated to be 1/10. Accordingly the probability of an accident resulting in release to water is 10^{-11} per transport kilometer.

The total transportation distance (empty runs not included) for the reference waste (RWCS and SFPCS) from 30 years of operation of 6 reactors (500 MWe), is 288,000 kilometers for cementized waste in moulds. The probability of an accident resulting in release to water is accordingly about 10^{-7} per year for the reference system containing 6 reactors.

8.4.2.2 BITUMEN FIRE

A traffic accident involving a fire may cause ignition of the bituminized waste. When the waste is incorporated in concrete, a fire will have little or no effect. Another conceivable consequence of a collision is fragmentation of the waste package. It is found that this is of no effect for bituminized waste, and that fragmentation of cementized waste also may lead only to doses many orders of magnitude lower than the ones that may result from a bitumen fire. Accordingly only the bitumen fire is analysed.

For transportation by truck, the drums are placed in transport containers made from reinforced concrete. The containers take 8 drums each. The accident is assumed to breach one container only, and to involve an oil and gasoline fire lasting long enough to ignite the waste/bitumen mixture in the 8 drums. Experiments have indicated that a fire in bituminized waste does not start unless the bitumen is exposed to an oil fire for 15-20 minutes /15,16/.

It is assumed that the fire will last one hour, and that the release of radioactive materials is uniformly distributed over this time period. Furthermore it is assumed that 100% of the cesium and 85% of the other nuclides are released to the atmosphere. This is a conservative assumption according to recent experiments. 30% and 5% may be more realistic under reasonable circumstances.

Calculations are performed as for the fire in storage, described in chapter 8.4.1, and most of the data are the same. The important differences are the amount of radioactive materials involved and the effect of the heat generated by the fire. Waste when transported will have an additional five years of radioactive decay compared to the waste assumed to be involved in the storage fire. Furthermore the heat generated by the fire will give the release a thermal buoyancy, and lead to lower radiation doses close to the site of the fire, and higher doses further away, as compared to the storage fire case.

The individual doses will vary with distance and weather conditions, but the maximum dose is about 0.01 Sv (1 rem), all exposure pathways taken into consideration, and integrated over all future (as a result of the life-times of the nuclides of importance, this is effectively about 50 years).

The risk is also calculated, as for the storage fire, and the risk curve is somewhat lower in the transportation case (mostly due to the extra 5 years of radioactive decay). An additional calculation has been performed, for the postulated case of the transportation accident taking place in the middle of a city of half a million inhabitants. The individual doses are of course the same, but both the risk and the maximum consequences are higher. The risk curves are included in Technical Part II.

The risk curves are conditional, like figure 8.1, i.e. it is assumed that the release has taken place. To know the absolute probabilities, the probabilities in the risk curves must be multiplied by the probability of the release taking place. A study performed at the Sandia Laboratories /17/ gives the probabilities of road transportation accidents resulting in fires of different duration. The probability of a road accident resulting in a fire of about 20 minutes duration is about 3×10^{-9} per transport kilometer.

8.4.2.3 SHIP WRECKAGE

Wreckage of a ship carrying radioactive waste may result in a loss at sea of from a single waste package up to the total payload. As the ship wreckage accident scenarios were relatively recently analysed in a Swedish study /14/, it was not considered necessary to perform an independent analysis in the present study.

Two of the accident scenarios are considered as being of dominating importance:

- One concrete mould is lost and left undamaged on the sea floor, and the radionuclides are released to the sea by diffusion.
- One concrete mould is lost and left severely damaged on the sea floor. The total activity is assumed to be released to the sea at a steady rate over 6 months.

It is assumed in the calculations performed that there is total damage to the transport containers. Undamaged containers will effectively prevent release from the waste packages for very long time periods, permitting recovery of the waste. Furthermore it should be mentioned that a release period of only 6 months is rather unrealistic.

Other possible accident scenarios are in connection with a bitumen fire aboard the ship. The containers have very good fire resistance properties and, if intact, will protect the drums for more than 24 hours in case of a fire in the cargo hold of the transport vessel. A bitumen fire could conceivably take place in one or a few containers, if these were damaged during a collision, and the collision was followed by a fire. Many or all containers might be involved in a bitumen fire if an extensive and lasting fire takes place. Such a fire is only credible in connection with a collision with a tanker carrying crude oil or gas. In ref. /4/ it is claimed that the probabilities of both of these scenarios are so low that they need not be considered further. Further there is reason to believe that the risk is considerably lower than the risk from a fire following a truck collision, as only the crew is involved if fire in a few containers takes place at sea.

However circumstances might be imagined under which the risk might be significant, such as an accident which takes place while the vessel is close to the coast. If all

containers are involved in a fire the doses to the crew, and accordingly the risk, might become rather high. These cases ought perhaps to have been considered.
No calculations have been performed for the ship wreckage case in the present study. The results are taken directly from ref. /4/, multiplied by the proper factors to correct for differences in nuclide content. Accordingly the results presented for the ship wreckage case here and in Technical Part II are valid for the conditions used in ref. /4/. Of particular interest is the population distribution used for calculation of the collective doses. The calculations in ref. /4/ are performed assuming that the accident takes place in a position about 5 kilometers off-coast of a town of about 10,000 inhabitants in the Southern part of the Baltic Sea.

In the worst case for ship wreckage considered the individual committed doses are below 10^{-5} Sv (10^{-3} rem), and the collective dose commitments are lower than 0.1 manSv (10 manrem).

8.4.3 DISPOSAL

As mentioned in chapter 2, three different disposal concepts are considered in this study; shallow land burial, near-surface concrete bunker and rock cavern. These three repository concepts are considered in various combinations with three types of geological formations; sandy till, clayey till and crystalline rock.

There are several basically different time periods in the "history" of a disposal facility. In this study these periods have been defined in the following way:

- Period of operation. During this period the repository is successively filled with waste, and at the end of this period the repository is closed. In this study it is assumed that the operation period will last 30 years.

- Period of supervision. During this period unauthorized access to the site is prevented by physical means, such as fences or guards. The period of operation is included in the period of supervision.
- Period of institutional land use control. This period is also defined in such a way that the previous period, the period of supervision, is included. But after the period of supervision has come to an end, there is a change in situation. It is envisaged that there will at this time be free access to the area, but that institutional control will prevent all types of construction work, building of houses, digging trences for pipe-lines etc., unless these activities are known to entail negligible risk. The period of land use control is assumed to end 200 years after closure of the repository.
- Period of unrestricted use. In effect it is assumed that when this period starts, at the time when the period of land use control ends, the presence of the repository is totally unknown. This may be unlikely, but is the worst case imaginable.

Passive marking of the area (inscription on a stone or metal plaque) might be a way in which to enhance the efficiency of a land use control system; and might also serve to identify the site after restrictions have ceased. In the present study it has been assumed that the land use control is efficient, but after this period any benefit from e.g. passive marking has been disregarded.

8.4.3.1 RELEASE SCENARIOS

No analysis for the operational period of a disposal facility has been needed. The possible release scenarios will be very similar to the ones for storage, and neither concequences nor probabilities will be significantly more unfavourable.

During the period of instutional land use control there will only be one type of release scenarios, and those are the ones caused by natural processes. When a repository is closed, the ground water level will gradually rise to reach its natural level in the area. Water will gradually fill the repository, and will eventually penetrate the waste packages and matrixes, though this may take very long time. The radionuclides may sooner or later be dissolved, leach out of the repository, and be carried along with the ground water movement. Once the nuclides are outside the repository, they may in due time reach humans, via several possible pathways, and cause radiation exposure.

After start of the period of unrestricted use the types of release scenarios mentioned in the above are relevant, but in addition release scenarios of another type must be taken into consideration. These involve human activities of one type or another, like drilling of new wells, building of houses, performing other types of construction work, or using the area for agricultural purposes.

