NKA NORDIC LIAISON COMMITTEE
FOR ATOMIC ENERGY

NKA/AO (81) 5
August 1981

ENLARGED NORDIC COOPERATIVE PROGRAM ON NUCLEAR SAFETY

NORDIC STUDY ON REACTOR WASTE

Technical Part I
Reference System

and

Technical Part II
Safety Analysis

A Joint Scandinavian Research Project
Sponsored by

The Nordic Council of Ministers

Organization
of the Nordic Study on Reactor Waste

Project Board

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

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

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

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

---

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

---

Author of Technical Part I: P.O. Nielsen

Author of Technical Part II: U. Tveten

Author of Technical Part III: M. Bonnevie-Svendsen

---

* replaced Mr. J. Heinonen in 1980

** replaced Mr. B. Berlin in 1979

NORDIC STUDY ON REACTOR WASTE

TECHNICAL PART I

REFERENCE SYSTEM

LIST OF CONTENTS

		Page
1.	GENERAL DESCRIPTION	1
2.	WASTE	5
2.1.	Reference waste source and waste types	5
2.2.	Characteristics of unconditioned waste	5
2.3.	Amounts of wastes	6
2.4.	Nuclide inventory	7
2.4.1.	Nuclides considered in the present study	7
2.4.2.	Corrosion products	8
2.4.3.	Fission products and actinides	10
2.4.4.	Carbon-14	12
2.4.5.	Summary of nuclide inventory	13
2.5.	Waste packages	16
2.5.1.	Primary packaging containers	16
2.5.2.	Steel drums for bituminized or cementized waste	16
2.5.3.	Concrete mould for cementized waste	17
2.5.4.	Waste content in different packages	18
2.5.5.	Nuclide contents in different packages	18
2.5.6.	Radiation dose rate from different packages	18
2.6.	Diffusion and leaching characteristics of the conditioned waste	24
2.7.	Fire characteristics	25

Page

2.8.	Impact characteristics	26
2.9.	Compressive strenght of the cementized product and the moulds	26
2.10	Swelling in water	27
3.	SITE DATA	28
3.1.	General	28
3.2.	Meteorology	28
3.3.	Population distribution	30
3.4.	Radioecological data	33
3.5.	Geology	35
3.6.	Hydrology	35
4.	STORAGE	37
4.1.	General	37
4.2.	Storage building for bituminized waste	37
4.3.	Storage building for cementized waste	40
4.4.	Rock cavern storage	42
5.	TRANSPORT	46
5.1.	General	46
5.2.	Road transport	47
5.3.	Sea tranport	51
6.	DISPOSAL	54
6.1.	General	54
6.2.	Shallow land burial	54
6.3.	Concrete bunker	55
6.4.	Rock cavern	57
REFERENCES		59

APPENDIX A
Summary of data assumed for REFERENCE SYSTEM

1 GENERAL DESCRIPTION

An important part of the Nordic studies on system- and safety analysis of the management of low and medium level radioactive waste from nuclear power plants, is the safety analysis of a Reference System. This reference system was established within the study and is described in this Technical Part I. The safety analysis of the reference system is reported in Technical Part II.

The reference system covers waste management Schemes that are potential possibilities in either one of the four participating Nordic countries. Although some of the data that are used to characterize the reference system are in fact related to specific locations, the description and the data chosen for the reference system should be considered as examples only, and, in particular, the data chosen should not be considered as recommendations for a waste management system or parts thereof.

The reference system is illustrated schematically in figure 1.1. It is based on:

Power System: A power reactor system consisting of 6 BWR's of 500 MWe each, operated simultaneously over the same 30 year period.

Waste Types: Deep bed granular ion exchange resin from the Reactor Water Clean-Up System (RWCS and powdered ion exchange resin from the Spent Fuel Pool Cleanup System (SFPCS). These two waste types normally contain more than 90 % of the activity in the radioactive waste resulting from the operation of light water cooled nuclear power plants (excluding discarded core components) but may constitute less than 10 % of the total waste volume.

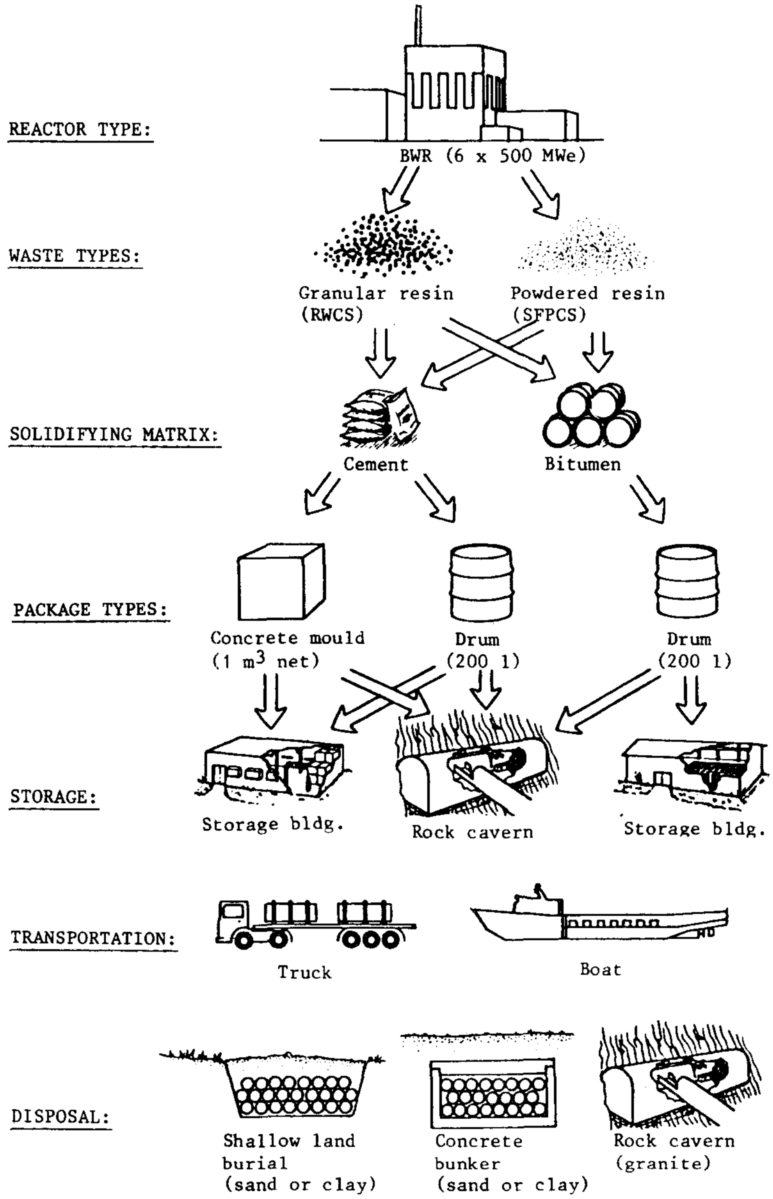


Fig. 1.1 The reference waste management system.

Solidification: Both waste types are supposed to be solidified by mixing with a matrix material. As matrix, cement and bitumen are chosen, because these materials are being used at nuclear power stations in Sweden and Finland and at research stations in Denmark and Norway.

Package Type: Two basic types of containers are considered, both in use in the Nordic countries: Standard 200 liter steel drums and specially made cubicreinforced concrete moulds with a net volume of 1 m^3 (one variant has a heavier shielding, but same outer dimensions).

The steel drums are used for bituminized as well as for cementized waste, while the concrete moulds are used only for cementized waste.

Storage: Facilities for temporary storage of the solidified waste at or near the nuclear power station is a requirement in many countries operating nuclear power reactors. The Nordic study assumes such a storage for at least 5 years, and a maximum of 50 years before the waste is transferred to the disposal site. Three different storage facilities are envisaged: A storage building for cementized waste, a storage building for bituminized waste, and finally a rock cavern storage for all types of waste. As a matter of fact, the last storage is identical to the rock cavern for disposal, but until it is permanently sealed, the cavern is considered to be equivalent to a temporary storage, as it is in continuous operation, under continuous surveillance and control, and the waste may be (fairly) easily removed from the cavern for alternative disposal.

Transportation: Transportation of the waste from the storage facility to the disposal site will be by road or sea, while transportation by train is not considered in this study. Transportation by road will be by means of a normal type truck, for transportation by sea, a specially designed vessel is envisaged. The waste will be transported in containers, and when necessary, transport containers with shielding capability will be used.

Disposal: Three different disposal facilities are considered: Shallow land burial, near surface concrete bunker, and rock cavern with about 30 m rock cover. For each of the first two facilities two alternative geologic formations are considered: Sandy till and clayey till. The rock is assumed to be granite.

More details on the assumptions and the parameters related to the reference system are given in the following chapters. In principle, these descriptions have been limited to include only data and other information which was found to be essential in order to carry out the safety analysis in Technical Part II. However, some non-safety-related information is included in order to provide sufficient understanding of the functioning of the Reference System.

A comprehensive list of all important Reference System charasteristics is compiled in tabular term in Appendix A.

2 WASTE

2.1 REFERENCE WASTE SOURCE AND WASTE TYPES

The reference reactors are the ASEA-ATOM type Boiling Water Reactor (BWR utilizing deep bed granular ion exchange resins in the Reactor Water Cleanup System (RWCS) and in the Liquid Waste Treatment System, whereas precoat type powdered resins are utilized in the Condensate Polishing System, Spent Fuel Pool Cleanup System (SFPCS) and also in some cases in Liquid Waste Treatment System. Amounts of evaporator concentrates are normally insignificant at the power plants concerned.

In the present study two waste types are selected for detailed analysis: RWCS and SFPCS resins. The reason for this choice is that these waste types are generally the most active constituents of wet wastes. The restriction to only two types of wastes is justified by the fact that the present study is a demonstration of a safety analysis methodology rather than a quantitative safety analysis of reactor waste management.

2.2 CHARACTERISTICS OF UNCONDITIONED WASTE

A mixed bed of synthetic organic ion-exchange resin is most commonly used at BWRs for RWCS. The need for the replacement of the bed is generally determined by the pressure drop or conductivity increase, or it is based on a fixed time schedule. A typical lifetime of the bed at an ASEA-ATOM reactor is one month.

In the SFPCS the powdered mixed bed resin functions both as filter and ion exchanger. The main factor determining the replacement of the bed is the pressure drop. A design basis for the lifetime of the bed at ASEA-ATOM reactors is two weeks.

2.3 AMOUNTS OF WASTES

RWCS and SFPCS resins normally constitute less than 10 % of the total waste volume.

The estimates made are based on the operating experience from 15 American BWRs with an output of 200 MW(e) or more, /1/, on the operating experiences gained at Swedish BWRs /2, 3/ and on the estimates made in /4/. Table 2.1 summarizes the estimates for the annual waste volumes and masses.

As a unit for the waste amount, the dry solid mass of resin is selected, because it will not change significantly when the waste flows through the selected treatment processes. In order to convert this unit into volumes, dry solid mass of resin per unit volume of solidification product must be known.

Table 2.1 Estimated average annual amounts of RWCS and SFPCS resins for a 500 MW(e) reactor.

Waste type	Annual volume decanted per reactor (m^3/a/)	Annual dry solid resin per reactor (kg/a/)
RWCS resin	12	2 400
SFPCS resin	10	1 000

The annual amounts of waste naturally vary considerably due to operating conditions etc. As most of the variation is of random character, it is smoothed out effectively during the lifetime of the reactor. The integral amounts of waste arising from 6 reactors operating 30 years each are not expected to vary by more than a factor of two around the average.

2.4 NUCLIDE INVENTORY

2.4.1 NUCLIDES CONSIDERED IN THE PRESENT STUDY

The nuclide content and composition in spent resins depends heavily on the reactor type and cleanup system in question. The activities can vary greatly depending on operating conditions and especially on cladding defects in the fuel pins. The same is true for the proportions between activities of corrosion products and fission products. On the other hand, the relative proportions of various corrosion product nuclides as well as fission product nuclides are not expected to vary greatly. For these reasons it is in most cases justifiable to select one predominant corrosion product and one predominant fission product, to estimate their activity for RWCS and SFPCS resins, and to express the activities of other corrosion and fission products as proportional to the selected reference nuclides.

The original activity in this report is defined as the activity of the waste at the end of the year during which radionuclides are produced.

In many cases the resin is stored in tanks for six months or even longer before it is solidified. For the Reference System, however, it is assumed that the waste is solidified at the end of the year in which it is produced. Only relatively long-lived nuclides can possess certain risk potential in the reference cases. The nuclides, which have been included in the study, are presented in table 2.2.

Table 2.2 Nuclides considered in the reference case.

Nuclide	Half-life (a)	Origin
C-14	5 700	Activation product
Ni-59	80 000	Corrosion product
Ni-63	92	Corrosion product
Co-60	5.3	Corrosion product
Sr-90	29	Fission product
Tc-99	210 000	Fission product
I-129	17 000 000	Fission product
Cs-134	2.1	Fission product
Cs-135	3 000 000	Fission product
Cs-137	30	Fission product
Pu-239	24 000	Actinide

Many of the actinides are characterized by very long half-lives, high radiotoxicities and low leaching and migration rates. Only plutonium-239 has been chosen as an example to represent the actinides.

2.4.2 CORROSION PRODUCTS

Cobalt-60 is selected as a reference nuclide for corrosion products, because it is a strong gamma emitter and consequently been measured and reported.

On the basis of the references available /1, 2, 3, 5/ the following "best-estimates", for the average accumulation rate of cobalt-60 activity in RWCS and SFPCS resins are selected:

RWCS resins	3 000 GBq/a	Co-60	($\sim$ 100 Ci/a)
SFPCS resins	200 GBq/a	Co-60	($\sim$ 5 Ci/a)

The amounts of nickel-59 and nickel-63 relative to cobalt-60 depends on the construction materials of the primary circuit. Because of the differences in half-lives, the ratio is not even constant during the operating period of the reactor because the activities of nickel-59 and nickel-63 tend to increase relative to the activity of cobalt-60. Thus the estimates to be made are reactor-specific and average values over the operating period of the reactor.

When determining the ratios between the activities of nickel-59, nickel-63 and cobalt-60 it is possible to use cobalt-58 as an reference nuclide, because it has the same parent element (nickel) as nickel-59 and nickel-63. There is also quite a strong correlation between the activities of cobalt-58 and cobalt-60, which has been frequently measured and reported. If the various factors significant to the ratio are considered (e.g. isotopic fractions, cross sections, neutron fluxes, half-lives), the following approximate ratio can be estimated for the activities of the nuclides of interest:

nickel-59: nickel-63: cobalt-60 = 0.0005: 0.1:1

Statistical analysis /6/ of the data from measurements on waste packages from Oskarshamn and Barsebäck showed, that the variations in annual activities of corrosion products are at least one order of magnitude. As far as cumulative amounts during 30 years of operation are concerned, statistical variations are smoothed out. However, variations of a systematic character will remain, and therefore it is expected that the variation range of the cumulative activities of corrosion products is about or less than one order of magnitude.

2.4.3 FISSION PRODUCTS AND ACTINIDES

Cesium-137 is selected as a reference nuclide for fission products and actinides, because it is a relatively long-lived gamma emitter and its content in fuel and its tendency to leach are quite high. Concequently, contents of cesium-137 in reactor circuits have been frequently measured and reported.

Available data on cesium-137 contents in wastes are very varying. This is partly due to statistical factors (for example operating conditions) and partly due to systematic factors (for example fuel properties). The following estimates are made for the cesium-137 activity in various resins based on experiences from several reactors /1, 2, 3, 5/, and putting the greatest emphasis on the experiences from ASEA-ATOM reactors.

RWCS	resins:	1 000	GBq/a Cs-137	(~ 30 Ci/a)
SFPCS	resins:	500	GBq/a Cs-137	(~ 15 Ci/a)

Table 2.3 summarizes the relative concentrations in fuel (calculated by the ORIGEN code for ASEA-ATOM BWRs /7/) and the estimated relative leakage coefficients /8, 9/ for the fission products. On the basis of these data, the relative activities of various fission products in RWCS and SFPCS resins have been estimated. For plutonium-239 a leakage coefficient relative to cesium-137 of 10^{-4} to 10^{-3} has been assumed.

Table 2.3 Estimate of the relative contents of various fission products in RWCS and SFPCS resins.

Nuclide	Averange activity in fuel relative to Cs-137	Estimated range of leakage coefficient relative to Cs-137	Estimated activity in RWCS and SFPCS resins relative to Cs-137
Cs-137	1	1	1
Cs-134	1 (average burnup)	1	1 (RWCS)
	1.8 (high burnup)		1.8 (SFPCS)
Sr-90	0.7	10^{-3}...0.2 (operation)	0.05 (RWCS)
		10^{-4}...0.1 (fuel pool)	0.01 (SFPCS)
Tc-99	1.4×10^{-4}	0.1...1	6×10^{-5}
Cs-135	2.7×10^{-6}	1	3×10^{-6}
I-129	3.4×10^{-7}	1...20	2×10^{-6}
Pu-239	3.0×10^{-3}	10^{-4}...10^{-3}	1×10^{-6}

The extent of fuel failures is evidently a dominating factor concerning the accumulation of fission products in resins. If there are no fuel failures, the concentration of fission products is negligible, as the operating experiences from Barsebäck and from the Finnish power plants indicate. At Oskarshamn there have been some fuel failures during several years, and the annual fission product activities vary within about two orders of magnitude /6/.

