ENLARGED NORDIC COOPERATIVE
PROGRAM ON NUCLEAR SAFETY

NORDIC STUDY ON REACTOR WASTE

Main Report

Organization
of the Nordic Study on Reactor Waste

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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 individual 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

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Working documents belonging to the Nordic study on reactor waste

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 sys-
tem 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 represen-
tative 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 pro-
cess 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.

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REACTOR TYPE:

BWR (6 x 500 MWe)

WASTE TYPES:

Granular resin   Powdered resin
(RWCS)   (SFPCS)

SOLIDIFYING MATRIX:

Cement   Bitumen

PACKAGE TYPES:

Concrete mould   Drum   Drum
(1 m^3 net)   (200 1)   (200 1)

STORAGE:

Storage bldg.   Rock cavern   Storage bldg.

TRANSPORTATION:

Truck   Boat

DISPOSAL:

Shallow land   Concrete   Rock cavern
burial   bunker   (granite)
(sand or clay)   (sand or clay)
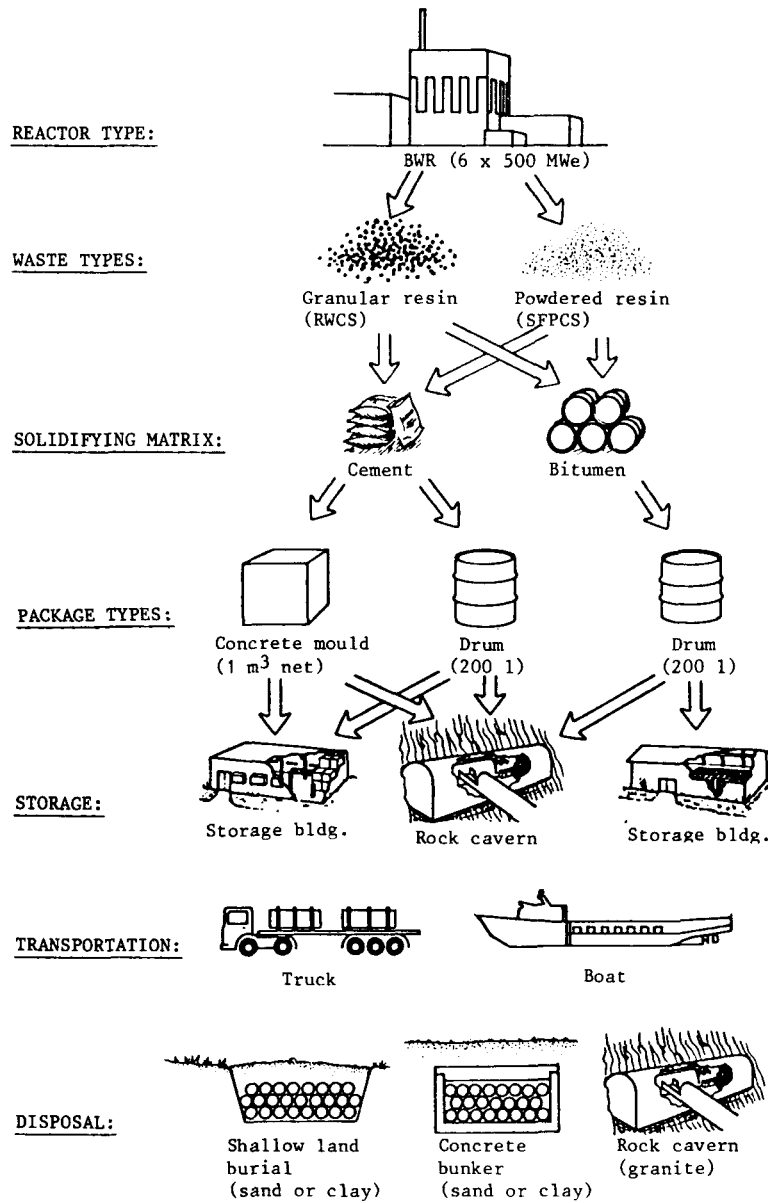
Figure   2.1   The reference waste management system.