It has been assumed that natural events like earthquakes or flooding do not occur. It should not be difficult in the Nordic countries to find areas where such events are extremely unlikely.

8.4.3.2 CALCULATION METHODS AND DATA

No really well-established methods for calculation of these release scenarios have been available. The calculations have been performed using a combination of available computer programs, and as in most complex calculations, a number of simplifying assumptions have been adopted. Several of these assumptions are very conservative.

One problem of particular importance in connection with leaching from repository and migration through soil or rock, is the difficulty of determining in what particular chemical form the various radionuclides occur. The data used in calculations of these processes are particularly concervative, as it has been assumed, when the opposite can not be proven, that the radionuclides have no chemical bindings; i.e. is that they are always completely dissolved in water.

Descriptions of methods and data are included in Technical Part II.

8.4.3.3 RESULTS

All results of the disposal calculations are expressed as maximum annual doses. This means that a dose value is the highest individual committed dose over a one-year period. There are a few scenarios with external radiation exposure. In these cases dose values given are doses actually received during the one-year period. Most scenarios, however, involve intake via water, nutrition or inhalation, and in these cases the dose value is the total dose resulting from intake during the one-year period, though the dose received is actually distributed over a much longer time period, or even the remaining life-time.

Dose variation year by year has not been calculated. Multiplication of the dose value given by 50 years to obtain life-time dose is not correct, and will be conservative to a varying degree, depending upon the circumstances. Added to this will be the effect of using conservative methods and data. This must be remenbered when comparing these maximum annual doses with doses resulting from an accident in storage or during transportation.

For most of the disposal scenarios the dose is dominated by nuclides with halflives of many thousands of years. As a result of this, there are two major problems connected with calculation of a collective dose commitment:

- what will the population be in a more or less distant future, and how will life-style change? (The latter is particularly important in connection with calculation of the collective dose commitment.)
- how far into the future should integrarion of doses be carried when calculating the dose commitment?
  For very longlived nuclides the dose commitment can not be calculated taking only radioactivive decay into account. The elimination rate from the biosphere must also be considered.

In this chapter doses per year only are presented. In most scenarios they are doses throughout that one-year period when they reach maximum. In the case when long-lived nuclides will build up, however, which is the leakage to the Baltic Sea scenario; it has been chosen to truncate the calculation at 500 years. It is assumed that the leakage is constant, at the maximum rate, throughout 500 years, and the dose commitment due to intake during the 500'th year is calculated. This is not necessarily the maximum, as the build-up (if elimination from the biosphere is neglected) will continue.

In connection with natural processes (repository undisturbed by human activities), the following release scenarios and/or exposure modes are relevant:

- direct radiation exposure of a person staying for extended time periods on top of the repository.
- leaching and migration of radionuclides to a well near repository, followed by consumption of this water.
- leaching and migration of radionuclides to a lake or sea, followed by exposure via many alternative exposure pathways.

After the period of supervision, it has been supposed, as the worst case, that a person chooses to camp on top of the repository throughout a whole summer vacation. Calculations show that the resulting doses would be negligible.

Migration to a well, and to a lake or the sea are interrelated, as it is assumed that water from a well, once used, is brought via the sewage system to the sea. The well scenario gives the highest individual doses, but the collective doses for the different scenarios are found to be almost equal, as these are in all cases dominated by doses to the large population group exposed via the sea.

During the period of supervision it is assumed in these calculations that there exists a well outside the site area, at a distance of 100 meters from the repository. When the area is left for unrestricted use, a well might be drilled adjacent to the repository. (It is assumed that it is not drilled right into the repository, as one would observe that a concrete structure has been hit.)

When the well is 100 meters from the repository, the maximum annual individual dose calculated is encountered when it is positioned near a shallow land burial repository. This dose is significant, as it is roughly ten times the natural background radiation. But again the conservative assumptions and data should be remembered. It is, for instance, assumed that all radionuclides leaching from the repository reach the water in the well. It is also assumed that all radionuclides are dissolved as soon as water fills the repository, that the repository is filled with water immediately after closure, and that there is no retention of carbon or iodine in soil. All individual doses calculated for this scenario and the well at 100 meters distance are dominated by carbon-14.

The doses calculated for the rock cavern and concrete bunker cases are much lower than for shallow land burial. It makes little or no difference whether the repository is intact or fractured, except in the rock cavern case, where fracture of the brittle part of the barrier increases the doses by a factor of ten. No calculations have been performed for bitumen drums for this scenario. The differences that might be expected would be caused by differences in chemical state and the leaching properties of the waste products.

When the area is left for unrestricted use, a well might be drilled right next to the repository. The individual dose from a well adjacent to a fractured repository is 0.4 Sv/a (40 rem/a). Even though this is a maximum dose, it will not differ that much from one year to the next. The dose is entirely dominated by cesium-137 (half-life about 30 years). In this case the difference between shallow land burial and concrete bunker is not so large, about a factor of seven for intact repository, and two for fractured repository; shallow land burial giving the highest doses. The doses from rock cavern are roughly four orders of magnitude lower.

It should be mentioned here that in the calculations it is assumed that cesium, but none of the other radionuclides, is retained by the soil, and the data used correspond to a migration time of the order of tens of thousands of years over a distance of only 100 meters. This illustrates the importance of the retention assumptions.

In connection with disturbances in the site area caused by human activities, the following release scenarios and/or exposure modes are relevant:

- dwelling on top of repository.
- intrusion in the area by excavation, giving external exposure and exposure via inhalation.
- agricultural activities, giving exposure via inhalation and via nutrition.
- drilling a well adjacent to the repository.
- causing fracturing of the repository.

The last two cases were already dealt with in the above, because of their close connection to the case of a well 100 meters from an intact repository.

Only at a shallow land burial site can dwelling permanently on top of the repository give doses of significance. If there is a soil cover of 2 meters thickness, the life-time individual dose would still be insignificant, but if the soil cover is completely removed, and the waste packages themselves are exposed, the dose would be unacceptable, of about 10^4 Sv/h (0.01 rem/h). 0.5 meters of concrete (as assumed on top of the bunker) would reduce this by a factor of thousand. It might accordingly perhaps be reasonable either to reduce the amount of radioactive materials so that the doses will be acceptable, or alternatively to require a concrete lid on shallow land burial repositories. The methods used for dose calculations in this scenario, though simple, are quite dependable, and there is no reason to expect these doses to be significantly conservative.

By excavation in the area, soil cover might also be reduced or removed, and external radiation doses in the same range as for dwelling might be encountered. In this case a concrete bunker might also be uncovered. The doses would however always be lower by about a factor of thousand, because of the concrete bunker wall. In the excavation scenario exposure via inhalation would, however, be additional. Two different modes are examined; blasting and digging, the former giving a dust concentration in air double the one caused by the latter. It is only the radioactivity that has leached out of the repository that may be involved, and the doses will depend upon the time since closure of the repository. The inhalation doses are in all cases negligible. The only reason for difference in doses from shallow land burial and concrete bunker will be due to differences in leaching. All doses in the excavation scenatio are dominated by cesium-137, in the calculations done for this study.

Two exposure pathways may be important in connection with farming in the site area; inhalation and ingestion. In this scenario also only activity that has leached out of the repository may be involved, and this activity is assumed to be homogeneously distributed in a soil volume of length equal to the migration distance at the relevant time, width equal to the smallest horizontal dimension of the repository, and depth equal to the distance from ground surface to bottom of repository. This scenario is not relevant to rock cavern, because a rock cavern is assumed to be placed at a much larger depth than the other two alternatives. The inhalation doses are found to be quite small. Of the nutrition pathways only the vegetable pathway has been included in these calculations, and the total dose with all nutrition pathways taken into account could be a factor of two to five higher. But this underestimation of total dose is probably more than counteracted by other conservative assumptions and data. The vegetable pathway doses are found to be small, but significant (more than a factor of ten thousand higher than the inhalation doses). The main contributor to the dose is cesium-137, but strontium-90 is also important.