The extent of fuel failures is mainly a statistical factor, but contains also strong systematic constituents, as there has been a trend towards increasing integrity of fuel. Because of great uncertainties it is difficult to estimate the variation ranges for cumulative activities. In order to cover the major part of the uncertainties, it is reasonable to assume that the high estimate is ten times the best estimate, and that the low estimate is one tenth of the best estimate.

2.4.4 CARBON-14

Carbon-14 activity in the primary coolant of LWR's arises from several sources according to the following nuclear reactions:

- O-17 (n, α) C-14 reaction of oxygen in fuel and coolant

- N-14 (n,p) C-14 reaction of nitrogen in fuel, coolant and impurities

A rough estimate shows that only the carbon arising from the oxygen-17 in the coolant has to be considered in the present study.

The fraction of carbon-14 collected in resins compared with the amount generated in coolant is insufficiently known. Literature data vary from 0.01 % to 10 % /10/. Measurements made in Finland and Sweden /11, 12/ indicated, that 0.03 %...0.8 % of carbon-14 formed in the coolant was found in resins, and more than 95 % of Carbon-14 in resins was found as CO_2. The experiments indicated, that in PWRs the fraction of carbon-14 in resins is probably greater than in BWRs. The age of wastes seems to have a significant effect on carbon-14 concentrations, because carbon compounds tend to escape from resins during storage. It is quite possible that the carbon compounds escape from resins almost totally during the drying phase of bituminization.

In the present study, the annual amounts of carbon-14 absorbed in RWCS resins and retained in the final products are estimated quite conservatively, around 1 % of the carbon-14 generated in the coolant:

RWCS resins: 3 GBq/a/C-14 (~ 100mCi/a)

SFPCS resins: 0.1 GBq/a/C-14 (~ 3mCi/a)

It must be emphasized that the figures given above should probably be reduced even by orders of magnitude, as chemical and physical conditions in the primary circuits or conditions during processing of wastes favour the escape of carbon compounds.

2.4.5 SUMMARY OF NUCLIDE INVENTORY

Table 2.4 summarizes the estimates of the average annual activities from a 500 MW (e) BWR. Table 2.5 summarizes the estimated cumulative activities from six 500 MW (e) BWR after an operating period of 30 years and a storage period of 5 years.

Table 2.4 Estimated average annual activities of RWCs and SFPCS resins from a 500 MW (e) BWR (original activity).

Nuclide	Decay constant	RWCS resins per reactor		SFPCS resins per reactor	
		GBq/a	Ci/a	GBq/a	Ci/a
C-14	$1.21x10^{-4}$	3	0.1	0.1	0.003
Ni-59	$8.66x10^{-6}$	1.5	0.05	0.1	0.003
Ni-63	$7.53x10^{-3}$	300	10	20	0.5
Co-60	$1.31x10^{-1}$	3,000	100	200	5
Sr-90	$2.43x10^{-2}$	50	1.5	5	1.5
Tc-99	$3.30x10^{-6}$	0.05	0.001	0.03	0.001
I-129	$4.08x10^{-8}$	0.002	0.00005	0.001	0.00003
Cs-134	$3.30x10^{-1}$	1,000	30	800	2
Cs-135	$2.31x10^{-7}$	0.003	0.0001	0.002	0.00005
Cs-137	$2.31x10^{-2}$	1,000	30	500	15
Pu-239	$2.84x10^{-5}$	0.001	0.00003	0.0005	0.00001

Table 2.5 Estimated cumulative activities of RWCS and SFPCS resins from six 500 MW (e) BWRs after an operating period of 30 years and a storage period of 5 years.

Nuclide	Decay constant (a^{-1})	RWCS resins per 6 reactors		SFPCS resins per 6 reactors	
		GBq/30a	Ci/30a	GBq/30a	Ci/30a
C-14	$1.21x10^{-4}$	540	15	18	0.50
Ni-59	$8.66x10^{-6}$	270	7	18	0.50
Ni-63	$7.53x10^{-3}$	52,000	1,400	3,300	90
Co-60	$1.31x10^{-1}$	70,000	1,900	4,500	120
Sr-90	$2.43x10^{-2}$	5,800	160	580	16
Tc-99	$3.30x10^{-6}$	9.0	0.25	5.4	0.15
I-129	$4.08x10^{-8}$	0.36	0.010	0.18	0.005
Cs-134	$3.30x10^{-1}$	4,000	110	3,300	90
Cs-135	$2.31x10^{-7}$	0.54	0.015	0.36	0.01
Cs-137	$2.31x10^{-2}$	120,000	3,200	58,000	1,600
Pu-239	$2.84x10^{-5}$	0.18	0.005	0.09	0.002

2.5 WASTE PACKAGES

2.5.1 PRIMARY PACKAGING CONTAINERS

Several types of containers are utilized for packaging of reactor wastes. In /13/ the following main types were identified:

- steel drums
- steel drums with concrete lining
- large steel containers
- disposable concrete containers
- wooden boxes

These types appear in a variety of sizes and shapes. For the reference system the following container types have been chosen:

- for bituminized waste:		0.2 m^3 steel drum
- for cementized waste:		0.2 m^3 steel drum
	or:	1 m^3 concrete container

The 0.2 m^2 drum is by far the most common type used for bituminized waste /13/. For cementized waste, the 0.2 m^3 steel drum is the most common type /13/, but in Sweden and Finland concrete moulds of cubic or cylindrical shape are in use or will probably be used for cementized waste (Ringhals, Oskarshamn, Loviisa).

Figure 2.1 gives a schematic presentation of the reference containers in the present study.

2.5.2 STEEL DRUMS FOR BITUMINIZED OR CEMENTIZED WASTE

The construction material of the steel drum is mild carbon steel. The dimensions of the drum are height 875 mm, inner diameter 571 mm, wall thickness 1 mm and filling volume about 0.2 m^3.

The side and bottom seams are welded. The closure mechanism in the case of bituminized waste is a ∅ 300 mm lid tightened remotely. In case of cementized waste a lid and a bolted closure ring is assumed. The inside and outside surfaces are assumed to be coated with zinc in order to reduce corrosion in storage. No reinforcements other than corrugation of the steel plate are utilized.

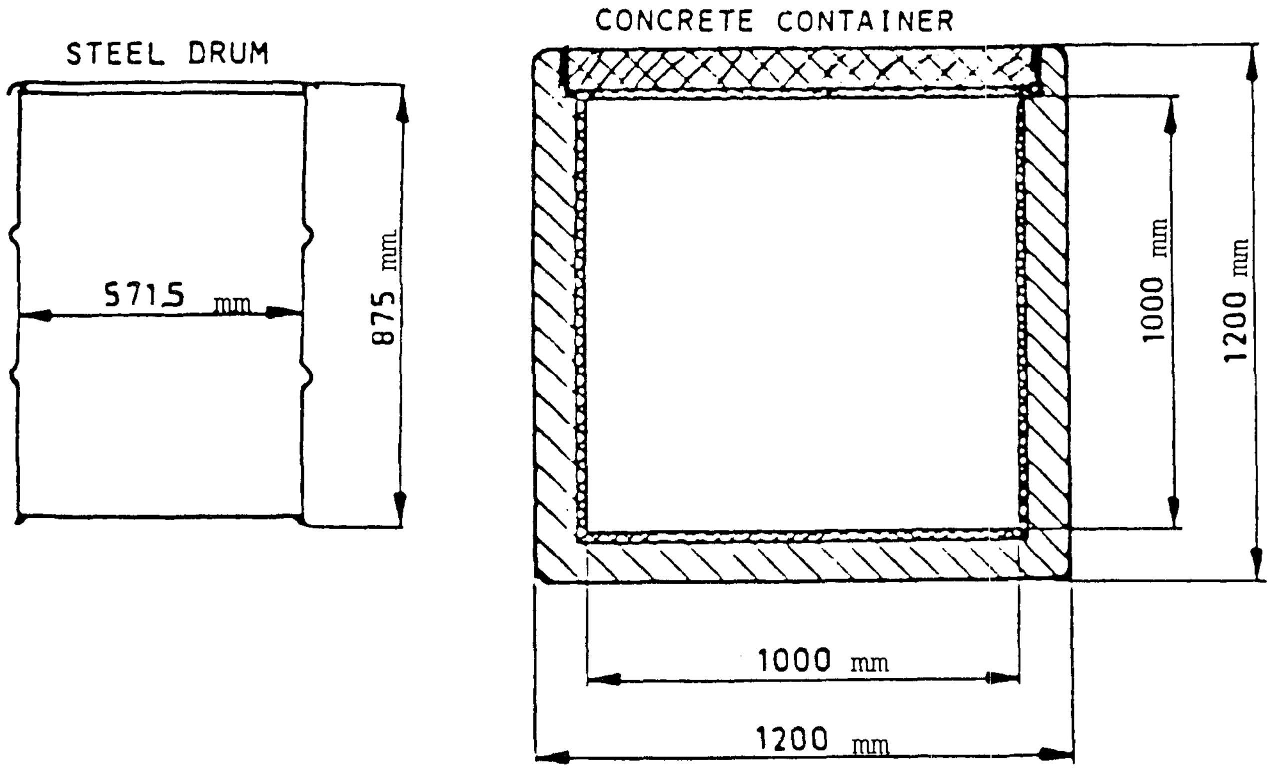


Figure 2.1 Containers used in packaging solidified reactor wastes.

2.5.3 CONCRETE MOULD FOR CEMENTIZED WASTE

The concrete mould selected for the reference system is the Swedish cubic type. In Finland probably a cylindrical concrete container with an optional inner steel lining will be used, but at present it is only in the design stage.

The construction material of the mould is reinforced concrete with a compressive strength of 40 MPa. The outer dimensions of the cube are 1.2 x 1.2 x 1.2 m, the wall thickness is 100 mm and the inner volume is 1 m^3. The total volume of the package is 1.7 m^3 and the total weight about 4 t. The waste is solidified with cement by mixing in the mould. The mould is closed by pouring a concrete lid on top of the waste mixture.

In order to avoid the cracking of the mould due to possible swelling of the solidification product, a cellular plastic lining (about 10 mm) inside the mould is assumed.

2.5.4 WASTE CONTENT IN DIFFERENT PACKAGES

The amount of dry resins in different packages is determined by the following factors:

- the net volume of the waste package (m^3)
- the ratio dry resin mass to unit volume of the final product (kg/m^3)

Table 2.6 summarizes the relevant parameters and waste content per package for different packages and waste types.

2.5.5 NUCLIDE CONTENTS IN DIFFERENT PACKAGES

Tables 2.7 and 2.8 summarize the average nuclide contents in the different packages for waste stored for 5 years and for all waste 5 years after termination of the operating period of 30 years.

2.5.6 RADIATION DOSE RATE FROM DIFFERENT PACKAGES

The radiation dose outside a package is caused only by gamma emitters. Among the nuclides considered only cesium-134, cesium-137 and cobalt-60 are significant gamma emitters.

Table 2.6 Amount of dry RWCS and SFPCS resins in different packages.

Package type	Net volume (m^3)	Waste type	Dry resin mass to unit vol. of solid prod. (kg/m^3)	Dry resin mass per package (kg)	Annual number of packages per reactor
Drum of bituminized waste	0.2	RWCS resin	400	80	30
		SFPCS resin	400	80	12.5
Drum of cementized waste	0.2	RWCS resin	200	40	60
		SFPCS resin	250	50	20
Concrete mould of cementized waste	1.0	RWCS resin	200	200	12
		SFPCS resin	250	250	4

Table 2.7 Average nuclide content in different packages.
Age of waste about 5 years.

Package type	Drum of bituminized waste (MBq/package)		Drum of cementized waste (MBq/package)		Concrete mould of cementized waste (MBq/package)	
Nuclide	RWCS resins	SFPCS resins	RWCS resins	SFPCS resins	RWCS resins	SFPCS resins
C-14	100	8	50	5	250	25
Ni-59	50	8	25	5	130	25
Ni-63	10,000	1,600	5,200	1,000	26,000	5,200
Co-60	52,000	8,300	26,000	5,200	130,000	26,000
Sr-90	1,500	360	750	220	3,800	1,100
Tc-99	1.7	2.4	0.85	1.5	4.2	7.5
I-129	0.07	0.08	0.03	0.05	0.17	0.25
Cs-134	6,600	14,000	3,300	8,600	17,000	43,000
Cs-135	0.1	0.16	0.05	0.1	0.25	0.5
Cs-137	30,000	36,000	15,000	22,000	75,000	110,000
Pu-239	0.03	0.04	0.02	0.08	0.08	0.13

Note: 1 MBq = 0.027 mCi

Table 2.8 Average nuclide content in different packages after an operating period of 30 years and a storage period of 5 years.

Package type	Drum of bituminized waste (MBq/package)		Drum of cementized waste (MBq/package)		Concrete mould of cementized waste MBq/package)	
Nuclide	RWCS resins	SFPCS resins	RWCS resins	SFPCS resins	RWCS resins	SFPCS resins
C-14	100	8.0	50	5.0	250	25
Ni-59	50	8.0	25	5.0	130	25
Ni-63	9,300	1,500	4,700	920	30,000	4,600
Co-60	13,000	2,000	6,500	1,300	32,000	6,300
Sr-90	1,100	260	550	160	2,700	800
Tc-99	1.7	2.4	0.85	1.5	4.2	7.5
I-129	0.07	0.08	0.03	0.05	0.17	0.2
Cs-134	740	1,500	370	920	1,900	4,600
Cs-135	0.1	0.16	0.05	0.1	0.25	0.5
Cs-137	22,000	26,000	11,000	16,000	56,000	8,000
Pu-239	0.03	0.04	0.02	0.03	0.08	0.13

Note: 1 MBq = 0.027 mci

The radiation dose rates outside the different packages at a distance of 1 m from the surface were calculated for various nuclides using the computer codes CYLDOS and RECDOS /14/. Table 2.9 summarizes the dose rates per 1 Bq for each nuclide.

On the basis of tables 2.7 and 2.9 the average dose rates outside the different packages of 5 year old wastes can be calculated; these are given in table 2.9. In addition, high and low estimates for the dose rates are given; these are based on the variation ranges of corrosion and firrion products discussed in chapters 2.4.2 and 2.4.3.

Table 2.9 Dose rate at 1 m from the surface of different packages containing 1 Bq of Co-60, Cs-137 or Cs-134.

Package type	Dose rate at 1 m for 1 Bq of Cs-134 (mSv/h)	Dose rate at 1 m for 1 Bq of Cs-137 (mSv/h)	Dose rate at 1 m for 1 Bq of Co-60 (mSv/h)
Drum of bituminized waste	7.5×10^{-11}	2.6×10^{-11}	1.2×10^{-10}
Drum of cementized waste	5.4×10^{-11}	1.9×10^{-11}	8.1×10^{-11}
Concrete mould of cementized waste	8.1×10^{-12}	2.8×10^{-12}	1.5×10^{-11}

NOTE: 1 mSv/h = 0.1 rem/h

Table 2.10 Estimates for dose rates at 1 m from surfaces of different packages.

Package type	Waste type	Dose rate 1 m from surface of package (mSv/h)		
		low estimate	best estimate	high estimate
Drum of bituminized waste	RWCS resins	2	7.5	30
	SFPCS resins	0.5	3.0	23
Drum of cementized waste	RWCS resins	0.75	2.5	12
	SFPCS resins	0.25	1.3	10
Concrete mould of cementized waste	RWCS resins	0.7	2.3	10
	SFPCS resins	0.2	1.0	7.5

Note: 1 mSv/h = 0.1 rem/h

2.6 DIFFUSION AND LEACHING CHARACTERISTICS OF THE CONDITIONED WASTE

The leaching of radionuclides from the waste is supposed to be governed by diffusion. This implies that the leach rate from a waste unit will diminish with the square root of leaching time, a behaviour which is often observed in experimental leaching tests of cementized and bituminized materials (Technical Part III). The relationship between leach rate $R_{(t)}$ and the diffusion coefficient D (= leach coefficient L) is

$$R_{(t)} = \sqrt{\frac{D}{\pi t}}$$

for units of large dimensions compared with the thickness of the leached layer.

Values of selected diffusion coefficients for the nuclides which have been found to be of interest in the safety analysis are given in table 2.11. To facilitate comparison with the values from literature the corresponding leach rates R_{100} at t = 100 days calculated from the formula above are also given. The values are in general regarded as conservative resulting from leach rates which are about a factor of 10 higher than typical experimental results for cementized ionexchange resins in deionized water /22, 23/ or bituminized ion exchange resins even in strong salt solutions. The values given for diffusive leaching of carbon-14, iodine-129 and probably also technetium-99 as TcO_4 from cementized waste are probably too conservative (Technical Part III, table 4.2.). Experimental results for leaching of these isotopes from bitumen products are not available.