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 satisfac-
tory. 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
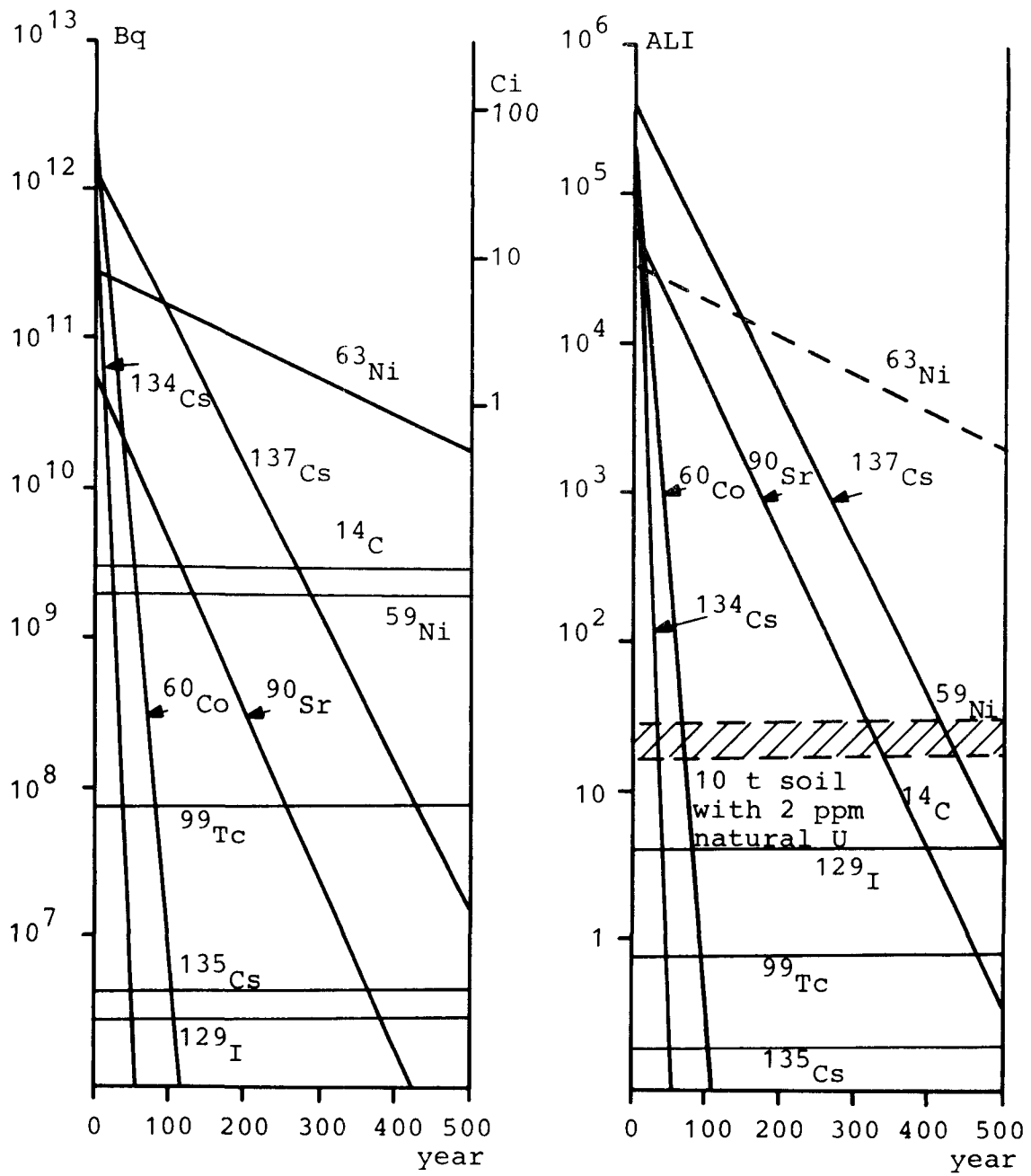
Figure 4.1   Decay of activities and relative toxicities
for an annual amount of 3.4 tonnes dry resin
waste from a 500 MW$_e$BWR.