9 PRODUCT CHARACTERISTICS IN RELATION TO SAFETY ANALYSIS

9.1 INTRODUCTION

The fixation of the waste in a suitable matrix material provides one of the barriers against the release of radioactivity to the biosphere. It is a complicated task to define feasible characteristics for the solidification products as a function of their relative importance in the management system. The relevance of the product properties will differ, according to the specific requirements at each step of the handling system.

Techniques are available to meet even very strict requirements on low- and intermediate level waste solidification products. Consequently, in the absence of defined integrated plans for the whole waste management cycle, and in want of criteria for product specifications, the need to design each step with an adequate safety margin can impose unnecessarily complicated solidification procedures.

Apart from high costs, too complex solidification techniques can even have such adverse effects as increased susceptibility towards technical process variables and possibilities for radiation exposures to the operators. Safety margins can also be increased by reducing the waste concentration (ratio of waste/matrix), but this approach may give rise to undesirably large waste volumes and production costs.

An important result of this study is that only a few of the many product properties seem relevant to the safety assessment. This is fortunate, since available results from small-scale laboratory tests performed during relatively short time periods can not easily be interpreted with respect to the long-term performance of full-scale products. The properties of the solidified products will also be influenced in a not too well-known manner by a number of process variables and by the characteristics of the waste itself and the matrix material. Assessments of these questions are presented in Technical Part III.

9.2 RELEVANT PRODUCT PROPERTIES

The most important characteristic in judging potential risk levels related to the properties of solidified waste is the retention of radioactive materials in the final product. The release of radioactivity from the product has been examined almost exclusively by studying the leaching of radioactive nuclides out of the product submerged in water. Swelling of the products and degradation of the mechanical properties by contact with water are also important.

Radioactive materials can also be released in case of breaking, cracking or crushing of the products. This kind of mechanical degradation may be due to external pressure, heating or freezing, mechanical stress or shock, radiation, or by chemical interactions with the environment.

As described in chapter 8, the most important properties during waste storage and transportation are related to mechanical strength, fire and freeze resistance. Water and leach resistance are the crucial properties in disposal.

Table 9.1 lists properties of solidified wastes or of waste products plus container, according to their potential relevance in specified normal and abnormal events.

Table 9.1 Waste properties of interest at different abnormal and normal events
B = bitumen, C = cement
x = effect
- = minor effect
blank = irrelevant

PROPERTIES	Mechanical stability		Impact resistance		Temperature resistance						Water resistance and leach properties		Effect of micro org.		Radiation resistance	
					Low T		Elevated T		Burning prop.							
	B	C	B	C	B	C	B	C	B	C	B	C	B	C	B	C
ABNORMAL EVENTS																
1. Collision, drop	x	x	x	x	x	-	x	-	x	x						
2. Fall in water acc.	x	x	x	x	-	-					x	x			-	-
3. Fire	-	-	x	x			x	x	x	x					x	-
4. Damage of container	x	x	x	x	-	-	-	-					-	-	-	-
6. Admin.errors	x	x	x	x	-	-	-	-	-	-	x	x	x	x	-	-
7. Retrieval	x	x	x	x	-	-	x	x	-	-	x	x	-	-	-	-
NORMAL EVENTS																
1. Piling, storage	x	x	x	x	x	-	x	-	-	-	-	-	-	-	-	-
2. Transp.,handl.	x	x	x	x	-	-	-	-	-	-	x	x	-	-	-	-
3. Degrad.of barr.	-	-	-	-	-	-	-	-	-	-	x	x	x	x	x	x
4. Disposal	-	-	-	-							x	x	x	x	x	x

The full scale tests simulating fall and collision accidents have shown that the release of radioactivity in all cases is very small, even for waste products without a container.

Containers filled with cementized or bituminized waste were found to be temperature and even fire resistant, but bituminized products without a container can be vulnerable to fire. Apart from potential effects of burning properties in case of a bitumen fire, the safety analysis has not identified any impact of product properties on possible radiation doses for waste storage and transportation.

Some effects of leach properties on radiation doses from disposed waste products have been quantified in chapter 8. But in spite of very conservative assumptions about leach coefficients, radiation doses from leached main nuclides (Cs-137, Sr-90, Co-60) were in most cases found to be insignificant, mainly due to long retention time in the surrounding geological barrier. Except in the extreme case of a well drilled directly outside the repository, dose contributions were exclusively from long-lived nuclides (C-14, I-129, Cs-135), mainly from C-14 and I-129. It should, however, be noticed that the relatively high doses from C-14 and I-129 are based on accumulated worst estimates, i.e. on the use of extremely conservative leach rates and not taking into account retarding effects such as of chemical reactions, isotope exchange reactions and dispersion. With somewhat more realistic assumptions (see table 9.2 on p. 72) individual radiation doses even from these nuclides will probably be insignificant.

According to the safety analysis the steel or concrete containers and the engineered and geological barriers in the repository provide the main protection against spreading of radioactivity. This reduces the influence of product properties such as mechanical strength, leach and temperature resistance on possible dose commitments from the different stages of waste management. On the other hand regard to possible interactions with the outer barriers can shift the emphasis towards properties other than those considered for unprotected products. In this connection non-corrosiveness and slight tendency to swelling are relevant. The latter might be of special importance for ion exchange waste.

Regardless of identified demands on product properties, the additional safety gained by a fixation of the radioactivity in well defined stable products will probably have to be maintained. To meet this requirement it is important that:

- product qualities are well defined and reproducible.

- the solidification process has wide tolerance limits towards technical process variables, so that acceptable homogeneities can be ascertained.

- physical and chemical reaction mechanicms are so far understood that reasonable evaluations of long term stabilities are possible.

Table 9.2 Leach (diffusion) coefficients

Nuclide	Matrix	Leach coefficient (m^2/a)		
		Safety* analysis	Recommended** conservative	More realistic
Cs-137 / Cs-135	Cement	$3x10^{-4}$	$3x10^{-4}$	$3x10^{-5}$
	Bitumen	$1x10^{-6}$	$1x10^{-6}$	$1x10^{-7}$
C-14	Cement	$3x10^{-4}$	$3x10^{-6}$	$3x10^{-7}$
	Bitumen	$2x10^{-6}$	$3x10^{-6}$	?
I-129	Cement	$6x10^{-4}$	$6x10^{-6}$	$6x10^{-7}$
	Bitumen	$2x10^{-6}$	$6x10^{-6}$	?
Ni-63 / Co-60	Cement	$4x10^{-9}$	$4x10^{-8}$	$4x10^{-9}$
	Bitumen	$4x10^{-9}$	$4x10^{-8}$	$4x10^{-9}$
SR-90	Cement	$7x10^{-6}$	$7x10^{-5}$	$7x10^{-6}$
	Bitumen	$3x10^{-9}$	$3x10^{-7}$	$3x10^{-9}$

* Technical Part II

** Technical Part III

9.3 TEST METHODS

In order to ensure that satisfactory operation of the waste immobilization process is taking place, control and testing of the products at the plant site will usually be required. This is especially important in the absence of full and detailed characterization of the physical and chemical nature of waste to be solidified. The need to ensure that the waste immobilization process is taking place satisfactorily and also that the products themselves are suitable for further handling, will lead to different test routines and methods. Four lines of tests will be needed for

- control of materials to be used in solidification operations
- simple control of product properties at the plants site
- laboratory control of products, parameter studies, aimed at evaluation of the impact and tolerance range for process variables
- safety assessments and prediction of long-term product performance

A great variety of test methods are available, as outlined in table 9.3. Some of them have been adapted from test catalogues for "conventional" products, such as concrete, bitumen and plastics. Others have been developed for relative studies and are not always sufficiently standardized to ascertain comparable results from other sources.