Table 2.11 Assumed diffusion coefficients and calculated leach rates for important nuclides /15, 16/.

Nuclide	Diffusion coefficient (m^2/a)		Leach rate at 100 days (m/d)	
	cement	bitumen	cement	bitumen
Carbon	$3x10^{-4}$	$2x10^{-6}$	$5x10^{-5}$	$4x10^{-6}$
Nickel	$4x10^{-9}$	$4x10^{-9}$	$2x10^{-7}$	$2x10^{-7}$
Technetium	$6x10^{-4}$	$2x10^{-6}$	$7x10^{-5}$	$4x10^{-6}$
Iodine	$6x10^{-4}$	$2x10^{-6}$	$7x10^{-5}$	$4x10^{-6}$
Cesium	$3x10^{-4}$	$1x10^{-6}$	$5x10^{-5}$	$3x10^{-6}$
Strontium	-	-	$1x10^{-6}$	$1x10^{-6}$
Cobalt	-	-	$1x10^{-7}$	$1x10^{-7}$

In reality there may not be much difference in the leach rates for the two materials. The importance of variations in diffusion coefficients for the result of the safety analysis can be checked by parameter variation using the formulas given in Technical Part II chapter 6. Such sensitivity calculations may also to some degree cover the risk of change in the diffusion coefficients with time due to the possibility of gradual degradation of the soldified materials.

2.7 FIRE CHARACTERISTICS

To assess the risks in the connection with fire the following assumptions were used for bituminized waste* /17, 18/:

During the fire

- 100 % of the cesium is released as gas.

* Results of the latest experiments performed within the project have indicated that the assumptions are very conservative. (See discussion in Technical Part III, chapter 6).

Of the remaining nuclides

- 60 % is released as gases or as particles smaller than 10 μm
- 25 % is released as particles larger than 10 μm and
- 15 % will remain in the ashes.

The heat of combustion for bitumen is supposed to be equal to that of heavy fuel oil, about 40 MJ/kg. The heat of combustion for the anion/cation resin mixture is supposed to be about 20 MJ/kg /19/. From these values the obtained heat of combustion for bitumen/resin mixture (1:1) is 30 MJ/kg.

2.8 IMPACT CHARACTERISTICS

In order to assess the possible risks in connection with the transport accidents some fall tests with fullscale (0.2 m^2) drums and small specimens were performed. On the basis of these tests it can be estimated, that 5 weight-% of the particles formed in the tests (cementized waste products) are smaller than 10 μm /20/.

2.9 COMPRESSIVE STRENGTH OF THE CEMENTIZED PRODUCT AND THE MOULDS

- 10 weight-% dry resin in products 30 MPa
- 15 " " " " " 20 MPa
- 20 " " " " " 10 MPa

The compressive strength of the moulds is 40 MPa immediately after processing and is supposed to be about 35 MPa after 50 years of storage. The compressive strength of cementized waste products is supposed to decrease 50 % after water test of one hundred days /21/.

2.10 SWELLING IN WATER

In the performed tests the following swelling properties for the solidification products were verified /21, 22, 24/

- swelling in water
  - cementized products* 0.3 volume-%
  - bituminized products** 5.0 volume-%

* If the swelling of the cementized products in water exceeds approximately 0.5 volume-% risk for total destruction of products exists.

** The result of the latest experiments completed within the project have indicated that the swelling could be significantly more than 5 % (see discussion in Technical Part III, chapter 6).

3 SITE DATA

3.1 GENERAL

The safety analysis reported in Technical Part II is not related to any particular place, and the dose calculations perfomed within the Nordic study should primarily be seen as examples illustrating how such analysis can be performed. On the other hand, the examples are also selected to provide an impression of the nature and the magnitude of the risk related to storage, transport and disposal of the low- and medium level waste from nuclear power plants.

For calculation of the consequences of certain events with release to the atmosphere, data on meteorology, population distribution and certain data on land use must be available. For this study, site data for one of the two locations used in the Norwegian reactor accident analysis /25/ was used as reference. The selected site is located in the Oslofjord area and represents an area with a moderate population density, and the fraction of the land area used for agriculture is also an average value. The reference site is thus neither typical of the areas with intense agriculture found, for instance, in Denmark and central Sweden, nor is it typical of the forest or mountain areas found, in particular, in the middle and Northern parts of the Nordic countries. However, the reference site is reasonably representative for large areas of the Nordic countries.

3.2 METEOROLOGY

In order to calculate radiation doses caused by activity-release to the atmosphere, the following three types of site-specific meteorological data have been used in the calculations:

- Mixing heights
- Wind direction distribution
- Hour-by-hour wind speed, stability and rain

The mixing heights of the reference site are specified for two stability classes, stable and non-stable, and for the four seasons. Pasquill E and F are defined as the stable categories in this connection, and the others as non-stable. The four seasons are three months each, with winter including December, January and February etc. The mixing height value have been determined by NILU (Norwegian Institute for Air Research) from radio-sounding measurements /26/. The values are given in the table 3.1.

Table 3.1. Mixing height (meters) for the Oslofjord area.

	Spring	Summer	Fall	Winter
Stable	300	200	200	230
Nonstable	1,300	1,700	900	350

The wind direction distribution has also been determined by NILU from measurements conducted at Brenntangen in the Oslofjord area /26/. The distributions are given in the table 3.2. They are given in sixteen 22 $1/2^{o}$ sectors, where the number under N signifies the probability that the wind is blowing from North $\pm$ 11.25^{o}. The numbers are in percent, and there are four distributions in the table, one for each of the seasons.

Hour-by-hour meteorological data for one whole year is used in the calculations. For each hour the wind speed (m/s), stability (Pasquill class), and rain indication are given. For a location at the Oslofjord (Brenntangen), such meteorological data file was compiled by NILU /26/. However, for practical reasons, this large volume of data is not included in the present report.

Table 3.2. Wind direction distribution (%) in the Oslofjord area.

	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
Spring	11.2	13.4	5.0	3.1	2.0	1.6	3.4	5.7	14.6	16.0	8.7	3.3	2.5	2.2	2.2	5.2
Summer	6.5	6.9	4.8	2.3	1.9	3.1	5.3	7.8	15.1	17.4	10.5	3.6	2.7	1.9	4.5	5.7
Fall	15.2	11.1	5.6	2.6	1.5	1.3	3.2	5.3	15.2	18.3	5.9	2.7	1.1	1.5	3.0	6.5
Winter	8.4	16.1	7.4	3.4	3.0	3.9	4.4	6.4	18.0	17.8	4.0	2.0	1.9	0.7	1.1	1.4

3.3 POPULATION DISTRIBUTION

The population distribution for the Oslofjord site is shown in table 3.3 which is taken from /25/.

While the Oslofjord site could be representative as a site for a storage or disposal facility as well as a location for a transport accident, a transport accident may, of course, also occur in a heavily populated area. In order to perform dose calculations for a transport accident in a town, leading to release of radionuclides to the atmosphere, the actual population distribution within a radius of 9.5 km was replaced by a population of 1763 persons per km^2, simulating a city of 1/2 million inhabitants and a radius of approximately 10 km. For the population distribution outside the 9.5 km radius, the data shown in table 3.3 were maintained. This partly hypothetical site is designed as the "city site".

A rough comparison of the population distribution around the two sites is shown in figure 3.1.

All other data remain the same for the two sites.

For reference, the population distribution around the nuclear power plants at Ringhals, Barsebäck, Loviisa and Olkiluoto is also shown in figure 3.1.

Table 3.3. Population distribution. Oslofjord site.

SECTOR / DIST.KM	EAST						SOUTH					WEST				NORTH
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16
0- 0,8	-	-	-	3	-	3	9	-	-	-	-	-	-	-	-	4
0,8 1,6	17	-	11	-	6	7	6	-	7	15	4	2	17	-	28	24
1.6 2,4	12	6	53	17	8	28	3	14	-	-	-	-	-	27	12	9
2,4 3,2	33	198	12	12	3	71	32	114	20	232	-	-	-	39	157	105
3,2 4,0	29	165	22	39	-	-	83	172	106	-	-	-	-	15	114	180
4,0 4,8	96	349	83	25	9	-	3	-	-	-	-	-	-	-	114	92
4,8 5,6	200	42	211	30	21	18	24	-	-	-	-	-	-	-	36	105
5,6 6,4	18	60	72	24	21	6	-	-	-	-	-	-	-	-	1919	940
6,4 7,2	3	15	18	72	75	12	-	-	-	-	-	-	-	-	1800	1361
7,2 8,0	12	9	9	60	66	273	126	-	-	-	-	-	-	-	2217	1845
8,0 9,5	30	12	-	105	15	300	444	-	-	-	-	-	-	93	9090	7725
9,5 11,5	15	24	3	1655	191	265	847	-	-	20	950	400	-	6	5004	3385
11,5 13,5	213	131	6	318	156	200	357	-	-	3000	990	1265	10487	-	36	271
13,5 16,0	120	60	2	575	513	533	170	-	-	3752	1063	430	7088	-	45	355
16,0 20,0	500	340	281	400	757	15875	710	-	-	16870	9888	710	1280	260	-	1344
20,0 24,0	354	181	193	300	19060	38000	692	-	373	6700	1950	1180	1010	70	2558	1050
24,0 28,0	418	179	384	1088	20218	3500	591	-	530	2900	1750	900	1540	5600	843	2506
28,0 32,0	855	839	1103	1400	6793	1028	590	-	1363	1530	1790	770	3310	735	2081	993
32,0 40,0	3848	12677	4695	1200	2438	1700	1200	-	957	30856	1660	690	2121	2991	3385	13468
40,0 48,0	8108	4294	3000	2000	25000	500	500	-	-	12576	2048	1261	800	3000	12074	15000
48,0 56,0	6053	3000	2000	1000	3000	1000	5000	-	-	15701	1851	1000	700	40638	44841	23522
56,0 64,0	36812	7417	1000	1000	500	2000	1000	-	-	4000	62914	702	900	23845	5000	346344
64,0 72,0	34888	2000	3000	500	500	2000	1000	-	-	2000	10991	590	15254	5000	4000	14800
72,0 80,0	6093	1300	1000	4000	1000	1000	2000	-	-	2000	1000	2715	2500	6000	5344	7739
80,0 88,0	5392	500	1000	3000	5500	2000	1000	-	-	4000	200	5500	8970	1390	2500	6000
88,0 100,0	25405	1500	4000	1500	4500	2000	5500	-	-	6000	4518	5211	4000	2500	27456	5218
100,0 115,0	19513	6500	21000	2500	7500	5000	10500	-	-	8060	3348	4274	3955	2837	2998	16047
115,0 140,0	14284	17500	9000	32500	14500	15000	24000	-	-	16539	2488	3617	3950	3200	1755	15314
140,0 175,0	35383	15500	24500	74500	25500	131000	33500	-	-	48021	3222	4112	3850	3449	17012	108237
175,0 240,0	23000	28000	42000	95000	151500	197500	610000	166000	5000	56007	4881	6333	4125	10601	11918	49982
240,0 310,0	10000	50000	189000	66550	155500	215000	100000	290500	6500	4047	52380	241237	21000	34668	24964	16980
310,0 400,0	17000	71000	500000	639000	186000	261000	1334400	763000	27000	-	-	10000	33000	72000	42000	24000
400,0 500,0	109000	196000	78000	1093500	115000	273000	950000	1294000	20000	-	-	-	-	110000	131000	306000

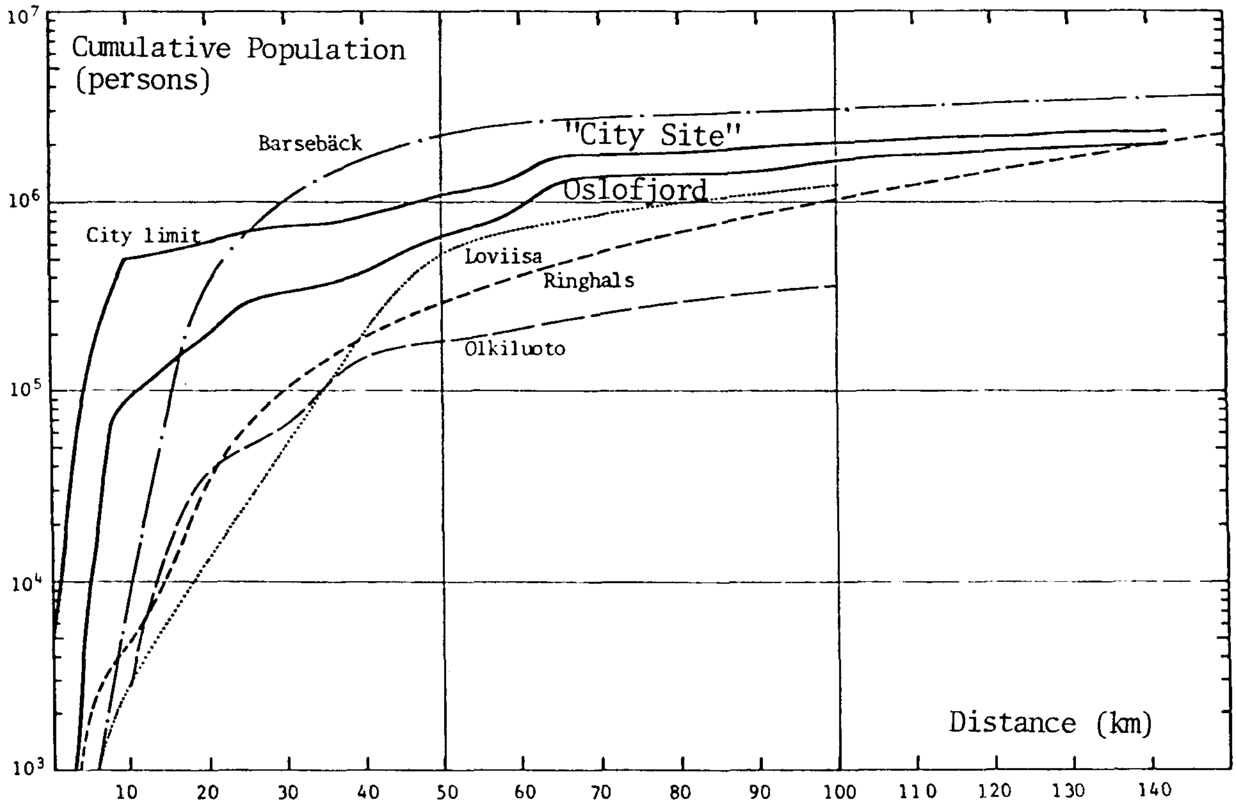


Figure 3.1. The population distribution around the Oslofjord site and the "City Site". Population around Ringhals, Barsebäck, Loviisa and Olkiluoto is shown for comparison.

3.4. RADIOECOLOGICAL DATA

The fractions of habitable land for the site are shown in percentages in table 3.4.

The fraction of habitable land that is farm land is also used in the calculations and was determined to be 0.76 over the whole area, with exception of the sector elements including Oslo and Gothenburg, where it is set equal to 0.001. The fraction of farm land that is dairy land is set equal to 0.086 /25/.

Table 3.4 Fraction of habitable land (%). Oslofjord /25/.