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 con-

tainers 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 radio-
activity. 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 under-
standing 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 be-fore 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 radio-
  activity, 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 conse-
quence 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 correspon-
ding 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.
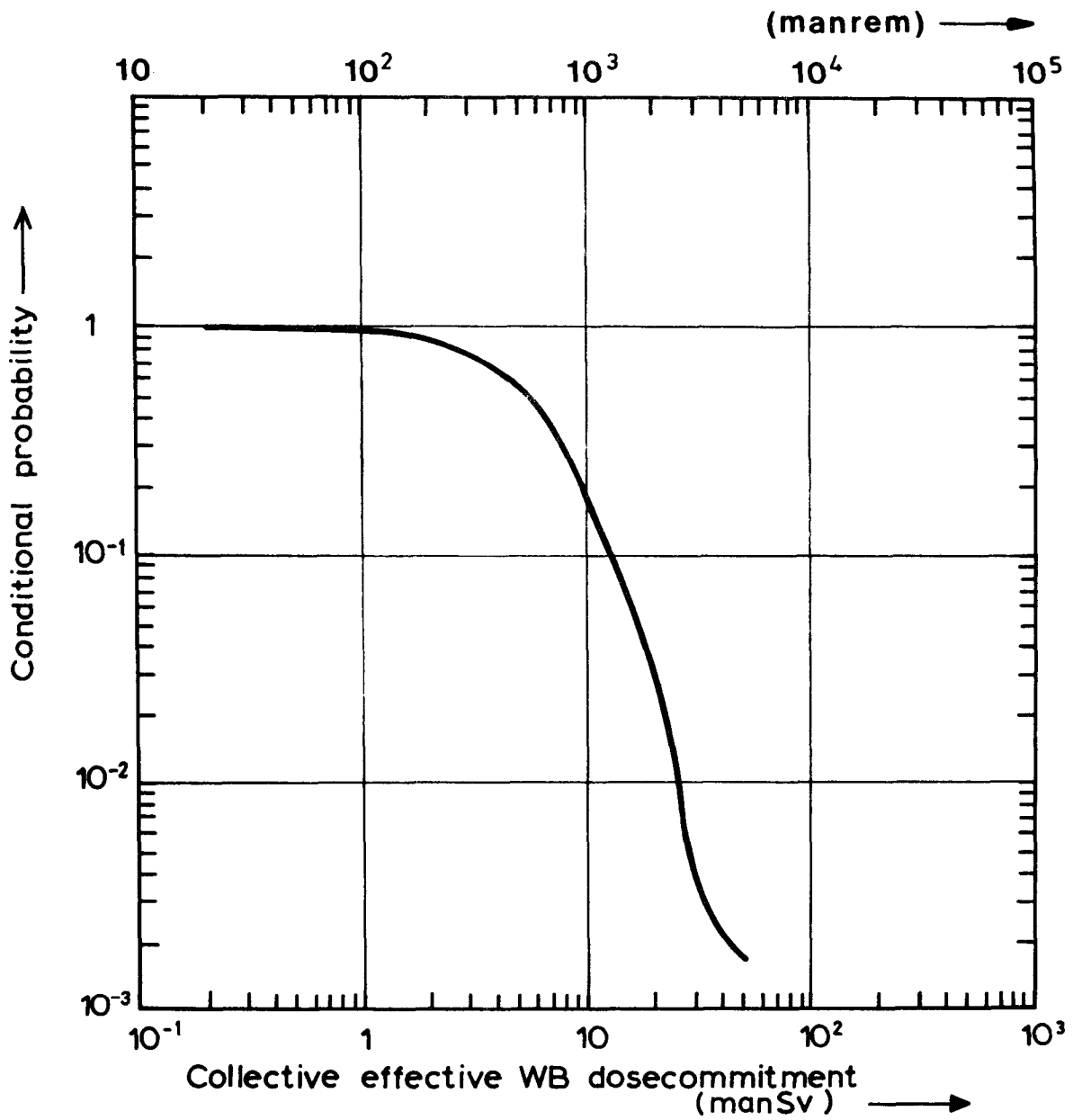
Figure 8.1   Calculated risk curve. Bitumen fire in storage.