All tests are of relatively short duration and test conditions differ so much from actual waste management situations that it is difficult to establish quantitative correlations with the actual performance of full-scale technical products.

Table 9.3 Tests for characterisation of bituminized (B) and cement (C) waste products (see Part III, chapter 4).

Properties	Test	Matrix
Mechanical strenght	Compressive strengh	C
	Tensile strenght	C
	Impact resistance	B,C
	Fall tests	B,C
Form stability ductility	Penetration	B
	Ring-ball test	B
	Sag test	B
	Hole migration	B
	Cylinder bending	B
Thermal stability	Break point	B
	Softening point	B
	Melting point	B
	Flash point	B
	Heating tests (e.g. 105°C)	C
	Fire tests	B,C
	Freeze/thaw cycling	B,C
Radiation stability	Gas evolution from irradiated samples. Effects on mech. properties, water and leach resistance	B(C)
Water resistance and chemical stability	Qualitative immersion tests	B,C
Leach behaviour	"Longterm" leaching of radionuclides in deionized water and in representative ground water	B,C

The adaption of all available tests to the routine control of technical products would be prohibitive for any specific solidification plant. Such control programmes must be limited to a few simple and reliable tests for relevant properties and should be based on statistical assessments of product homogeneities and impacts of process variables. The need for strict control of matrix materials is emphasized by the observed impact on product properties (Technical Part III, chapter 5).

The fact that utilities must be able to demonstrate that the management systems are reliable and "safe" both in the short and long term, leads to the requirement to develop and standardize testing methods which can provide such reassurance. This requires an analysis of objectives during each waste management phase and consideration of chemical and physical processes which govern releases of radioactivity. Then the required product stability in each phase will need to be determined, which in turn leads to requests for relevant characterization methods. In establishing such a scheme, possible interactions with the container and with the disposal site must also be evaluated.

9.4 LONG-TERM ASPECTS

The leaching of radionuclides in contact with water is the main means for radioactivity release from the repository provided a direct intrusion can be eliminated. In this connection the longterm water- and leach-resistance are of prime importance. Other properties such as mechanical strength, resistance against bacteriological degradation and other natural events are also important for the maintenance of the leach resistance. In this connection even minor damages during storage and transportation can have delayed effects on the longterm performance in the repository. Requirements for resistance during normal storage and transportation will therefore partially be governed by those for the longterm product performance.

The immobilizing properties of the solidified waste will probably degrade more rapidly than the engineered and geological barriers. They may be prolonged if efforts are made to achieve a chemical binding of more longlived radionuclides in the matrix material. Improved properties may be obtained with new production processes or better understanding of the behaviour of available products, but in all cases the assessment of the risk level should be based on the overall disposal system.

10 CONCLUSIONS

It has been demonstrated through the study that methods and data needed for performance of a safety analysis are available. Not all methods and data are, however, equally satisfying, and in particular for some release scenarios in disposal facilities it is evident that better methods and/or data should be found. Knowledge about the normal behaviour of some important long-lived nuclides is also insufficient.

The ultimate aim of a safety analysis is to compare the risk in the different parts of the management system, or between alternate solutions of the same part of the system. Critical parts of the system can thus be identified, and be modified if necessary; or the best of alternative solutions may be chosen.

But comparisons are only possible if the risks can be expressed in identical manner in the different parts of the system. This may often be very difficult, as also in the present case.

The results calculated in the present safety analysis vary much in type, due to differences in methods, data and general approach; and the doses calculated for the different parts of the management system are accordingly not directly comparable. The doses calculated for the disposal scenarios particularly stand apart. This is mainly due to the fact that doses from these scenarios are dominated by very longlived nuclides, combined with the fact that the doses are calculated using simplified methods.

The well scenarios may serve as illustration: it is assumed that the well just gives sufficient supply of drinking water, which means that there will be no accumulation of radioactive materials in the well. They are consumed as they seep into the well. For long-lived nuclides and a constant leak rate into the well, the individual doses will be about the same from year to year. After consumption the radioactive nuclides will go with waste water to some water recipient (in the present calculations the Baltic Sea). Here there will be accumulation, and if no nuclide-removing mechanisms are taken into account, the collective committed dose after 500 years of this release will be almost 500 times the committed dose from the first year. There are, however, content-reducing mechanisms in addition to radioactive decay. Transformation of the nuclides into chemical form less soluble in water or less likely to be incorporated in the human body, and incorporation in longlived organic material are examples. Very sophisticated methods are needed to take these effects properly into consideration, and there is also a lack of applicable data. In the present calculations it has as an alternative been chosen to assume that the dose during year 500 (the last of 500 years with the same release) represents the maximum annual dose. It has not been possible to calculate dose commitments. The calculated collective doses are based upon the present population density. Real calculation of collective dose more than 500 years from now is impossible. Nothing is known about population densities or distributions so far into the future.

On the other hand, it is often possible to draw valid conclusions, even though the available information is not complete, and this has also been the case in the present study.

Two types of comparisons may be carried out: comparison of one part of the management system with the other parts; and comparison of alternate solutions of the same part of the system. The first case involves comparison of storage, transportation and disposal. The second case involves comparison of bitumen drums, concrete drums and concrete moulds; comparison of sea and land transportation; and comparison of shallow land burial, concrete bunker and rock cavern.

It has not been the intention of this study to specify general requirements, but nevertheless it is possible from the results to indicate some possible requirements to product properties, design of the various parts of the management system, and nuclide content.

Likewise it is possible to indicate conditions that are not critical, e.g. the fact that the doses resulting from a transportation accident will not increase much, even if the 5 year extra radioactive decay in storage are not taken into account.

In figures 10.1 and 10.2 are shown individual and collective doses for the various accident scenarios, compared to the natural background radiation level. It is important to remember what was said in the previous paragraphs about the difficulties of comparing doses from the various accident scenarios, when evaluating these results. Reference should be made to the more comprehensive list of results in section 8.1 of Technical Part II.

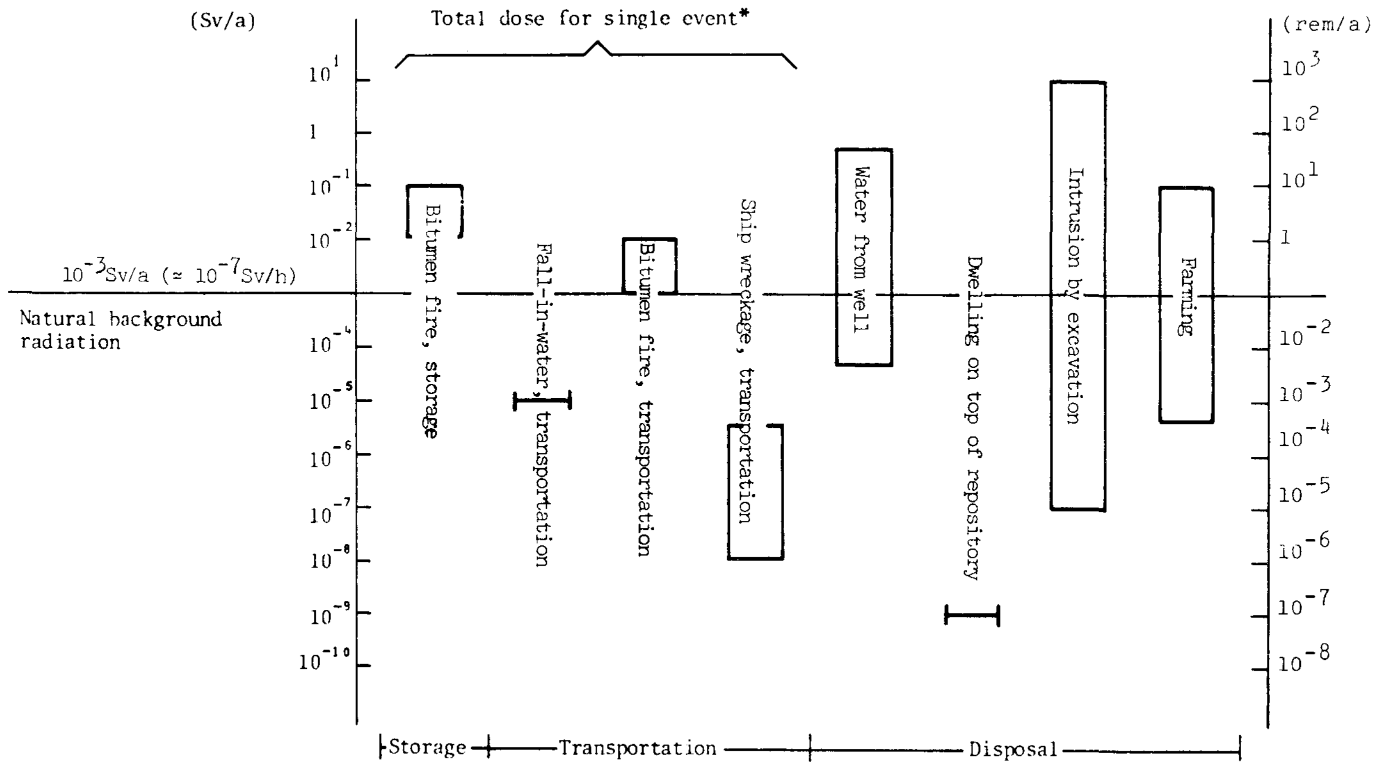
The natural background radiation levels, as well as radiation levels in the human environment, may serve as a rough yard-stick to which calculated doses and dose rates may be compared in order to gain an improved feeling for their relative importance.