DIST.KM	SECTOR	EAST 1	2	3	4	5	6	SOUTH 7	8	9	10	WEST 11	12	13	14	NORTH 15	16
0-	0,8	–	–	20	40	20	80	–	–	–	20	20	20	–	–	30	10
0,8	1,6	60	60	30	40	10	30	80	40	10	60	–	–	10	–	50	60
1,6	2,4	50	20	40	60	70	10	30	10	30	–	–	–	–	–	20	50
2,4	3,2	30	40	60	30	40	40	10	20	30	–	–	–	–	–	40	60
3,2	4,0	80	80	100	90	20	10	20	40	–	–	–	–	–	–	–	70
4,0	4,8	100	90	100	80	40	–	–	10	–	–	–	–	–	–	10	80
4,8	5,6	80	60	90	80	70	20	10	–	–	–	–	–	–	–	–	80
5,6	6,4	60	80	70	80	40	30	10	–	–	–	–	–	–	–	–	60
6,4	7,2	–	60	80	70	50	10	–	–	–	–	–	–	–	–	–	90
7,2	8,0	–	10	60	90	60	20	–	–	–	–	–	–	–	–	–	60
8,0	9,5	–	10	–	50	70	30	10	–	–	–	–	–	10	–	50	60
9,5	11,5	20	10	–	40	30	20	20	–	–	–	10	–	–	–	40	10
11,5	13,5	20	20	–	40	70	30	30	–	–	10	50	50	50	–	–	10
13,5	16,0	10	10	–	30	50	80	20	–	–	30	60	60	40	–	–	10
16,0	20,0	30	30	10	20	30	60	10	–	–	30	70	60	50	–	–	20
20,0	24,0	30	80	30	20	70	70	10	–	–	40	60	60	60	–	10	30
24,0	28,0	40	10	20	40	80	30	–	–	–	30	40	40	80	10	20	30
28,0	32,0	40	10	50	20	60	30	–	–	–	30	50	30	30	10	10	30
32,0	40,0	60	70	60	20	30	10	–	–	–	40	20	10	20	20	10	60
40,0	48,0	30	40	40	50	20	10	–	–	–	30	10	20	10	20	20	20
48,0	56,0	10	40	10	10	10	–	10	–	–	10	–	–	10	10	40	10
56,0	64,0	10	20	10	–	–	–	20	–	–	10	20	–	10	20	30	50
64,0	72,0	30	20	20	20	–	10	20	–	–	–	10	–	–	10	10	40
72,0	80,0	50	20	20	10	–	20	30	–	–	–	–	10	–	10	10	10
80,0	88,0	50	20	10	10	10	20	20	–	–	–	–	10	–	10	30	10
88,0	100,0	40	20	20	–	10	20	30	–	–	–	–	10	10	10	20	–
100,0	115,0	20	20	20	20	20	30	40	–	–	–	–	–	–	–	10	20
115,0	140,0	10	10	40	50	20	60	40	–	–	–	–	–	–	–	–	10
140,0	175,0	20	10	40	30	20	60	40	–	–	–	–	–	–	–	–	30
175,0	240,0	10	10	10	50	60	50	40	30	–	–	–	–	–	–	–	–
240,0	310,0	–	10	30	70	40	20	30	40	40	–	–	–	–	–	–	–
310,0	400,0	–	10	10	40	30	30	40	50	50	–	–	–	–	–	–	–
400,0	500,0	10	10	20	30	10	20	50	50	20	–	–	–	–	–	–	10
500,0	600,0	–	–	–	–	10	–	10	50	–	–	–	–	–	–	–	–

3.5 GEOLOGY

The key parameter for the reference sites used in the analysis are given in table 3.5.

For the shallow land burial and the concrete bunker, two types of soil have been considered:

- sandy till, and
- clayey till.

The rock cavern is assumed to be situated in an extensive formation of bedrock mainly consisting of granites and gneisses of various kinds.

3.6 HYDROLOGY

The key parameters for the reference sites are given in table 3.5.

The reference site for shallow land burial and concrete bunker is situated in an area with a smooth inclination giving only a moderate movement of surface water and ground water, but still allowing a good drainage of the area.

The real ground water velocity can be calculated on the basis of the geological data given in table 3.5:

- in sandy till 5.2 m/a
- in clayey till 0.16 m/a

The rock cavern has a rock cover of about 30 m. The velocity of the ground water in the cracks is calculated in accordance with the data given in table 3.5:

- In rock formations 2.1 m/a

Table 3.5 Reference site parameters for geology and hydrology.

	Sandy till	Clayey till	Rock
Hydraulic conductivity, K_p (m/s)	$1 \cdot 10^{-7}$	$1 \cdot 10^{-9}$	$1 \cdot 10^{-8}$
Hydraulic gradient, i (m/m)	$5 \cdot 10^{-2}$	$5 \cdot 10^{-2}$	$2 \cdot 10^{-2}$
Kinematic porosity, E_k	$3 \cdot 10^{-2}$	$1 \cdot 10^{-2}$	$3 \cdot 10^{-3}$
Diffusion porosity, E_d	0.4	0.2	-
Dry particle soil density, ρ_{tp} (kg/m^3)	2,650	2,650	-
Geometric surface area, a_2 (m^2/kg)	-	-	30
Fracture spacing, s (m)	-	-	0.4
Distribution coefficient, K_d (m^3/kg) for:			
- carbon	0	0	0
- technetium	0	0	0
- iodine	0	0	0
- cesium	0.1	0.5	0.1
- strontium	0.02	0.1	0.01
- cobalt	0.1	1	0.1
- nickel	0.1	1	0.1
Retention factor*, K_i for:			
- carbon	1	1	1
- technetium	1	1	1
- iodine	1	1	1
- cesium	400	5,300	420
- strontium	80	1,060	42
- cobalt	400	10,600	420
- nickel	400	10,600	420

* Calculated from the K_d-values

4 STORAGE

4.1 GENERAL

The facilities for temporary storage designed and built in different countries exhibit a wide range of technical solutions. The three different designs chosen for the. Reference Systems in this Nordic study reflect design solutions in use at various Swedish nuclear power stations.

The selected storage facilities are:
- A storage building for bituminized waste
- A storage building for cementized waste
- An underground rock cavern for storage of both bituminized and cementized waste.

The last is in principle identical to the rock cavern for disposal, as the cavern during the operating period and until final sealing could be considered as a temporary storage facility.

The different alternatives are described in the following subsections.

4.2 STORAGE BUILDING FOR BITUMINIZED WASTE

The storage building is a concrete structure and consists mainly of a hall for handling the waste drums, pits for storing the waste drums, a transport airlock and some space for auxiliary equipment, see figure 4.1.

The waste handling hall is a relatively light concrete structure covering the total area of the waste storage building. The waste drums are stored in pits - or wells - below the floor of the hall. The pits are surrounded by heavy concrete shielding, and each pit is covered on top by a removable 80 cm thick concrete cover. The top of the covers are flush with the floor of the hall, and actually

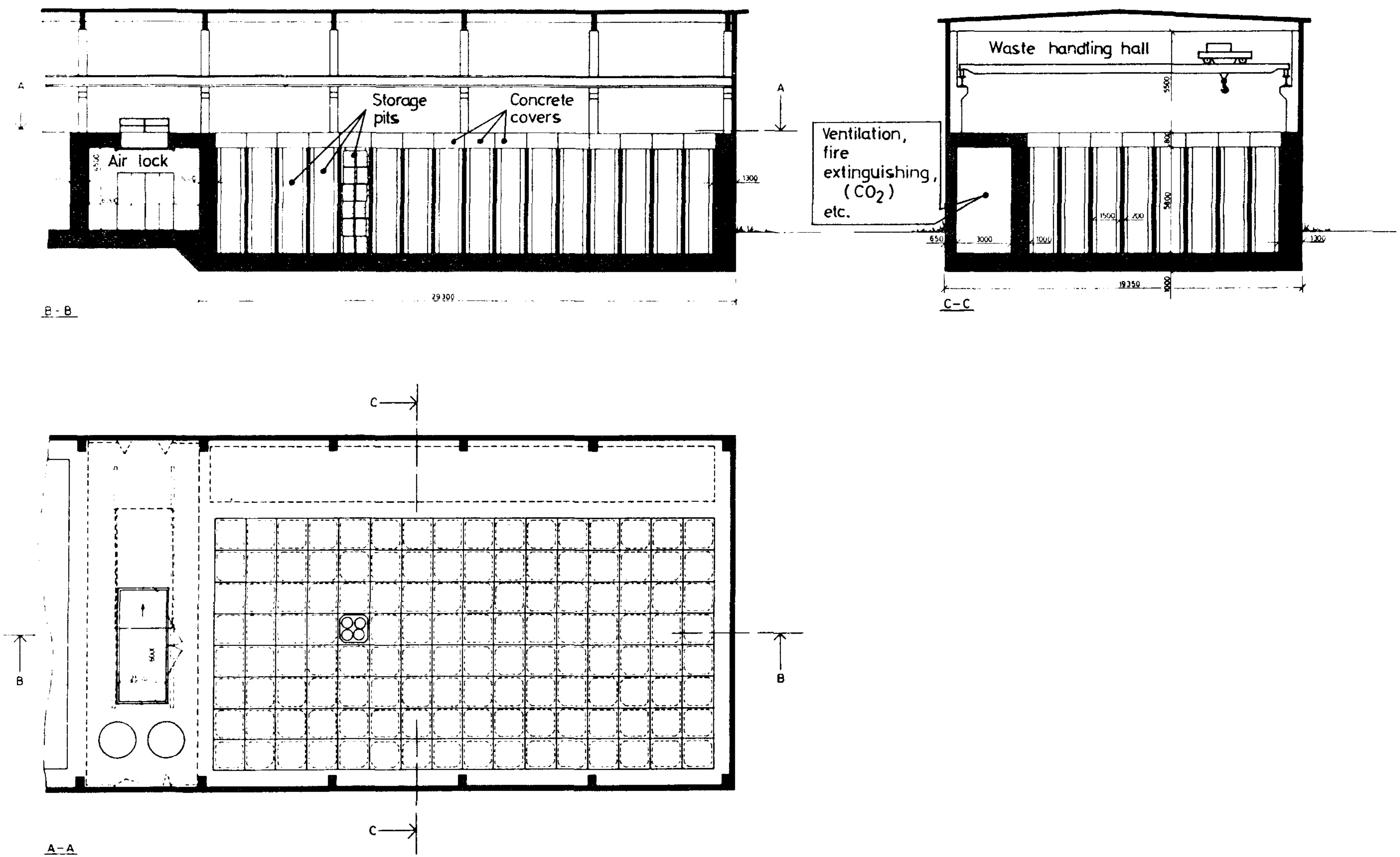


Figure 4.1 Storage building for waste incorporated in bitumen

compose the major part of the floor area. The hall is serviced by a travelling overhead crane, and all handling of the waste drums as well as of the covers is done by means of this crane. Mirrors and a closed circuit TV-system permits semi-remote operation of the crane.

Each storage pit has a cross section sufficient for storage of 4 drums side by side, and a depth sufficient for piling 6 drums on top of each other, thus each pit has a storage capacity of 24 drums. In order to stabilize the drum-piles, a steel plate corresponding to the cross section of the pit is placed for every second of third layer of drums.

The waste drums are transfered to and from the storage via a transport airlock. This airlock is located below the hall floor, and a sliding hatch provides connection between the airlock and the hall above. The waste drums are transported to the storage building by means of a fork lift. If required, the drums are transported in a bell-shaped shielded handling container with a removable bottom. The handling container with the waste drum is lifted - by means of the overhead crane - from the airlock through the hatch, and transferred to the designated storage pit. The container is positioned and lowered into the pit, the drum is released and the transport container removed from the pit for repeated use.

The maximum lifting height during handling of the drums is about 7 m (from the hall floor to the bottom of the pit).

Transportation of waste drums away from the storage (e. g., for disposal) will be by truck or by boat. For mass transportation of waste drums, transport containers will normally be used (see chapter 5).

The storage building is equipped with a forced ventilation system. Air is extracted from the bottom of each storage pit, filtered, dried, monitored and returned to the hall. The ventilation system ensures dry atmospheric conditions in the storage, thus preventing corrosion of the steel drums which would be caused by high humidity. At the same time, the radiation level due to airborne activity in the hall, airlock and other operational areas, is monitored and controlled.

Because of the fire hazard from the large volumes of bitumen stored in the building, the storage is equipped with smoke detectors and automatic fire fighting equipment.

4.3. STORAGE BUILDING FOR CEMENTIZED WASTE

The storage consists of a fairly simple concrete building, with dimensions approximately 40 m x 30 m. For shielding, the walls are 70 cm thick and the roof 30 m thick. Except for a shielding wall at the entrance and the columns supporting the roof, the total floor space is open and available for storage, see figure 4.2.

The ventilation in the storage building is by natural circulation through labyrint shielded vent openings in the side walls on both sides of the building. Neither filtering of the air nor any other air conditioning is provided for.

All handling of the waste packages in the storage is by means of a fork lift with a shielded cabin. The maximum lifting height is about 6 m allowing piling of the packages up to a total height of about 7 m (roof elevation is approximately 8 m above the floor).

Transfer of the waste packages from the radwaste treatment plant to the storage will utilize the above mentioned fork lift. Transportation away from the storage to the repository will be by truck or by boat. Transport containers will normally be used (see chapter 5).

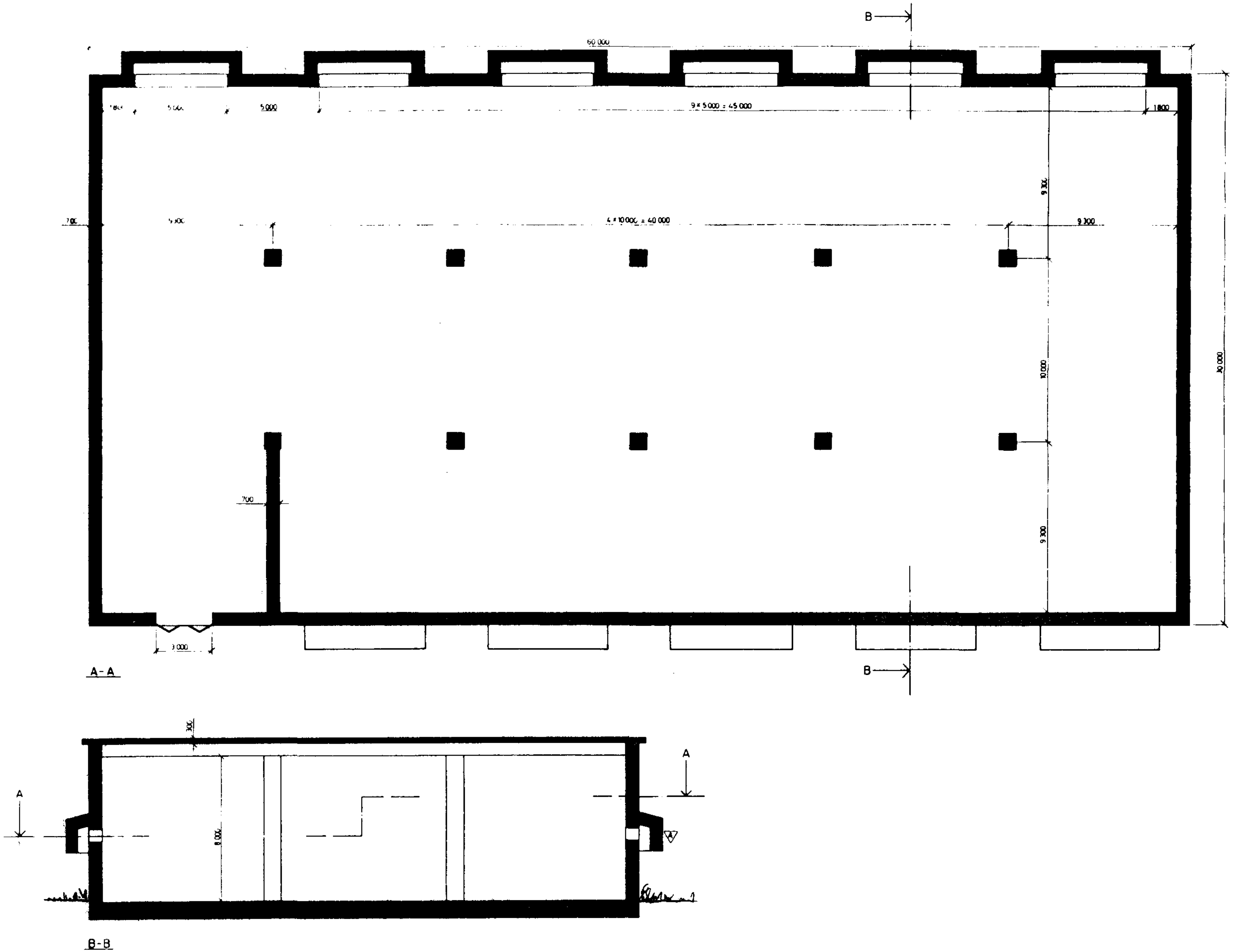


Figure 4.2 Storage building for waste incorporated in cement.

4.4. ROCK CAVERN STORAGE

As mentioned in section 4.1, the temporary rock cavern storage is identical to the rock cavern for final disposal. The cavern is in principle almost identical to the original ALMA concept for rock cavern storage of low and medium level radioactive waste /27/, /28/, however, some minor modifications have been assumed for this study.

The rock cavern storage system consists of 3 main parts: Acces Tunnels, Storage Hall and Service Section, see figures 4.3. and 4.4.

The Access Tunnel has a cross section of approximately 35 m^2 permitting transportation of the container types foreseen in the ALMA-project for transportation by boat, see chapter 5. If such large containers are not used, the tunnels could be somewhat smaller.

It is assumed that the tunnel entrance at the surface and the cavern storage are designed and constructed in a manner that excludes the possibilities of flooding of the storage.

The loading/unloading of waste packages takes place in a section of the main rock hall, most of which serves as storage area. Transfer of waste packages between the loading/unloading section and the storage area is carried out by an overhead crane, and a hatch in the roof above the receiving section provides the necessary communication to the storage. The hatch is normally covered by a concrete plug for shielding.