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 (PWCS 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 trans-
port containers made from reinforced concrete. The contai-
ners 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 bi-
tuminized waste does not start unless the bitumen is expo-
sed 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 distri-
buted over this time period. Furthermore it is assumed that
100% of the cesium and 85% of the other nuclides are relea-
sed 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, des-
cribed 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 $3x10^{-9}$ 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 undistur-
bed 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 interre-
lated, 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 reposi-
tory. This dose is significant, as it is roughly ten times
the natural background radiation. But again the conser-
vative 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 radio-
activity 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 repo-
sitory, 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 nutri-
tion pathways only the vegetable pathway has been included
in these calculations, and the total dose with all nutri-
tion 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 requi-
rements 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 rela-
tively 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

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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
$\begin{cases} Cs\text{-}137 \\ Cs\text{-}135 \end{cases}$	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	$2x10^{-6}$	$6x10^{-6}$	?
$\begin{cases} Ni\text{-}63 \\ Co\text{-}60 \end{cases}$	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 radio-
active 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
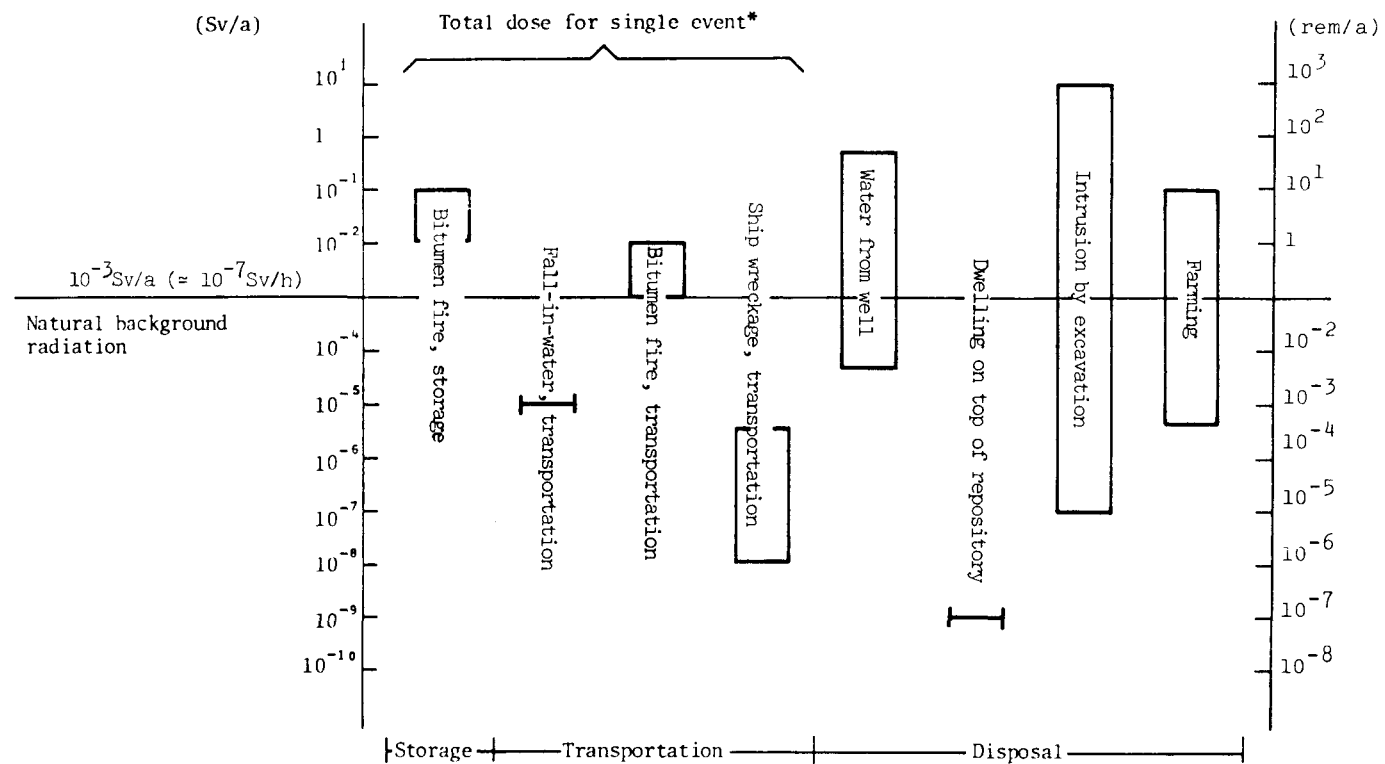
Figure 10.1   Individual doses and dose rates

(man Sv/a)   (man rem/a)