In table 10.1 are given typical values of annual doses, gathered from numerous references.

Table 10.1 Average value of equivalent dose to population.

Source of radiation	Effective dose equivalent (mSv/a)
Radon daughters indoors	2
Natural background	1
X-ray diagnostic	1
Radiofarmaca	0.2
Nuclear weapon test fall-out	0.04
All other sources	<0.03
Total	3 - 4

1 mSv = 100 mrem



Not dose rates, but total committed doses from single, very improbable accidents.

Figure 10.1 Individual doses and dose rates

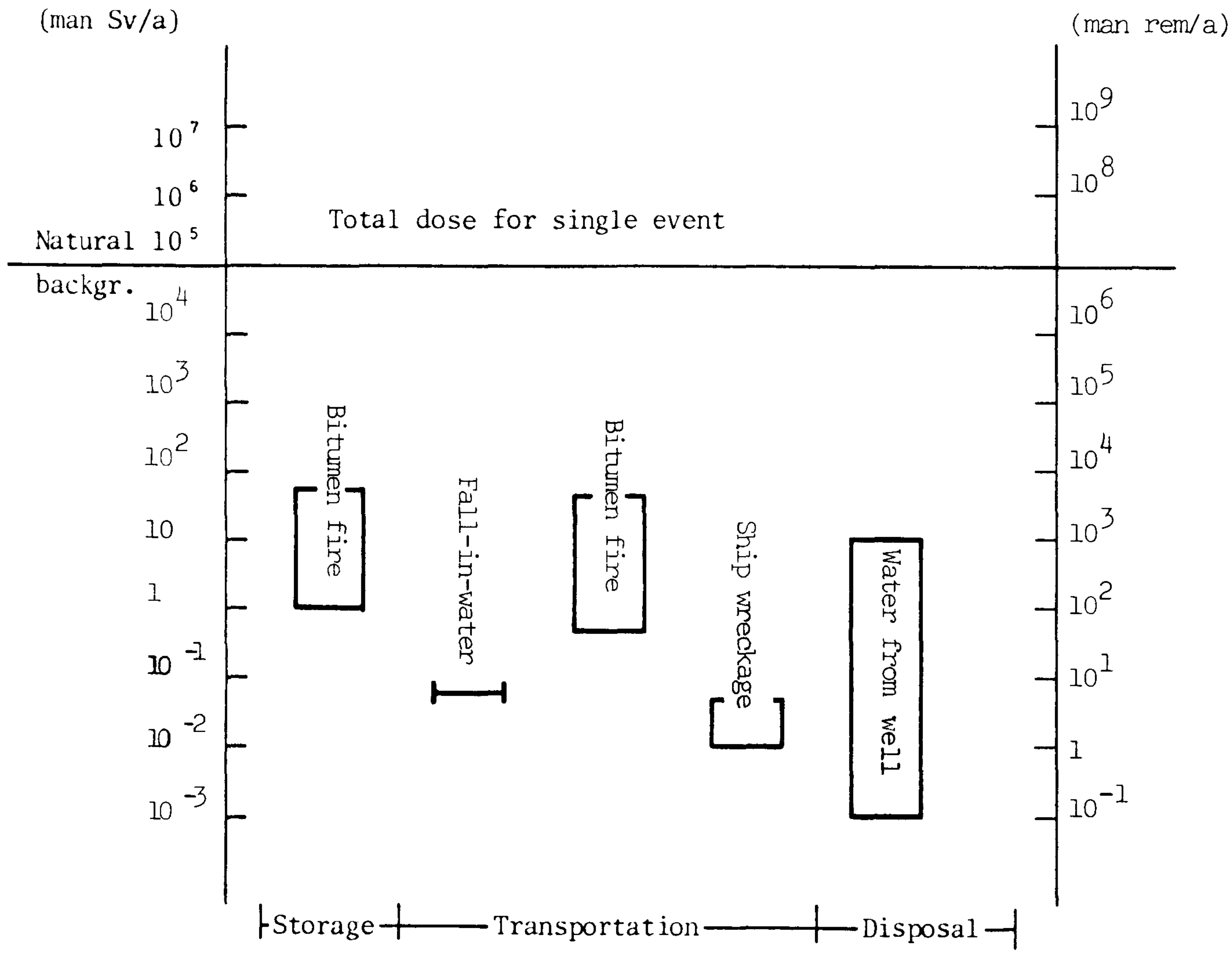


Figure 10.2 Collective doses and dose rates

It has been possible to draw conclusions of many different types from the study, and in the following sections the most important of these conclusions are summarized.

10.1. CONCLUSIONS RELEVANT TO PRODUCT PROPERTIES

- Only very few product properties are critical in relation to the different waste handling sequences studied.

- It appears that in the past excessive requirements to product performance have often been put forward, due to absence of system-governed product specifications.

- The most important properties include leaching, swelling, and disintegration phenomena by water. This is especially relevant to disposal. During storage and transportation the thermal resistance of bitumen products can be an essential product property.

- The mechanical strenght inherently achieved in normal solidification processes, in conjunction with the outer container (steel drum, transport container, etc.), is sufficient to withstand effects occurring in the various storage and transportation scenarios.

- Of the available spectrum of test methods, only few are well suited to give an adequate picture of the real properties of full-scale waste products. Better correlation between laboratory tests used for R & D work and real product properties needs to be established.

10.2 CONCLUSIONS RELEVANT TO STORAGE

- A fire following an airplane crash is the only scenario considered. All other accident scenarios are judged even less probable, and they will not lead to larger consequences.

- Although the individual dose commitments resulting from a bitumen fire in storage exceed all other dose commitments or annual committed doses calculated in the study, the probability is so low that the risk from this scenario is deemed lower than from many of the others. See also conclusion about bitumen fire during transportation, section 10.3.