The Storage Hall is mined with a width of 24 m and a total height of about 30 m and with a rock cover of at least 30 m (somewhat depending on the quality of the rocks). The initial lenght is chosen as 100 m, the storage capacity can be increased as required by expanding the lenght of

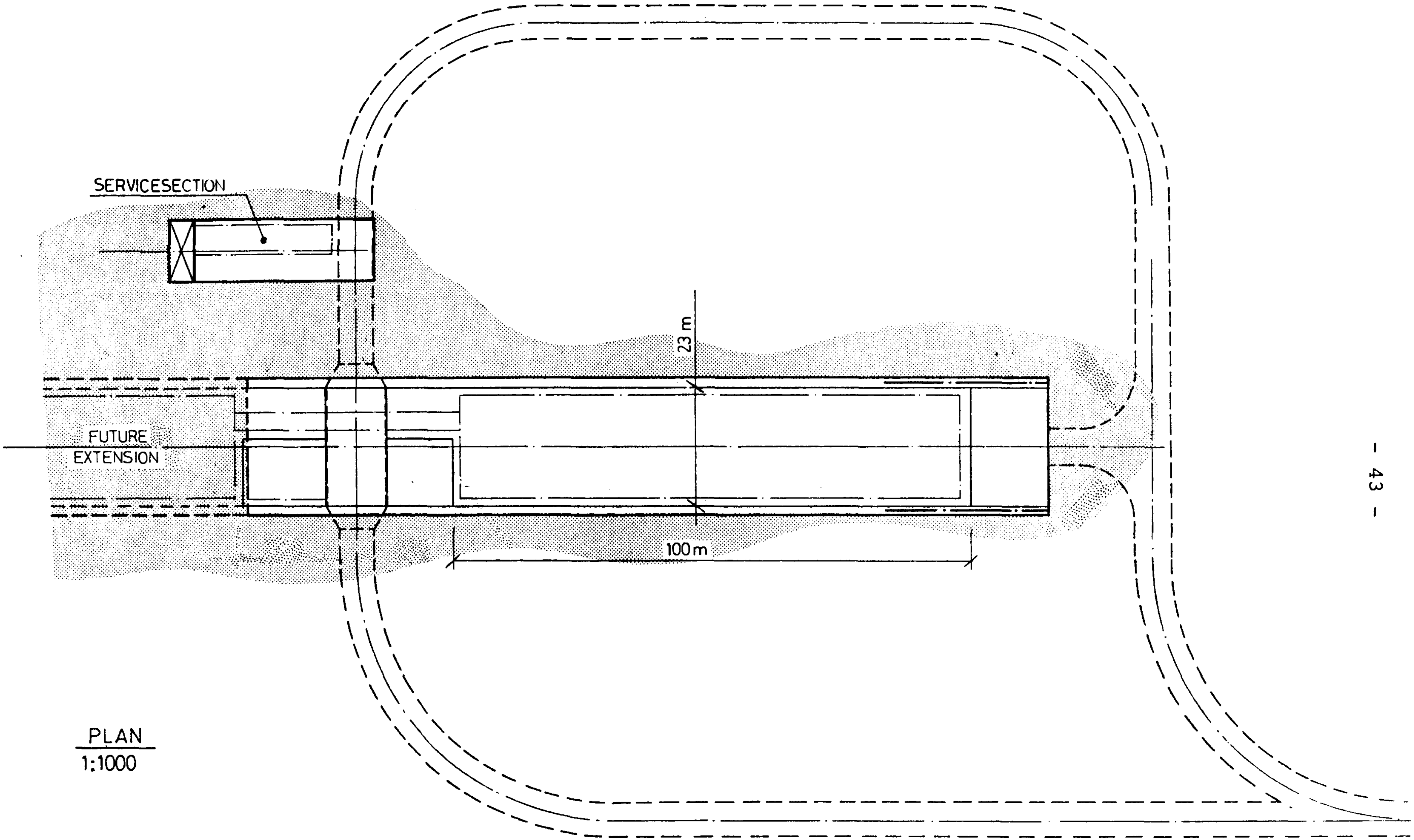


Figure 4.3 Rock cavern storage facility - General layout.
Based on /27/.

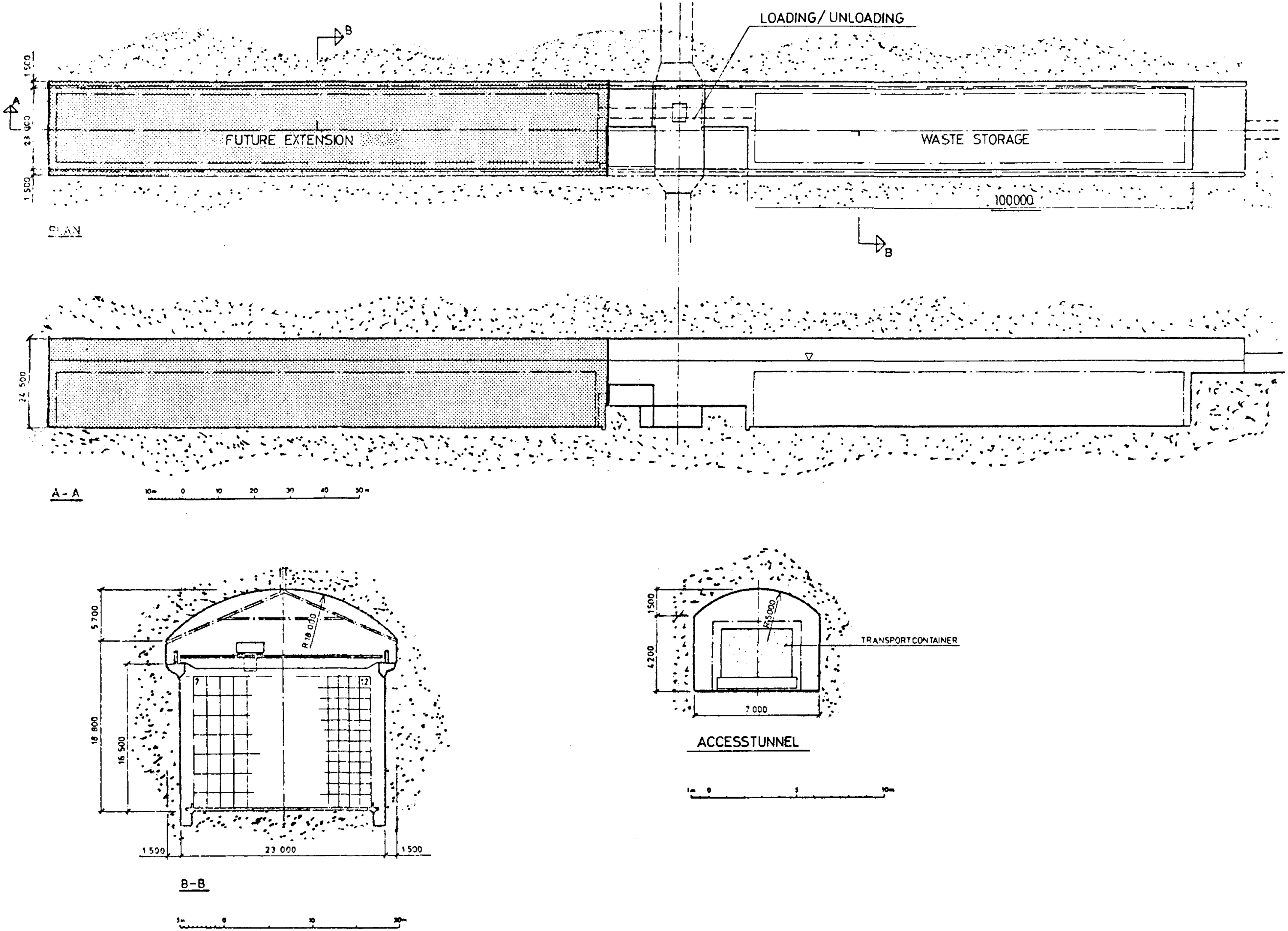


Figure 4.4 Rock cavern storage - Storage area. Based on /27/.

the cavern and/or by mining a second cavern parallel to the first one.

The surrounding rock is assumed to be of good quality, but the presence of some cracks is taken into consideration in the safety analysis. Reinforcements, bolting and injections shall be carried out as required.

Handling of the waste is done by means of an overhead crane. The maximum lifting height is about 14 m. Control and manoeuvering is by remote control, utilizing closed circuit TV-equipment. During normal operation, no personnel will be allowed to enter the storage area.

The waste packages are stacked as high as possible. supported and stabilized as necessary. Drums with waste incorporated in bitumen are stored in the concrete containers (see figure 5.1) in order to reduce the fire hazard and to facilitate sufficiently high stacks without damaging the drums.

The Service Section is located in a smaller mined rock cavern in close connection with the access tunnel, see figure 4.3. The service section houses the central equipment for the electrical power supply, ventilation, drainage and water treatment, and access to lift and stairways.

5 TRANSPORT

5.1 GENERAL

After some years of temporary storage at the power plant, the waste will normally be transferred from the storage to a respository for disposal. The transfer could be by truck, by train or by boat. This study has concentrated on the following two modes of transportation:

- Road transport by means of truck (semi trailer)
- Sea transport by means of a specially designed vessel (roll on - roll off)

Both modes will utilize transport containers. For those waste packages particularly investigated in this study, transport containers with shielding capability will be necessary, but the majority of the waste packages of other categories contain activities so low that they can be transported in unshielded containers made from relatively thin steel plates, corresponding to IAEA's requirements for "strong industrial package".

Because of load limits on the roads, smaller containers are envisaged for truck transports than for boat transports, however, no attempt has been made to optimize the transport system, for instance no "standard" container suitable for all three types of waste packages was developed within this study.

The two modes of transportation are described in the following subsections.

5.2 ROAD TRANSPORT

Road transport will utilize heavy trucks, semi-trailers seem to be most suitable. The vehicle will typically have a net capacity of about 20 - 25 tons* of which 1 - 3 tons must be reserved for shielding the driver.

In this study it is assumed that transportation will proceed with normal speed and that no particular restrictions are applied to transportation through towns and heavily populated areas.

The waste packages will be transported directly from "door to door" in containers. The design of the container will vary with the type of waste package and with the nuclide content. Most of the packages contain waste of very low activity and can be transported in containers made of steel plate.

The reference waste, in particular the RWCS-resin, normally contains waste of so high activity that shielding is necessary.

*NOTE: Maximum permissible over-the-road weight depends on the distance between the first and the last wheel-axle of the transporter:

	Axel distance (m)			
	5	10	16	22
Norway	17 t	33 t	42 t	--
Denmark	19 t	24 t	32 t	--
Finland	19 t	28 t	39 t	--
Sweden	16-22 t	22-31 t	30-44 t	37-51 t
International Road Traffic Commission	19 t	25 t	32 t	

Drums with waste solidified in bitumen will be packed in containers made of reinforced concrete with 125 mm thick walls, each containers housing 8 drums, see figure 5.1. These containers provide radiation shielding /32/ and at the same time protect the drums against fire. Drums of sufficiently low activity are transported in these containers without further packaging.

Containers with drums of higher activity will be placed in another container of reinforced concrete with 300 mm thick walls (alternatively 90 mm steel) which will be sufficient for the majority of the waste drums. The few drums that require still more shielding could either be stored for an additional period for further decay, or special measures must be taken for transportation.

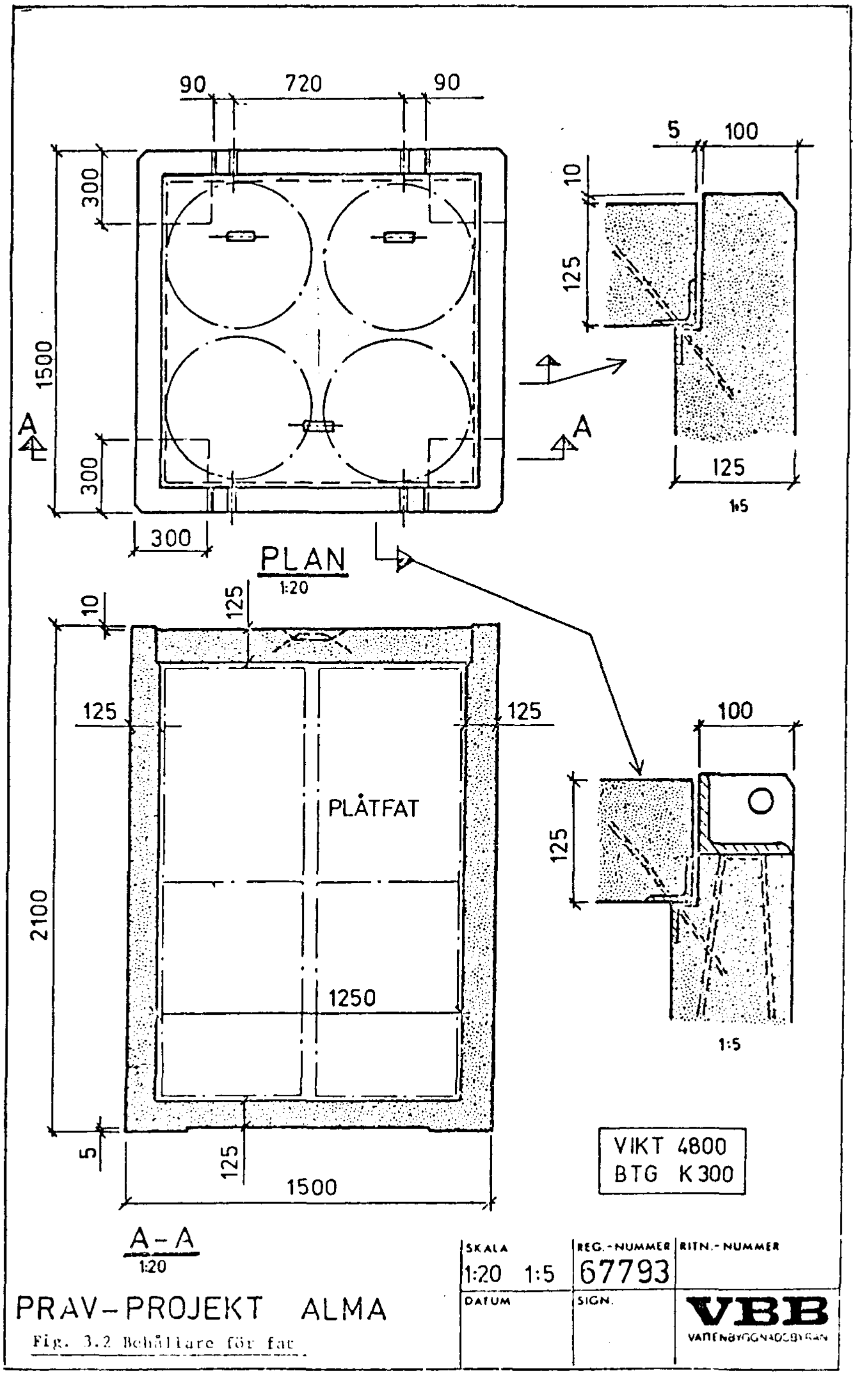


Figure 5.1 Concrete container for 8 drums /29/.

<u>Drums with waste incorporated in cement</u> could be packed and transported in container systems as assumed for bituminized waste. For drums of low activity it might, however, be more practical to use a larger container, for instance made of steel plate and having a capacity of 24 drums.

Concrete moulds of low activity will be transported in containers made from steel plate. Each container will take two concrete moulds. Moulds with higher activity content will be transported in containers made of reinforced concrete, with 300 mm thick walls (alternatively 90 mm steel plates), housing 2 moulds. The few moulds with a nuclide content too high to permit transportation in this type of container, could either be stored for an additional period for further decay, or special measures must be taken for transportation.

A summary of the containers envisaged for road transport is given in table 5.1.

Table 5.1 Different types of transport containers.

		Drums with bitumenized waste	Drums with cementized waste		Concrete moulds with cementeized waste
Type 1	Walls	125 mm conc.*	125 mm conc.	8 mm steel	8 mm steel
	Capacity	8 drums	8 drums	24 drums	2 moulds
	Tot.weight	≃ 6,5 t	≃ 8,5 t	≃ 21 t	≃ 10 t
Type 2	Shielding	300 mm conc. ** (or 90 mm steel)**	300 mm conc.** (or 90 mm steel)**		300 mm conc. (or 90 mm steel)
	Capacity	8 drums	8 drums		2 moulds
	Tot. weight	≃ 21 t	≃ 23 t		≃ 22 t

* The 8-drum concrete container, from figure 5.1

** In addition to the 8-drum container. See note p. 47.

Based on the use of Type 2 transport containers, the total transport volume for the reference wastes (RWCS + SFPCS resins), assuming a distance of 200 km from each reactor to the repository, is shown in table 5.2.

Table 5.2. Total transport volume for reference waste.

Waste incorporated in - and packed in	bitumen drums	cement drums	cement moulds
Packages per year per reactor	42.5	80	16
Packages from 6 reactors in 30 years	7,650	14,400	2,880
Number of transport	960	1,800	1,440
Total transport distance,(km)	192,000	360,000	288,000

5.3 SEA TRANSPORT

The Nordic study has not performed any analysis of sea transportation of its own, but merely adapted the system design recently developed within the Swedish ALMA-project. This project has developed a concept for a roll on - roll off vessel for transport of radioactive waste and spent fuel. The vessel design is shown in figure 5.2, while a cross section of the vessel loaded with waste containers is shown in figure 5.3. The maximum payload of the vessel is about 1 100 ton /29/, /30/.