10^7

10^6

Natural 10^5 Total dose for single event

backgr. 10^4

10^3

10^2 Bitumen fire   Bitumen fire

10   Fall-in-water   Ship wreckage   Water from well

1

10^{-1}

10^{-2}

10^{-3}

10^9

10^8

10^6

10^5

10^4

10^3

10^2

10^1

1

10^{-1}

|Storage—+——————Transportation————+—Disposal—|
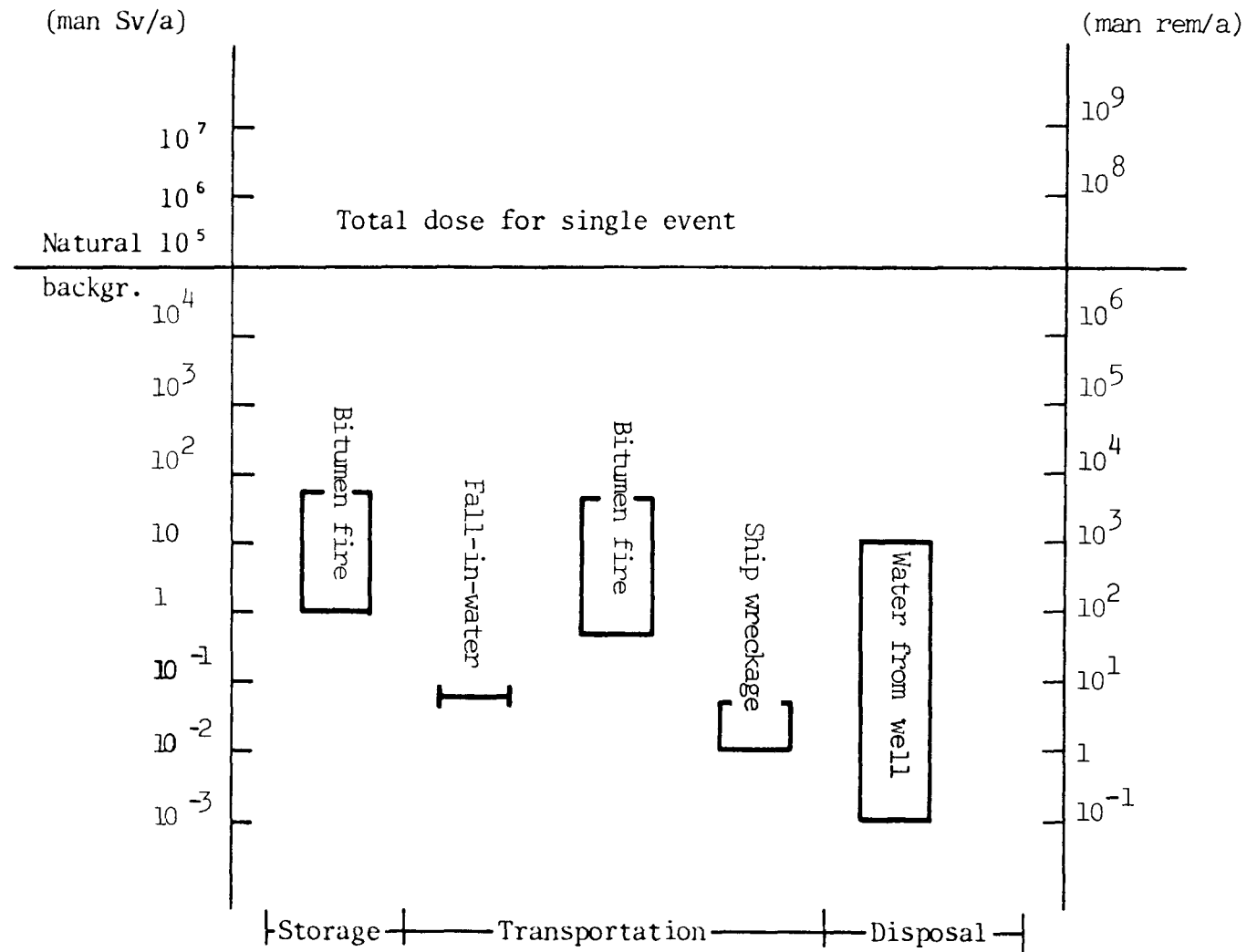
Figure 10.2   Collective doses and dose rates

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 transporta-
  tion, 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.

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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.

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AO(78)16   Water qualities. P.O. Nielsen. 1978-10-05.
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AO(78)17   System description. Proposal for classification.
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AO(78)18   Some views on system analysis concerning barrier
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