10.3 CONCLUSIONS RELEVANT TO TRANSPORTATION

- As for storage the risk from a bitumen fire is not considered critical. It must be ensured, however, that the amount of radioactive materials contained in the release from one event can not be significantly larger than assumed. The doses may then reach a level where acute health effects (occuring a short time after exposure) may be encountered, and the basis for risk judgement is drastically changed if this is the case. A ten times higher nuclide content in the storage fire case and hundred times higher in the transportation fire case, would probably be judged unacceptable. However, the possibly unrealistically high estimated fraction of nuclide content released during fire should be kept in mind.

- From a pure risk point of view sea transportation seems to be preferable. The analysis of the sea transportation, however, was performed for one accident location only. Other possible accident locations might lead to significantly higher individual committed doses. Change of accident location in the land transportation case will not lead to significantly different doses.

- The doses resulting from a transportation accident will not increase much, even if the 5 years extra radioactive decay in storage is not taken into account.

- Any requirements to mechanical strenght of the waste packages from a transportation safety point of view are not motivated.

10.4 CONCLUSIONS RELEVANT TO DISPOSAL

- The models available for calculation of doses in the various diffusion scenarios for disposal contain no detailed description of the complicated physical and chemical phenomena actually taking place. This is a serious shortcoming, as these phenomena, described in different ways, may change doses and the times at which exposure takes place significantly, especially in the long term.

- The models used for calculation of doses in the various diffusion scenarios for disposal can not take into account in a satisfactory manner the barriers between waste and the environment. The methods, as presently used, probably result in very conservative doses and times at which exposure takes place.

- Parameter variations show that retention time in soil is particularly important in relation to the "water from well" scenarios, provided the well is not directly outside the repository. In the present calculations it is assumed that there is no retention of precicely those nuclides that happen to dominate the doses. The calculated doses may be very conservative if the retention times should have been significantly different.

- As expected, the calculations show that the doses from scenarios involving concrete bunker and rock cavern are always lower than from those involving shallow land burial, with rock cavern lower of the two. If it is chosen, as an additional safety measure, to cover the shallow land burial with a concrete lid, possible doses from all excavation scenarios, may be reduced to a negligible level. The additional dose reduction obtained when choosing concrete bunker or rock cavern, even if significant in relative values, seems to give only marginal benefit, when the doses are compared to other radiation doses in the human environment.

Only in a few of the long-term release scenarios it is the long-lived nuclides C-14 and I-129 that dominate the doses. In absolute terms, the calculated doses are low. If more realistic assumptions about concentrations in reactor waste and about leach coefficients of these nuclides are applied, their importance will be negligible. Accordingly, the emphasis currently being put on doses originating from these very long lived nuclides is probably unwarranted.

10.5 CONCLUSIONS RELEVANT TO SYSTEM AS A WHOLE

- All collective doses calculated in the reference cases are very low compared to the natural background radiation level.

- Only in a few of the release scenarios are the individual committed doses or dose commitments higher than or on the level of magnitude of the natural background radiation. Most of these scenarios are, however, very unlikely (bitumen fire, water from well close to repository, intrusion by excavation) and/or very conservative (water from well).

- The farming and the intrusion by excavation scenarios deserve, however, special attention; the first because it can not easily be classified as either unlikely or as conservative, and the second because it is not conservative and may result in very high dose rates.

REFERENCES

1. Handling of spent nuclear fuel and final storage of vitrified high level reprocessing waste. Kärnbränslesäkerhet Vol. I-V. Solna 1978.

2. Handling and final storage of unreprocessed spent nuclear fuel. Kärnbränslesäkerhet Vol. I-II, 1978.

3. Kjellbert, N., Häggblom, H., Cederström, M., Lundgren, T., Central repository for low- and intermediate-level waste (ALMA), conceptual design, siting and safety study. Report PRAV 1.29. 1980.

4. Devell, L., Edlund, O., Kjellbert, N., Grundfelt, B., Milchert, T., Safety analysis of sea transportation of solidified reactor wastes. Report PRAV 1.31, dated June 1980.

5. Annals of the ICRP Vol. 2 No. 1-4. ICRP Publication No. 30 Part 1 and 2. Limits of intakes of radionuclides by workers. Pergamon Press, Oxford 1979.

6. Snellman, M., A preliminary study on carbon-14 behaviour in LWR circuits. NKA/AO(80)21. 1980.

7. Hultgren, Å., et al., Status report 1980.10.01 on work on inorganic ion-exchangers and extraction technique for processing of radioactive waste.(In Swedish). Report PRAV 3.24, 1980.

8. Valkiainen, M., Nykyri, M., Incineration of ion exchange resins in the fluidized bed, IAEA Research Contract No. 2358/Cf. To be published.

9. Report by a group of experts of the OECD Nuclear Energy Agency. Objective, concepts and strategies for the management of radioactive waste arising from nuclear power programmes. Sept. 1977.

10. Spent nuclear fuel and radioactive waste. (In Swedish). SOU 1976:31.

11. Barbrean, A., Bardet, G., Escalier des Orres, P., Jourde, P., Scheidhauer, J., Sousselier, Y., Experience industrielle de stockage de dechets radioactifs sur le site de la Hague, IAEA-SM-137/18, Sept. 1970.

12. Disposal of high level waste from Danish nuclear power plants. Investigation of salt dome.(In Danish). Report prepared by ELKRAFT and ELSAM, June 1981. (English version will be issued later).

13. Reactor Safety Study. An assessment of accident risks in U. S. commercial nuclear power plants. Washington D. C. 1975. Nuclear Regulatory Commission. WASH-1400.

14. Evans, S., BIOPATH, - A computer program used for calculation of dose and dose commitment from release of radioactive nuclides to the biosphere. (In Swedish). Studsvik K2/80/250. 1980.

15. Rämö, E., Effect of fire on solidified reactor waste packaged into steel drums. NKA/AO(79)18. 1979.

16. Nykyri, M., Valkiainen, M., The combustibility of ion-exchange resins. (In Swedish). NKA/AO(79)22. 1979.

17. Dennis, A.W., Foley, J.T., Hartman, W.F., Larson, D.W., Severities of transportation accidents involving large packages, Sandia Laboratories report SAND 77-0001, May 1978.

APPENDIX

Working documents belonging to the Nordic study on reactor waste

Appendix

Working documents belonging to the Nordic study on reactor waste

Documents are available from the following libraries:

Denmark:	Forsøgsanlæg Risø Postboks 49 DK-4000 Roskilde
Finland:	Technical Research Centre of Finland Vuorimiehentie 5 SF-02150 Espoo 15
Sweden :	Studsvik Energiteknik AB S-611 82 Nyköping
Norway:	Institutt for energiteknikk Postboks 40 N-2007 Kjeller

Chronological list of NKA/AO documents:

NKA/AO-A:1	Pre-project report, November 1977.
AO(78)1	System and risk analysis of reactor waste. Status per March 1. 1978. F. Marcus. 1978-03-07. (In Danish and Swedish).
AO(78)8	Half-year report, 1st half-year 1978. 1978-08-08. (In Swedish).
AO(78)13	System and risk analysis of reactor waste. Proposal for the 1979 program. B. Berlin. 1978-12-08. (In Swedish).
AO(78)14	Product specifications. P.O. Nielsen. 1978-08-24. (In Norwegian).

AO(78)15	Temporary storage. P.O. Nielsen. 1978-10-05. (In Norwegian).
AO(78)16	Water qualities. P.O. Nielsen. 1978-10-05. (In Norwegian and Swedish).
AO(78)17	System description. Proposal for classification. P.O. Nielsen. 1978-12-18. (In Norwegian).
AO(78)18	Some views on system analysis concerning barrier philosophy and planning of waste repository. K. Brodersen. Risø, December 1978. (In Danish).
AO(78)19	Some general rules concerning the location of a final repository above ground in Denmark. K. Brodersen. Risø, December 1978. (In Danish).
AO(78)20	Leaching from radioactive asphalt-ion exchange mixtures. K. Brodersen, A. Vinther. Risø, December 1978. (In Danish).
AO(78)21	Radioactive waste - mineralogical influence on capsule material. Rolf Hallberg. 1978-12-14. (In Swedish).
AO(78)22	Structure and work plan for project NKA-AO/A:1. P.O. Nielsen. 1978-12-01. (In Norwegian).
NIPA(78)2/NAK(78)16	O. Kleveland and H. Tollbäck: Solidification of grainy and powdery ion exchange material with Portland cement. (In Swedish).