The containers envisaged for sea transportation are considerably larger than those envisaged for transportation by truck. Only one container size will be used, but two versions have been developed: an unshielded and a shielded container.

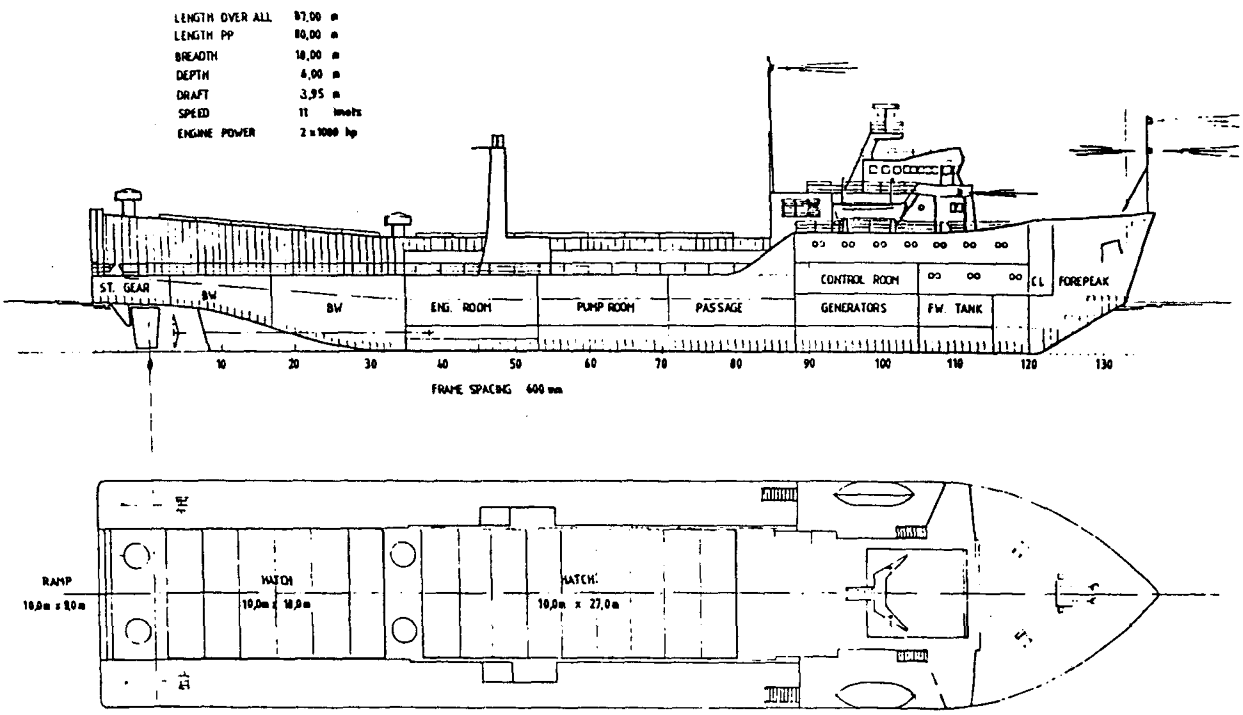


Figure 5.2 The ALMA-Vessel /30/.

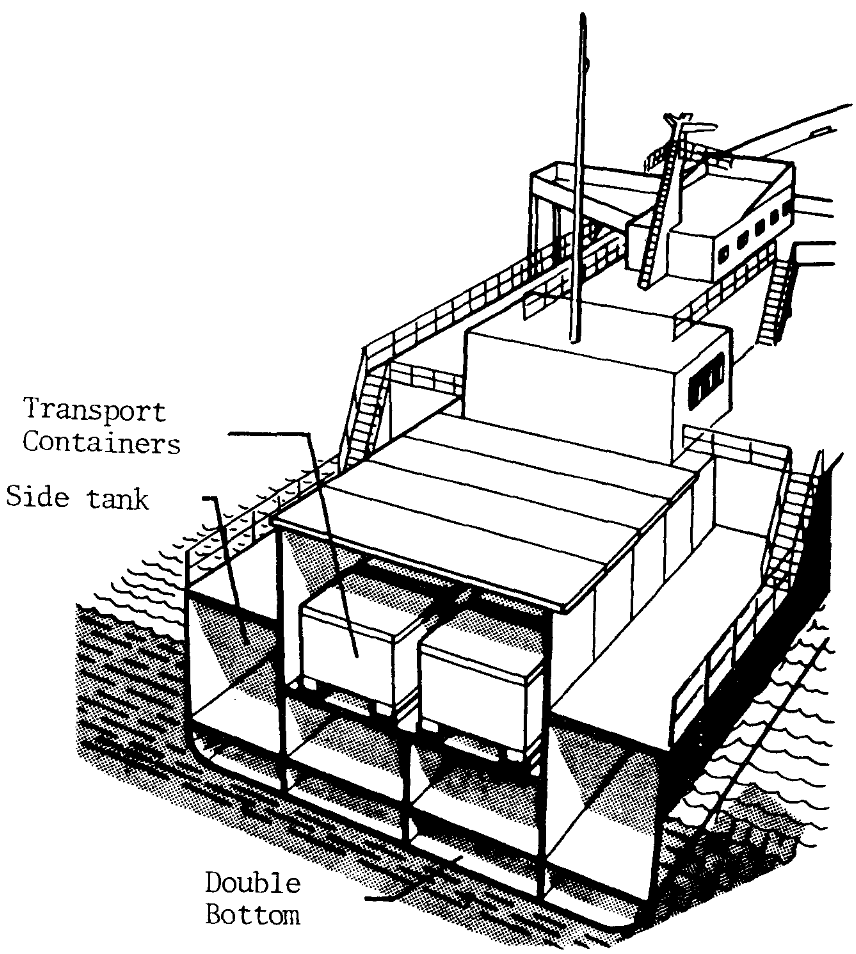


Figure 5.3 Cross section of the ALMA-Vessel /31/.

The shielded version is shown in figure 5.4. The weight of the shielded container is about 40 tons, loaded with concrete moulds, the total weight will be about 90 tons.

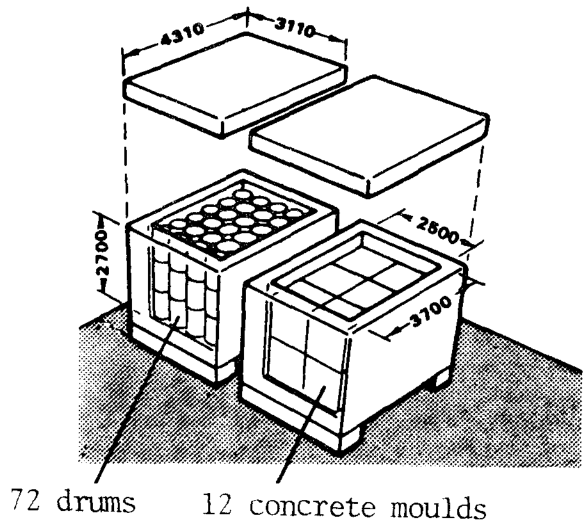


Figure 5.4 Shielded transport containers for drums or concrete moulds with radioactive waste. Unshielded version has the same inner dimensions /31/.

In the terminals the very heavy transport containers are handled by means of hydraulic lifting trucks of the type shown in figure 5.5

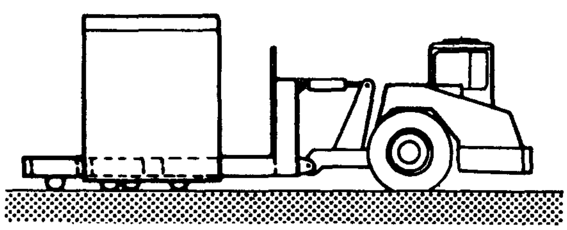


Figure 5.5 Hydraulic lifting truck for transport containers /31/.

6. DISPOSAL

6.1 GENERAL

As examples of disposal facilities, the following three alternatives have been chosen:

- Shallow land burial
- Concrete bunker
- Rock cavern

The shallow land burial and the concrete bunker are both situated near the surface with an earth cover of about 2 m, while the rock cavern is situated at a moderate depth with a rock cover of about 30 m.

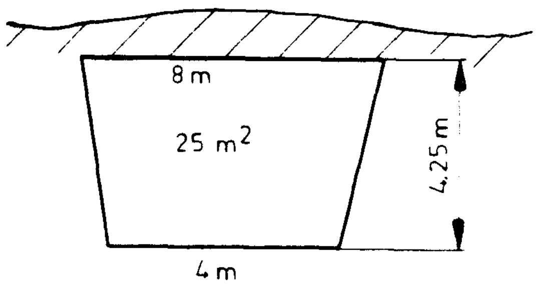
For these alternatives, analyses have been performed using general data that are supposed to be realistic for nordic conditions.

In order to facilitate a comparison of the different alternatives, the site data are, as far as possible, assumed to be identical for all three alternatives and are accounted for in chapter 3.

The types of waste packages and the waste volumes discussed in the following refer to the reference waste specified in chapter 2.

6.2 SHALLOW LAND BURIAL

The waste packages received at the burial site are placed directly in trenches excavated in the soil at the site. The dimensions of the trench are given in figure 6.1. together with the total volyme and surface of the waste packages. In this case the leakage from the repository is released to the total surfaces of all waste packages.



	Waste packages		
	Length, m	Volume, m^3	Surface, m^2
Drums, bituminized waste	100	1,700	16,000
Drums, cementized waste	185	3,500	30,000
Concrete moulds	240	4,900	25,000

Figure 6.1 Shallow land burial - Size of trench.

For the reference study, the spaces between the waste packages are assumed to be back-filled and covered with the excavated masses. The covering soil will be about 2 m thick.

The disposal area is surrounded by a fence that gives a distance between the trenches and areas of unrestricted use of at least 100 m.

6.3 CONCRETE BUNKER

The repository consists of reinforced concrete structures located in the same types of soils as the shallow land burial.

The outer concrete walls are assumed to have a thickness of 0.5 m.

Each bunker is divided into smaller cells (4 m x 4 m) in which the waste drums or moulds are stacked, see figure 6.2.

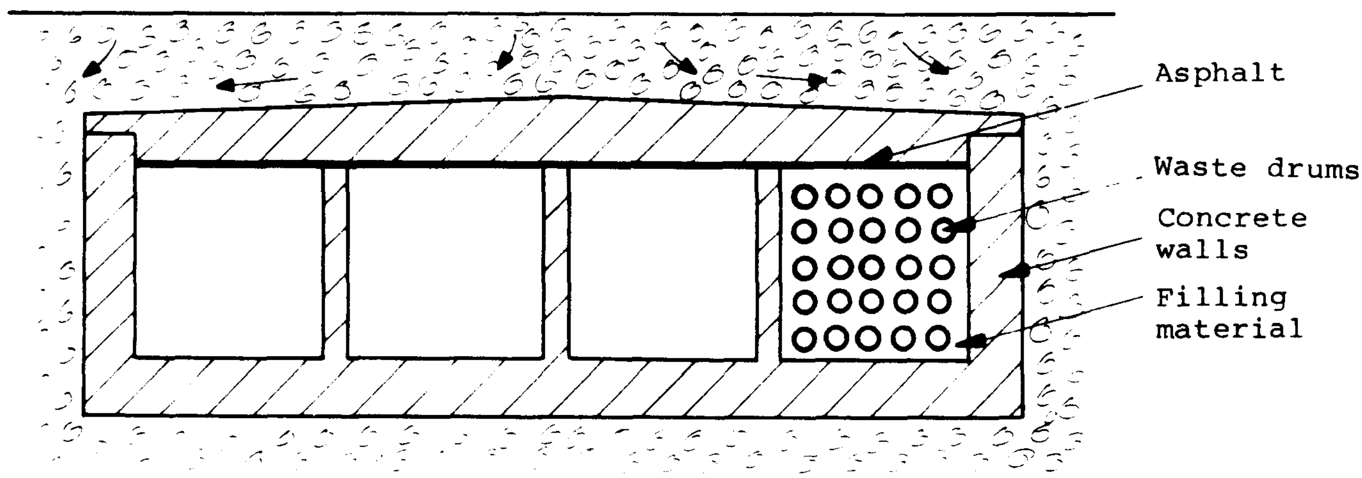
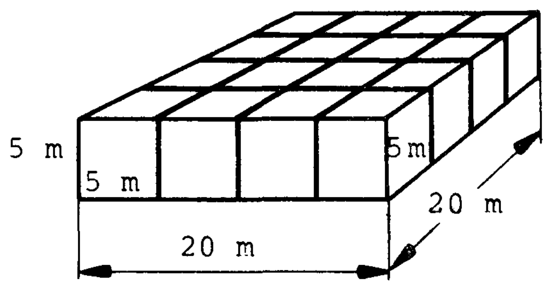


Figure 6.2 Concrete bunker.

The spaces between the waste packages are filled with a material such as clay, concrete etc. (In the calculation the filling material is supposed to be the same material as the matrix in the waste packages, bitumen and concrete respectively). When completely filled, the cells are sealed with a layer of asphalt and a concrete lid is poured in place. Finally, the whole repository is covered with a 2 m layer of soil.

The total number of bunker units for the different waste types are given in figure 6.3.



	Number of bunkers	Volume, m^3	Surface, m^2
Drums, bituminized waste	3.2	6 400	3 800
Drums, cementized waste	6	12 000	7 200
Concrete moulds	7	14 000	8 400

Figure 6.3   Number and size of concrete bunkers.

6.4 ROCK CAVERN

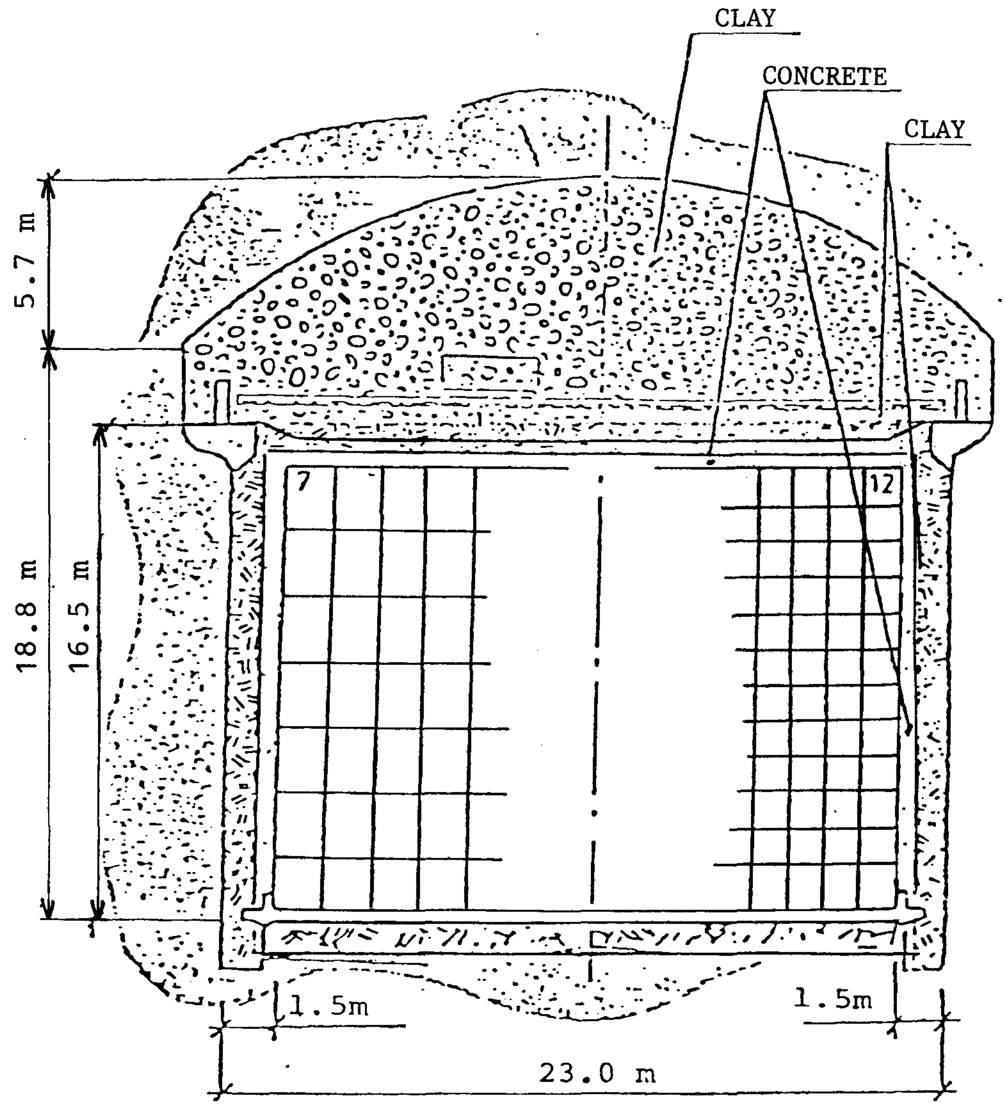
For this study, a Swedish concept /33/ has been used as a reference plant. The same concept has also been used as a reference design for a rock covern storage and its general layout is described in section 4.4.