Reports which have not been written for the NKA-project but which have a connection to the scope of the project.

COLUMN Numerical Solution of Migration Equations Involving Various Physicochemical Processes. P. Bo. Risø Feb. 1978.

Ion Exchange Properties of Soil Fines. Peter Bo. EF-contract No. 038-78-1 WASDK. Half-year Report Jan. 1 - Sept. 30, 1978.

The Characterization of Carbonate Complexes which may be formed in the Ground Water. Karen Nilsson. EF-contract No. 038-78-1 WASDK. Half-year Report Jan. 1 - Sept. 30, 1978.

AO(79)1 Draft	Enlarged Nordic co-operation on the field of nuclear safety. Annual Report for 1978. F. Marcus. (In Danish).
AO(79)2	An analysis of materials durability and its testing with reference to radwaste concrete. S.E. Pihlajavaara. Jan. 1979. (In English).
AO(89)3	Radwaste concrete: Solidification of nuclear wastes with Portland Cement. S.E. Pihlajavaara, J.P. Aittola. April 1979. (In English).
AO(79)4	Renewed report for the second half-year 1978. System and Risk Analysis of Reactor Waste Management. Subproject Final Deposition. Jorma K. Miettinen. 1979-01-03. (In English)

AO(79)5	Product characteristics - Relevant questions 1. Fire characteristics in de-watered ion exchange material. Product characteristics - Relevant questions 2. Product characteristics - Relevant questions 3. Final repository - Relevant questions 4. Proposal for a description of the final repository. P.O. Nielsen. 1979-01-26. (In Norwegian).
AO(79)6	Storage in clay. J.E. Lundby. 1979-01-29.
AO(79)7	Proposal for the 1979 work program. P.O. Nielsen. 1979-02-06. (In Norwegian).
AO(79)8	Proposal for the 1979 program. System description and safety analysis. 1979-02-14. (In Norwegian).
AO(79)9	Proposal for the 1979 program. Product charasteristics. J-P. Aittola. 1979-02-15. (In Swedish).
AO(79)10	Proposal for the 1979 program. Final storage. Sub-project within the Nordic waste project, NKA/AO-A:1. Per Linder. 1979-02-12. (In Swedish).
AO(79)11	System and resource analysis for reactor waste. Detailed program for 1979. B. Berlin et al. 1979-02-22. (In Swedish).
AO(79)12	Background for half-year report, 2nd half-year 1978. Bo Berlin. 1979-04-23. (In Swedish, Danish, Norwegian).
AO(79)13	NIPA topical meeting May 29, meeting for the leading group May 30. K. Brodersen. 1979-05-02. (In Danish).
AO(79)14	Project proposal. The significance of the activity contents for the setting up of a system concerning reactor waste. Per Linder. 1979-05-07. (In Swedish).

AO(79)15	Safety analysis. 1st iteration with 3 appendices. P.O. Nielsen. 1979-05-15. (In Norwegian).
	Appendix 1: Product characteristics, VTT (J.P.Aittola). 1979-05-18. (In Swedish).
	Appendix 2: Consequence Analysis - Waste in Bitumen or Concrete. A short survey of methodology, IFA (U. Tveten), 1979-05-15. (In English).
	Appendix 3: Dose calculation for release from 8 waste drums on fire, IFA (D. Thomassen), 1979-05-22.
AO(79)16	Sub-project "Product characteristics". Running and planned works 1979. P. Aittola. 1979-05-23. (In Swedish).
AO(79)17	Damage of reactor waste packages in transport accidents. A. Rämö, U. Pulkkinen, J-P. Aittola. May 1979. (In English).
AO(79)18	Effect of fire on solidified reactor waste packaged into steel drums. E. Rämö. 1979-07-30. (In English).
AO(79)19	Sub-project: Final storage. Renewed list of data considering proposed viewpoints. C. Stenqvist. 1979-06-29. (In Swedish).
	The significance of micro-organisms for activity leaching from bituminized radioactive waste. Knud Brodersen. May 1979. (In Danish).
	System description. 1979-03-19. (In Danish).
	Containers used in packaging reactor waste. A paper presented in the symposium on the on-site management of reactor waste, March 26-30, 1979. E. Rämö. 1979-03-16. (In English).

Sorption of radioactive nuclides in soils; characterization of 14 soils samples. Progress report. University of Helsinki. M. Kiuru, J.K. Miettinen. April 1979.

AO(79)20 rev. Contents of activity in reactor waste. P. Linder. 1979-08-20. (In Swedish).

AO(79)21 Distribution of particle size in a fire. Hahkala, Mehtonen, Aittola. (In Swedish).

AO(79)22 Ion exchangers´ characteristics in a fire. 1979-08-30. (In Swedish).

AO(79)23 Diffusion-, migration- and dose calculations. C. Stenqvist. 1979-08-14. (In Swedish).

AO(79)24 Solidification of grainy ion exchange material in slag cement. Tollbäck, Kleveland, Aittola. 1979-06-19. (In Swedish).

AO(79)25 Workplans, 2nd part 1979.(In Danish, Swedish, Norweg).

AO(79)26 Waste field. Half-year report of 1st half-year 1979. F. Marcus. 1979-08-13. (In Danish).

AO(79)27 Waste field. Plans for 1980. F. Marcus. 1979-08-13. (In Danish).

AO(79)28 1st half-year report for 1979 (Background material) 1979-08-10. (In Swedish, Norwegian).

AO(79)29 List on site-data. C. Stenqvist. 1979-08-23. (In Swedish).

AO(79)30	Characteristics of bitumen products (Aittola/ Kleveland). (In Swedish).
AO(79)31A	Dose calculation of release during transport of waste. U. Tveten. 1979-10-25. (In Norwegian).
AO(79)31B I&II	Final storage. Applied models and calculation models. C. Stenqvist. 1979-10-17. (In Swedish).
AO(79)32	Project proposal for NKA/AO-A:2. Collecting of background information. 1979-10-16. (In Swedish).
AO(79)33	Amounts of and analysis methods for long-lived Ni-isotopes. K. Brodersen. October 1979. (In Danish).
AO(79)34	Background material for the assessment of the necessity for continued research on long-lived isotopes in reactor waste. K. Brodersen. October 1979. (In Danish).
AO(79)35	System description of reactor waste management for safety analysis - waste amounts and nuclide inventories. E. Ruokola and E. Rämö. 1979-12-05. (In Swedish)
AO(79)36	Circumstances which may cause release from reactor waste stored in concrete bunkers. K. Brodersen. December 1979. (In Danish).
AO(79)37	Distribution and dose calculations. C. Stenqvist. 1979-12-17. (In Swedish).
AO(79)38	Sorption of radioactive nuclides in soils. Final report. Nukula, Kiuru, Miettinen. Nov. 1979. (In English).
AO(79)39	System and risk analysis. Final storage. A critical assessment of K_d-values. K. Heiskanen, E. Peltonen, M. Snellman. Sept. 1979. (In Swedish).