For the isolation of the waste in the disposal facility, there is a concrete wall 0.5 m thick between the waste piles and the rock walls, leaving a space of 1 m, that can be filled with bentonite (see figure 6.4). The waste pile has a width of 20 m and a height of 15 m.

The waste packages are stacked with a space of approximately 10 cm between the packages. At convenient intervals, these spaces are filled with concrete, stabilizing the stocks and giving a sealing between the packages.

When the storage facility is completely filled up, or operation is terminated, the facility is sealed by filling, as far as possible, all empty spaces with clay. Finally the transport tunnel is closed.

This size of the repository is given in figure 6.4. As for the other alternatives, only the reference type waste is considered.



	Length, m	Volume, m^3	Surface, m^2
Drums, bituminized waste	10	3 000	1 300
Drums, cementized waste	19	5 700	1 800
Concrete mould	23	6 900	2 200

Figure 6.4 Rock cavern - cross section and size of repository. Figure from /27/.

REFERENCES

1. Mullarkey T.B., et al., A survey and evaluation of handling and disposing of solid low-level nuclear fuel cycle wastes. AIF/NESP-008, 1976, 143 s.

2. Annual summaries of waste amounts and nuclide contents in waste packages from Oskarshamn power plant. Oskarshamn, 1980, unpublished.

3. Annual summaries of waste amounts and nuclide contents in untreated wastes from Barsebäck power plant. Barsebäck, 1980, unpublished.

4. Alternatives for managing wastes from reactors and postfission operations in the LWR fuel cycle. ERDA-76-43, Vol. 1 of 5. 1976.

5. Clark, W., The use of ion exchange to treat radioactive liquids in light water cooled reactor power plants. NUREG/CR-0143, ORNL/NUREG/TM-204, 1978.

6. Ruokola, E., Kätkä, M.: Statistical analysis of waste activity data from Oskarshamn and Barsebäck. NKA/AO(80)35.

7. Nuclide contents in the fuel to TVO power plant computed by the ORIGEN code. YDI, Helsinki, 1979 umpublished.

8. Measurement data of nuclide contents in primary water and liquid effluents of Oskarshamn power plant. Oskarshamn, 1980, unpublished.

9. Final environmental statement concerning proposed rulemaking action: Numerical quides for design objectives and limiting conditions for operation to meet the criterion "as low as practicable for radioactive material in light-water-cooled nuclear power reactor effluents. WASH-1258, US Atomic Energy Comission. July 1973.

10. Schwibach, J., Riedel, H. and Bretschneider, J., Investigation into the emission of carbon-14 compounds from nuclear facilities, its measurement and the radiation exposure resulting from the emission, Commission of the European Communities, Nov. 1978.

11. Salonen, L., Snellman, M., Carbon-14 determination from ion-exchange resins. NKA/AO(80)42.

12. Aittola, J-P., Olsson, A-L., Determination of C-14 in ion-exchange resins. (In Swedish). NKA/AO(81)3. 1980.

13. Rämö, E., Containers used in packaging reactor waste. Presented in the symposium on the on-site management of reactor wastes, Zürich, Switzerland, March 26-30, 1979. Technical Research Centre of Finland.

14. Ruokola, E., Calculations made by the CYLDOS and RECDOS radiation shielding codes. Espoo, VTT/Rea, 1979, unpublished.

15. Kjellbert, N., et al., Central repository for low- and intermediate - level waste (ALMA). Conceptual design, siting and safety study. Report PRAV 1.29. 1980.

16. Eriksson, A-M., et al., An inventory of possible types of accidents and frequencies during transportation. (In Swedish). PRAV 1.16. 1978.

17. Hahkala, M., Mehtonen, A., Aittola, J-P., Incineration of ion-exchange resin/bitumen mixture - particle size distribution in exhaust gas. (In Swedish). NKA/AO(79)21. 1979.

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

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

20. Rämö, E., Pulkkinen, U., Aittola, J-P., Damage of reactor waste packages in transport accidents. NKA/AO(79)17. 1979.

21. Pihlajavaara, S.E., Aittola, J-P., Radwaste concrete: solidification of nuclear wastes with Portland cement. NKA/AO(79)3. 1979.

22. Aittola, J-P., Kleveland, O. Unpublished.

23. Ollila, K., Rämö, E., Leach testing of simulated concrete products. NKA/AO(80)19.

24. Peltonen, E., Heinonen, J., Kuusi, J., Incorporation of radioactive wastes from nuclear power plants into concrete and bitumen. IAEA-SM-207/5, Management of radioactive wastes from the nuclear fuel cycle, Vol. II, 1976.

25. Tveten, U., Thomassen, D., Kvaal, E.: Accident Analysis performed for the Norwegian Government Committee on Nuclear Power, Institute for Energy Technology, Kjeller, Norway, Report KP-156, February 1979.

26. Sivertsen, B.: meteorological data and dispersion - parameters used in the evalution of nuclear power and safety. (In Norwegian), NILU-report 21/78, 1978.

27. C.-O. Forsmark, Facility for low- and medium level waste (ALMA) - Rock cavern. (In Swedish), PRAV Report 1.13, Stockholm June 1978.

28. O. Degerman, N. Rydell and C. Thegerström, Central facility for low- and medium level waste, ALMA - A pre-project study. (In Swedish), PRAV Report 1.18, Stockholm July 1978.

29. Milchert, T., Transport of low- and medium level radioactive waste to the waste storage ALMA - A preliminary study of the transport system. (In Swedish), PRAV Report 1.15, June 1978.

30. Milchert, T., ALMA-transport container - mechanical influences in accidents and principal design. (In Swedish), PRAV-Report 1.21, August 1979.

31. Swedish National Council for Radioactive Waste. Report on activities from July 1977 to June 1979.

32. Olsson, G., Preliminary radiation shielding calculations for transport and storage. (In Swedish), PRAV Report 1.11, May 1978.

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

APPENDIX A

Summary of data assumed for REFERENCE SYSTEM

SUMMARY OF DATA
assumed for the
REFERENCE SYSTEM

1. POWER SYSTEM

Reactor type	:	BWR
Reactor output	:	500 MWe
Number of reactors	:	6
Operating period	:	30 years (simultaneuosly)

2. WASTE TYPES

Granular ionexchange resin from Reactor Water Cleanup System : RWCS-resin

Powered ionexchange resin from Spent Fuel Pool Cleanup System : SFPCS-resin

Note: Other waste types are not considered.

3. AMOUNT OF WASTE

Assumed amount of waste prior to solidification

Waste type	Decanted volume m^3/a/reactor	Dry resin weight kg/a/reactor
RWCS-resin	12	2,400
SFPCS-resin	10	1,000

4. SOLIDIFICATION MATRICES

Bitumen

Cement

5. WASTE PACKAGES

5.1 200 l drum

- dimensions : 0.57 m dia x 0,875 m high
- total volume : 0.224 m^3

used for waste incorporated in bitumen or cement

5.2 1,000 l concrete mould

- dimensions : 1.2 x 1.2 x 1.2 m
- volume, inner: 1.0 m^3
  total: 1.7 m^3

used for waste incorporated in cement

6. ANNUAL PRODUCTION OF PACKAGES PER REACTOR

	Package type	Waste type Resin from	Dry resin per package kg	Number of packages n/a/reactor
either:	Bituminized waste in drum	RWCS	80	30
		SFPCS	80	12.5
or:	Cementized waste in drum	RWCS	40	60
		SFPCS	50	20
or:	Cementized waste in conc. mould	RWCS	200	12
		SFPCS	250	4

7. NUCLIDE CONTENT IN THE WASTE

7.1 Agerage annual nuclide content in fresh waste from one reactor. (Values used for waste in storage).

Nuclide	Decay constant (a^{-1})	RWCS resins		SFPCS resins	
		GBq/a/BWR	Ci/a/BWR	GBq/a/BWR	Ci/a/BWR
C-14	$1.10x10^{-4}$	3	0.1	0.1	0.003
Ni-59	$8.66x10^{-6}$	1.5	0.05	0.1	0.003
Ni-63	$5.55x10^{-3}$	300	10	20	0.5
Co-60	$1.31x10^{-1}$	3 000	100	200	5
Sr-90	$2.39x10^{-2}$	50	1.5	5	1.5
Tc-99	$3.25x10^{-6}$	0.05	0.001	0.03	0.001
I-129	$4.36x10^{-8}$	0.002	0.00005	0.001	0.00003
Cs-134	$3.30x10^{-1}$	1 000	30	800	2
Cs-135	$2.31x10^{-7}$	0.003	0.0001	0.002	0.00005
Cs-137	$2.31x10^{-2}$	1 000	30	500	15
Pu-239	$2.89x10^{-5}$	0.001	0.00003	0.0005	0.00001

7.2 Average nuclide content (RWCS-resin only) in various packages. Fresh waste (Values used for waste in storage).

Package type / Nuclide	Drum of bitumenized waste MBq/package	Drum of cemented waste MBq/package	Concrete mould of cemented waste MBq/package
C-14	100	50	250
Ni-59	50	25	125
Ni-63	10 000	5 000	25 000
Co-60	100 000	50 000	250 000
Sr-90	1 700	800	4 200
Tc-99	1.7	0.8	4
I-129	0.07	0.03	0.17
Cs-134	33 000	17 000	80 000
Cs-135	0.1	0.05	0.25
Cs-137	33 000	17 000	80 000
Pu-239	0.033	0.017	0.08

7.3 Average nuclide content in various packages. Age of waste about 5 years. (Values used for waste being transported).

Package type / Nuclide	Drum of bitumenized waste (MBq/package)		Drum of cemented waste (MBq/package)		Mould of cemented waste (MBq/package)	
	RWCS resin	SFPCS resin	RWCS resin	SFPCS resin	RWCS resin	SFPCS resin
C-14	100	8	50	5	250	25
Ni-59	50	8	25	5	130	25
Ni-63	10 000	1 600	5 200	1 000	26 000	5 200
Co-60	52 000	8 300	26 000	5 200	130 000	26 000
Sr-90	1 500	360	750	220	3 800	1 100
Tc-99	1.7	2.4	0.85	1.5	4.2	7.5
I-129	0.07	0.08	0.03	0.05	0.17	0.25
Cs-134	6 600	14 000	3 300	8 600	17 000	43 000
Cs-135	0.1	0.16	0.05	0.1	0.25	0.5
Cs-137	30 000	36 000	15 000	22 000	75 000	110 000
Pu-239	0.03	0.04	0.02	0.08	0.08	0.13

7.4 Radionuclides accumulated in waste from 6 BRW's 5 years after termination of the 30 year operating period. (Values used for disposed waste).

Nuclide	RWCS resins		SFPCS resins	
	GBq	Ci	GBq	Ci
C-14	540	15	18	0.50
Ni-59	270	7	18	0.50
Ni-63	52 000	1 400	3 300	90
Co-60	70 000	1 900	4 500	120
Sr-90	5 800	160	580	16
Tc-99	9.0	0.25	5.4	0.15
I-129	0.36	0.010	0.18	0.005
Cs-134	4 000	110	3 300	90
Cs-135	0.54	0.015	0.36	0.01
Cs-137	120 000	3 200	58 000	1 600
Pu-239	0.18	0.005	0.09	0.002

8. PROPERTIES OF WASTE PRODUCTS

8.1 Combustion Properties of Bitumen/Resin Mixture

Energy release	:	~	3 MJ/kg
or	:	~	6 GJ per drum
Fire temperature	:		800 - 1000°C

Release by fire:

- 100% of cesium released as gas
- 60% of other nuclides (strontium and cobalt) released as gas as particles smaller than 10 μm
- 25 % of other nuclides (strontium and cobalt) released as particles larger than 10 μm
- 15 % of other nuclides (strontium, cobalt) remain in ashes

(used for bitumen fire in storage and under transportation).

8.2 Mechanical properties of cementized waste products

Compressive strenght:

- 10 W % resin in product:	$\geqq$ 30 MPa
- 15 W % resin in product:	$\geqq$ 20 MPa
- 20 W % resin in product:	$\geqq$ 10 MPa
- concrete mould	> 35 MPa

Impact characteristics

- 0.1 W % released as particles < 10 μm

8.3 Water resistance and swelling

Water resistance: better than 100 days

Swelling in water:

- concrete products: max. 0.3 % (volume)
- bitumen products : max. 5 % (volume)

8.4 Leach rates

Leach rates, cm per day after 100 days

	Cesium	Strontium	Cobalt
Cementized waste	10^{-3}	10^{-4}	10^{-5}
Bituminized waste	10^{-4}	10^{-4}	10^{-5}

Values used for Fall-in-water Accident).

9. STORAGE

Storage time: 5 years

Types of storage facilities:

- storage building for bituminized waste
- storage building for cementized waste
- rock cavern storage for both types of waste

Maximum lifting height

- storage building for bituminized waste ≃ 7 m
- storage building for cementized waste ≃ 6 m
- rock cavern storage ≃ 14 m

10 TRANSPORTATION

(RWCS- and SFPCS-resin from 6 reactors operated for 30 years).

10.1 Total transport volume:

either : 7,650 drums of bituminized waste
or : 14,400 drums of cementized waste
or : 2,880 concrete moulds of cem. waste

10.2 Truck transport

Distance between reactor and repository: 200 km
Truck load: 25 - 30 ton
Number of packages per truck load:*
- bit. waste drums: 8
- cem. waste drums: 8
- moulds : 2

Number of transports:
- bit. waste drums: 960
- cem. waste drums: 1,800
- moulds : 1,440

Total transport distance (excl. empty return):
- bit. waste drums: 288,000 km
- cem. waste drums: 360,000 km
- moulds : 192,000 km

----

* Based on the assumptions that all waste packages are transported in shielded transport containers. Actually, several of the packages (in particular those containing SFPCS-resin) can be transported with less shielding permitting more units in one truck load. Only a very few units may require use of additional shielding during tranportation.

10.3 Sea transport

Distance between reactor and repository: 200 km
Pay-load of boat ≃ 1,100 ton

Number of packages per container:

- bit. waste drums: 45
- cem. waste drums: 72
- moulds : 12

Total number of containers with:

- bit. waste drums: 170
- cem. waste drums: 200
- moulds : 240

Number of containers per boat-load:

- bit. waste drums: 17
- cem. waste drums: 12
- mould : 12

Number of boat-loads:

- bit. waste drums: 10
- cem. waste drums: 17
- moulds : 20

11. DISPOSAL

11.1 Types of disposal facilities:

- shallow land burial
- concrete bunker
- rock cavern

11.2 Diffusion coefficients, m^2 per year

Nuclide	Concrete	Bitumen	Clay
Carbon	$3 \; 10^{-4}$	$2 \; 10^{-6}$	$3 \; 10^{-2}$
Technetium	$6 \; 10^{-4}$	$2 \; 10^{-6}$	$6 \; 10^{-2}$
Iodine	$6 \; 10^{-4}$	$2 \; 10^{-6}$	$6 \; 10^{-2}$
Cesium	$3 \; 10^{-4}$	$1 \; 10^{-6}$	$2 \; 10^{-4}$
Strontium	$7 \; 10^{-6}$		$7 \; 10^{-5}$
Cobalt	$4 \; 10^{-9}$	$4 \; 10^{-9}$	$8 \; 10^{-6}$
Nickel	$4 \; 10^{-9}$	$4 \; 10^{-9}$	

(Used for calculation of release from repository).