AO(80)1 Draft 2	Scandinavian work on disposal of waste from reactor operation. F. Marcus, R. Boge, M. Bonnevie-Svendsen and E. Rämö. Paper presented at the Waste Management '80 in Tucson, Arizona, 1980-03-12. 1980-03-30. (In English).
AO(80)2 Draft 1	Nordic symposium on reactor waste. F. Marcus/KJ. 1980-01-10. (In Swedish).
AO(80)3 rev.	Annual report 1979 on waste AO. F. Marcus and H. Reijonen. 1980-01-28. (In Danish).
AO(80)4 rev. 1	Work plan for project A:1 during 1980. 1980-02-07. (In Swedish).
AO(80)5 Draft	Proposal for assessment of relevant waste management meetings. R. Boge. Jan. 1980. (In Swedish).
AO(80)6	Proposal concerning visits to authorities/institutions. R. Boge. Jan. 1980. (In Swedish).
AO(80)7	Proposal for an investigation of K_d-values. R. Boge. Jan. 1980. (In Swedish).
AO(80)8 Draft	A proposal for reporting a meeting at IAEA in Vienna Dec. 17-21, 1979, concerning "Conditioning of Low and Intermediate Level Radioactive Waste for Storage and Disposal". R. Boge. 1980-01-14. (In Swedish).
AO(80)9	Nuclide content in reactor wastes. I Summary of the work done during 1979. II Proposals for the work during 1980. E. Ruokola. 1980-01-08. (In English).
AO(80)9 add.	Contents of activity in reactor waste. Annual report for 1979. E. Ruokola. 1980-01-15. (In Swedish).

AO(80)9.1	Nuclide contents in low and medium level reactor wastes. Summary of the work done in 1979. Appendix 1. S. Lindhe. 1979-11-16. (In Swedish).
AO(80)9.2	A literature study of waste sources, their characterization and distribution in PWR waste streams. Summary of the work done in 1979. Appendix 2. M. Snellman. 1979-12-30. (In English).
AO(80)9.3	Description of the RADTRAN code and some demonstration calculations. Summary of the work done in 1979. Appendix 4. G. Horvāth, E. Ruokola, M. Snellman. 1980-01-15. (In English).
AO(80)9.4	Proposal for ASEA-ATOM´s participation in the project NKA/AO-A:8. Proposals for the work in 1980. Appendix 5. O. Carlson. 1980-01-07. (In Swedish).
AO(80)9.5	Activity in reactor waste. S. Westerberg and H-P. Hermansson. 1980-01-08. (Partly in English, partly in Swedish).
AO(80)10	Description of dose calculation via the nutrition chain made for the Scandinavian project concerning waste risks. U. Tveten. (In Norwegian).
AO(89)11	The use of ultrasound for assessing the quantity of waste resin and the pressure resistance of waste resin concrete. K. Nikkari, H. Paroll. 1980-01-31. (In Swedish).
AO(80)12	System and risk analysis concerning reactor waste. Notes on earthquakes, erosion, hydrology. P.O. Nielsen. 1980-02-10. (In Norwegian).
AO(80)13	Multi-element analysis of leaching solutions. M. Bonnevie-Svendsen. (In Norwegian).

AO(80)14	The effect of a well on the radionuclide migration. K. Heiskanen, H. Härkönen, E. Peltonen, J. Puttonen. 1980-02-29. (In English).
AO(80)15 rev. 2	Fall-in-water accident. P.O. Nielsen. 1980-06-06. (In English).
AO(80)16	Assessment of individual doses and collective doses. U. Tveten. 1980-01-12. (In Norwegian).
AO(80)17	Calculation of Doses via Nutrition. U. Tveten. 1981-01-23. (In English).
AO(80)18 rev. 1	Dose calculations from final repository. Sub-project: Final Repository. C. Stenqvist. 1980-05-05. (In Swedish).
AO(80)19	Leach testing of simulated concrete products. K. Ollila, E. Rämö. 1980-03-31. (In English).
AO(80)20	A preliminary study on correlation between activities of fission products released from fuel. E. Ruokola. 1980-04-09. (In English).
AO(80)21	A preliminary study on Carbon-14 behaviour in LWR circuits. M. Snellman. 1980-04-09. (In English).
AO(80)22	Solidification of boron containing grainy ion exchange material in concrete. O. Kleveland, J-P. Aittola, H. Tollbäck. 1980-04-10. (In Swedish).
AO(80)23	Report on a task from project NKA/AO:A2 regarding the usefulness of K_d-values for project NKA/AO:A1 "System and risk analysis of reactor waste". K. Andersson. 1980-04-10. (In Swedish).

AO(80)24	List on parameter values in calculations of distribution and doses. C. Stenqvist. 1980-05-05. (In Swedish).
AO(80)25	Draft on the appendix 1/section 3 of the final report: Waste. E. Ruokola. 1980-05-20. (In English).
AO(89)26	Solidification of grainy ion exchange material in cement - comparing experiments with different types of cement. J-P. Aittola, O. Kleveland, H. Tollbäck. 1980-05-28. (In Swedish).
AO(80)27	Draft of the appendix 1/section 1 of the final report: Product characteristics. J.-P. Aittola. 1980-06-02. (In English).
AO(80)28	Variation of the diffusion coefficient in waste containers. C. Stenqvist. 1980-06-24. (In Swedish).
AO(80)29	Release of Radionuclides from the Repository. K. Heiskanen. 1980-06-30. (In English).
AO(80)30	Specific surface measurement of ion exchange material and cement solidified ion exchange material. J. Chyssler, B. Grönroos. J-P. Aittola. 1980-04-30. (In Swedish).
AO(80)31	Enlarged Nordic co-operation on the field of nuclear safety - Reference system. Drafts for sections 1, 2 and 3. P.O. Nielsen. 1980-08-12. (In English).
AO(80)32	Fall-in-water accident. Dose calculations. P.O. Nielsen. 1980-08-25. (In English).
AO(80)33	Probability of truck transport accidents. P.O. Nielsen. 1981-06-03. (In English).

AO(80)34	Radioactive waste transportation risk and temporary storage fire risk. U. Tveten. 1980-09-29. (In English).
AO(80)35	Half-year report for 1980. 1980-08-20. (In Swedish).
AO(80)36	On the diffusive migration of ^{60}Co, ^{90}Sr and ^{137}Cs from a concrete/clay repository. L. Carlsen and P. Bo. August 1980. (In English).
AO(80)37	Statistical analysis of waste activity data from Oskarshamn and Barsebäck. E. Ruokola and M. Kätkä. 1980-08-20. (In English).
AO(80)38	Safety analysis. Draft for section 4. C. Stenqvist. 1980-05-21. (In English).
AO(80)39 rev.	Reference system, draft for section 5. C. Stenqvist, P. Linder. 1980-09-01. (In English).
AO(80)40	Background material for the contents of the main report discussed at a work meeting in Stockholm 1980-10-16. M. Bonnevie-Svendsen, P. Linder, P.O. Nielsen, H. Reijonen and C. Stenqvist. 1980-10-21. (In English).
AO(80)43	Calculation of collective doses and presentation of risks. B. Persson. 1980-11-19. (In Swedish).
AO(80)44	Workshop on radionuclide release scenarios for geologic repositories. Report from a work meeting in Paris. S. Huseby. 1980-10-13. (In Norwegian).
AO(81)1	Risø Report M 2262. Lars Carlsen.
AO(81)2	Carbon-14 in ion exchange resins from Finnish nuclear power plants. Laina Salonen, Margit Snellman. 1981-01-15. (In English).

AO(81)3 Determination of C-14 in ion exchange material. J-P. Aittola, A-L. Olsson. 1980-11-20. (In Swedish).

AO(81)4 Nuclide content in reactor wastes. Draft of final report. E. Ruokola. 1981-02-20. (In English).

AO(81)5 Nordic study on reactor waste. Main Report. Technical parts I, II and III. August 1981. (In English).

AO(81)6 Disposal of reactor waste in bedrock - Sensitivity analysis of dose by response surface method. E. Peltonen, R. Sairanen. 1981-06-05. (In English).

AO(81)7 Testing and estimation of durability of concrete and radwaste concrete for long-term storage. S.E. Pihlajavaara. September 1981. (In English)