11.3 Thickness of man-made barriers (m)

	Concrete wall	Clay barrier
Shallow land burial	0	0
Concrete bunker	0.5	0
Rock cavern	0.5	1

11.4 Number of packages in repository and related volumes

		Bit.waste drums	Cem.waste drums	Conc. moulds
Number of packages	RWCS-resin	5,400	10,800	2,160
	SFPCS-resin	2,250	3,600	720
	Total	7,650	14,400	2,880
Geometrical volume		1,900	3,600	7,000
Shallow land burial:				
Stacked volume (m^3)		2,500(+30%)	4,700(+30%)	8,400(+20%)
Waste matrix volume (m^3)		1,530	2,880	2,880
Concrete bunker:				
Stacked volume (m^3)		6,400(+240%)	12,000(+230%)	14,000(+100%)
Waste matrix volume (m^3)		1,530	2,880	2,880
Rock cavern:				
Stacked volume (m^3)		3,100(+60%)	5,800(+60%)	8,400(+20%)
Waste matrix volume (m^3)		1,530	2,880	2,880

12. DISPOSAL SITE DATA

12.1 Miscellaneous data for soil and hydrology

	Sandy till	Clayey till	Rock
Hydraulic conductivity, K_p (m/s)	$1 \cdot 10^{-7}$	$1 \cdot 10^{-9}$	$1 \cdot 10^{-8}$
Hydraulic gradient, i (m/m)	$5 \cdot 10^{-2}$	$5 \cdot 10^{-2}$	$2 \cdot 10^{-2}$
Kinematic porosity, E_k	$3 \cdot 10^{-2}$	$1 \cdot 10^{-2}$	$3 \cdot 10^{-3}$
Diffusion porosity, E_d	0.4	0.2	-
Dry particle soil density, ρ_{tp} (kg/m^3)	2,650	2,650	-
Geometric surface area, a_2 (m^2/kg)	-	-	30
Fracture spacing, s (m)	-	-	0.4
Distribution coefficient, K_d (m^3/kg) for:			
- carbon	0	0	0
- technetium	0	0	0
- iodine	0	0	0
- cesium	0.1	0.5	0.1
- strontium	0.02	0.1	0.01
- cobalt	0.1	1	0.1
- nickel	0.1	1	0.1
Retention factor*, K_i for:			
- carbon	1	1	1
- technetium	1	1	1
- iodine	1	1	1
- cesium	400	5,300	420
- strontium	80	1,060	42
- cobalt	400	10,600	420
- nickel	400	10,600	420

* Calculated from the K_d-values

12.2 Data for well

Distance from repository: 100 m (downstream)
Drinking water capasity : 300 m^3 per year
Individual intake : 0.44 m^3 per year

12.3 Site area

Farmland	: 10% of surroundings land
Individuals' consumption of vegetables:	28 kg per year

Uptake in vegetables (factor vegetable/soil):

- carbon : 5.5
- nickel : 0.01
- strontium: 0.08
- cesium : 0.005

13. SAFETY ANALYSIS

The safety analysis covers the following areas:

13.1 Storage

- Normal events
  - Handling
  - Environment
  - Corrosion
  - Internal process
- Abnormal events
  - Drop and collision
  - Fire in bituminized waste
  - Fire around cementized waste

13.2 Transport

- Normal events
  - Handling
  - Environment
- Abnormal events
  - Road transport:
    Collision (mech. damage)
    Fall-in-water
    Fire in bitumen
- Sea transport:
  Loss of single package
  Fire in bitumen

13.3 Disposal

Institutional control period 0 - 100 years

A. Migration to well, lake or sea
  - repository intact
  - repository broken

After institutional control period > 100 year

A. Migration to well close to repository
   - respository intact
   - respository broken

B. Intrusion
   - dwelling
   - excavation
   - blasting

C. Farming
   - soil contamination
   - uptake in plants

NORDIC STUDY ON REACTOR WASTE

TECHNICAL PART II

SAFETY ANALYSIS

LIST OF CONTENTS

		Page
1	INTRODUCTION	1
2	SAFETY ANALYSIS IN THE PRESENT STUDY	2
2.1	Spesification of the system	2
2.2	Definition of scenarios	3
2.3	Calculation of the consequences	5
2.4	Evaluation of probabilities	9
2.5	Important factors and parameter value variations	9
3	STORAGE, DESCRIPTION OF CALCULATIONS	10
3.1	General	10
3.2	Potential Hazards	10
3.2.1	Normal events	10
3.2.2	Abnormal events	13
3.3	Drop and collision	14
3.4	Bitumen fire	14
3.4.1	Basic accident assumptions	14
3.4.2	Results of calculations	16
3.4.3	Probability of bitumen fire	22
3.4.4	Factors to be considered	23
3.5	Fire in a storage for cementized waste	24

Page

4.	TRANSPORTATION, DESCRIPTION OF CALCULATIONS	25
4.1	General	25
4.2	Potential hazards	25
4.2.1	Normal events	25
4.2.2	Abnormal events	27
4.3	Truck collision	28
4.3.1	Basic accident assumptions	28
4.3.2	Estimated consequences	29
4.3.3	Probability of accident	29
4.4	Fall-in-water accident	31
4.4.1	Basic accident assumptions	31
4.4.2	Results of calculations	33
4.4.3	Probability of accident	38
4.5	Bitumen fire	38
4.5.1	Basic accident assumptions	38
4.5.2	Results of calculations	40
4.5.3	Probability of bitumen fire	40
4.6	Ship wreckage	46
4.6.1	The ALMA-study	46
4.6.2	Basic accident assumptions	46
4.6.3	Results of calculations	47
4.6.4	Probability of accident	50
4.7	Bitumen fire aboard ship	52
5	DISPOSAL, DESCRIPTION OF CALCULATIONS	53
5.1	General description of calculations performed	53
5.2	Administration and release scenarios	54
5.2.1	Period of operation	55
5.2.2	Period of supervision	55
5.2.3	Period of institutional land use control	56
5.2.4	Period of unrestricted use	57
5.3	Calculations and results	58
5.3.1	Water from well near intact repository	61
5.3.2	Water from well near fractured repository	70
5.3.3	Dwelling on top of the repository	71
5.3.4	Water from well directly outside intact repository	72

		Page
5.3.5	Water from well directly outside fractured repository	73
5.3.6	Intrusion by excavation	74
5.3.7	Farming	78
6	DESCRIPTION OF MODELS	81
6.1	Bitumen fire	81
6.2	Fall-in-water accident	81
6.3	Release rate from waste and repository	82
6.3.1	DIFMIG	83
6.3.2	GEOPATH	84
6.4	Nuclide migration in ground	85
6.5	Nuclide migration in till	86
6.6	Nuclide migration in rock	87
6.7	Water consumption from a well	87
6.8	Soil contamination	88
6.9	Uptake in vegetables	89
6.10	Ingestion of vegetables	89
6.11	Inhalation of contaminated dust	90
7	PARAMETER VARIATIONS	91
7.1	Nutrition parameter variations	91
7.2	Diffusion coefficient	95
7.3	Distribution coefficient	96
7.4	Nuclide composition	96
7.5	Population density	98
7.6	Response surface method	98
8	COMPARISON OF RISKS	102
8.1	Results from the present study	102
8.2	Background radiation	110
8.3	Comparison	113
References		116

TECHNICAL PART II

1 INTRODUCTION

An important part of the Nordic study on system- and safety analysis of the management of low and medium level radioactive waste from nuclear power plants is a safety analysis of a reference system. The reference system was established within the study and is desribed in Technical Part I, while the safety analysis is reported in this part.

2 SAFETY ANALYSIS IN THE PRESENT STUDY

Some general aspects of safety analysis are discussed in the Main Report, chapter 3. This chapter presents the actual safety analysis performed, though still from a general point of view.

The safety analysis has in principle been performed in the following steps:

- Specification of the system.
- Definition of the scenarios of the events to be analysed.
- Calculation of the consequences.
- Evaluation of probability for these consequences, if possible.
- Discussion of important factors and the influence of variation of the assumed values of these parameters.

For some parts of the safety analysis, available methods were relatively easily adapted. For other parts, however, it was found that it was necessary to perform a series of preparatory calculations in order to determine relevant scenarios, parameter values and methods.

2.1 SPECIFICATION OF THE SYSTEM

In Technical Part I the reference system is specified to the extent that has been needed for the safety analysis performed. In parts of the analysis it has, however, because of limitations set by the methods available, been necessary to simplify or modify parts of the system. This should be kept in mind when evaluating the results calculated.

2.2 DEFINITION OF SCENARIOS

Table 2.1 is a schematic overview of the various combinations of operation steps and events involved. In this table is also given a purely qualitative evaluation of exposure and risk levels to operators and the general public. The expression "exposure level" is used in connection with normal operation, since these exposures by definition should be expected with a probability equal to one. Abnormal events should by definition have a probability significantly lower than one, and accordingly exposure level as well as probability is of importance.

For disposal it may be difficult to maintain a distinction between normal and abnormal events. There is a continuous spectrum of events or effects that could potentially increase the exposure of the general public, and these are characterized by higher or lower probabilities rather than by readily being classified as normal or abnormal.

Occupational radiation doses are not analysed in the present study, either for normal or for abnormal events. Such an analysis must be based upon such detailed information about the system and routines, that it will only serve a purpose if performed for a specific plant.

For normal operation only the disposal plant will give public exposure, and this exposure will be delayed for many years, due to the migration time of nuclides. This situation has been quantitatively analysed, as indicated with the frame drawn in table 2.1.

The risk levels of abnormal events, or accidents, are seen as a combination of consequence and probability, and these have been estimated as shown in table 2.1. In this table also the cases analysed are indicated by a frame.

Table 2.1 Qualitative presentation of exposure and risk levels

	Storage	Transports action	Disposal operational	Disposal after closure	Disposal after end of period
Normal operation		Exposure			
Operators	some	some	some	none	none
General public	very low	low	very low	very low	delayed
Abnormal events		Risk level			
Operators	low	low	low	none	none
General public	low	low	low	very low	significant in pessimistic cases

2.3 CALCULATION OF THE CONSEQUENCES

The consequences are in this safety analysis evaluated as radiation doses. The doses can be translated into casualties, and these can be translated into moneraty losses, for comparison with risks from other activities in society, and for economic considerations. In this report, however, the consequenses are expressed as doses, in an attempt at avoiding introduction of further uncertainties.

The doses are expressed as "effective dose equivalent" to make it possible to compare the detrimental effect from different kinds of exposure to radioactive materials.

- "Dose equivalent" indicates that the absorbed dose has been multiplied with a quality factor for the actual type of radiation.

- "Effective dose" indicates that doses to different parts of the human body are expressed as the weighted mean whole body dose.

The effective dose equivalent or in short the dose is obtained by using factors in an ICRP report, reference /1/.

For an intake of radioactive substance by inhalation and ingestion the dose is calculated as committed dose. That means that the dose to the body from a single intake is integrated over a period of 50 years, and this integration is included in the dose conversion factors used in this study. These factors have been taken from reference /2/. They may also be derived from a later ICRP publication /3/.

The doses can be calculated as individual doses and collective doses.

For a continuous release, like that caused by leakage from a repository, the maximum annual dose may occur some time in the distant future. In the present study it has been decided in this context to present as a result the dose received during year 500 after the maximum individual intake of radioactive materials during one year has taken place, but assuming that the leakage is the same during each of these 500 years. For calculation of this dose the mathematical relationship between truncated dose commitment and dose after a certain number of years of repeated release of the same magnitude is used. This relationship is illustrated in figure 2.1.

When a large population is involved, the collective dose is a relevant measure of the detrimental effect a release. In case of a single release the maximum annual collective dose, like the individual, will occur during the first year. In this case also the collective dose commitment is calculated as well, and presented in this report.

For a continuous release the collective maximum annual dose is calculated as in the case of the individual dose. The population is however assumed to be the same in the year in which the maximum dose is encountered, as it is today. This is clearly very improbable, but this assumption is used for lack of anything better.

For single release as well as continuous release the population group involved in the collective dose is quite large; in the first case the whole population within a 500 km radius, and in the second case the population around the Baltic Sea. The population groups chosen are dictated by the needs of the respective computer programs.

In the case of continuous release it has not been possible to calculate a total dose commitment. The reason is connected with the fact that the doses from these releases are dominated by very long-lived nuclides. The calculation models used are simplified in such a way that when a certain amount of radioactive materials has been released, the only reduction in dose with time is caused by radioactive decay. In reality there will be a number of other effects that will reduce the doses with time. For example the materials may gradually be fixed in soil so that they are no more available for root uptake by plants, they may settle as sediment in the deep ocean, be transferred to the upper layers of the atmosphere, or enter into organic matter not part of the portion of the biosphere relevant to humans. An alternative to simulation of these effects is calculation of a truncated dose commitment, but it is difficult to justify a specific truncation time. In the present study dose commitments have been calculated for some of the scenarios, only involving nuclides of moderate half-lives.

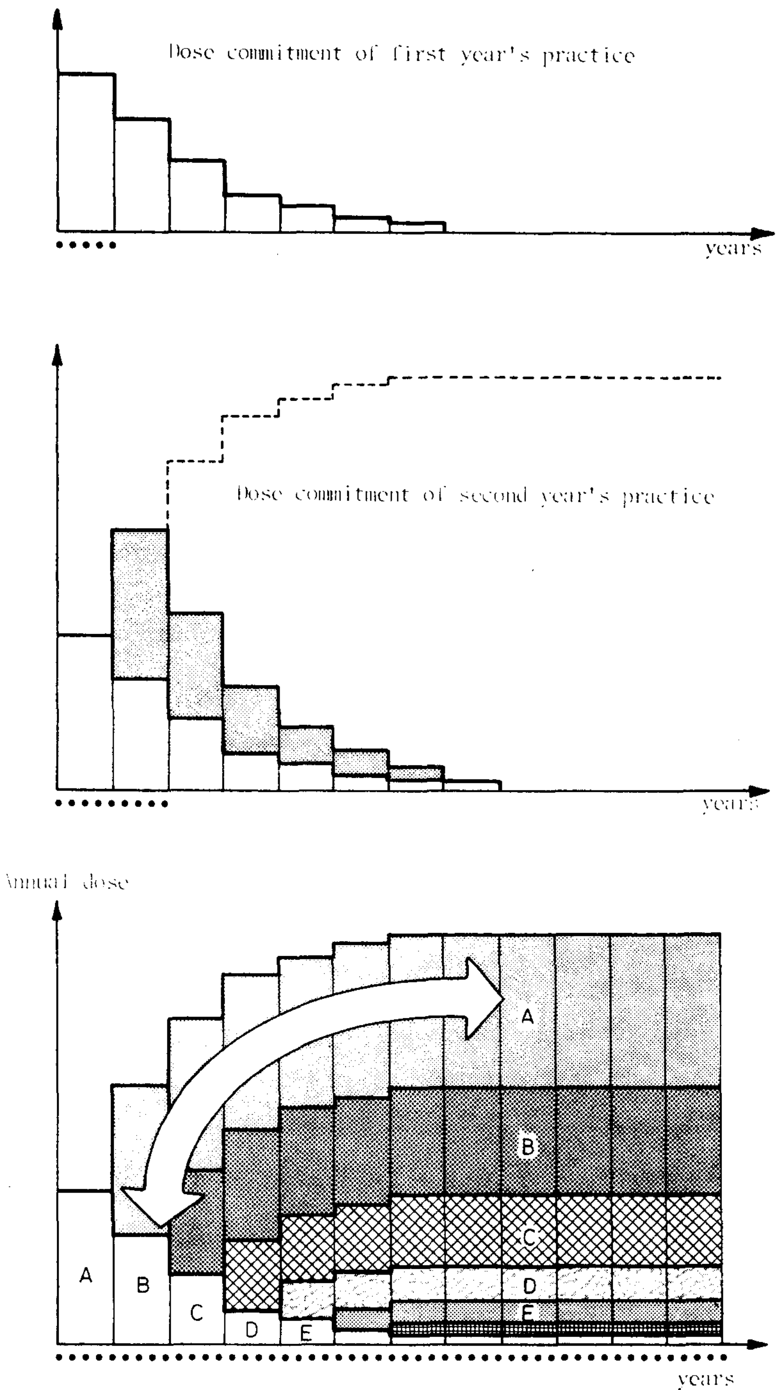


Figure 2.1 Illustration of the relation between the annual dose commitment and the annual dose in an equilibrium situation

2.4 EVALUATION OF PROBABILITIES

In this safety analysis events that could conceivably cause radiation exposure are identified. The magnitude of the consequences may be influenced by siting or design requirements. The probabilities depend upon the design requirements. Accordingly, if consequences or probabilities are unacceptable, it may be necessary to change the siting or design. For a majority of the scenarios analysed in the present study it has not been possible to determine the probability. Accidents of this type have not occurred, so the probabilities can not be derived from experience. In the case of transportation accidents an estimate has been made from relevant statictical material. This is also the case for the fire in storage, where the critical event is an airplane accident.

For the bitumen fire cases, so-called risk curves have been presented. These curves are not intended to give the full picture. They are based upon what is considered to be the maximum release from a fire. Possibly smaller releases would be more probable, and might dominate parts of the risk curve. The high-consequence end of the risk curve is however felt to be representative.

2.5 IMPORTANT FACTORS AND PARAMETER VALUE VARIATIONS

One aim of the safety analysis has been to identify factors of importance to safety, and certain specifications that a waste product or parts of a waste handling system should meet.

Another important aspect is identification of calculation assumptions and parameters to which the results are sensitive. A full sensitivity analysis of the various models used in the calculations would be very time-consuming. A limited analysis has however been performed, and is presented in chapter 7 of this report.

3 STORAGE, DESCRIPTION OF CALCULATIONS

3.1 GENERAL

In this and the following chapter will be described the safety analysis performed for the reference system, as described in Technical Part I.

This chapter discusses the risks related to temporary storage of solidified radioactive waste from operation of nuclear power plants. The discussion includes normal as well as abnormal events that may occur during storage and the related handling of the waste packages that may result in release of radioactive materials to the environment and, eventually, to radiation doses to the public.

Some of the waste packages, including the reference waste, contain sufficient radioactivity to make it necessary to provide radiation protection for the operators, for instance by using remote handling, shielding containers or shielded operator cabins at trucks or cranes used in handling operations.

The reference storage system and the reference waste packages are defined in Technical Part I.

3.2 POTENTIAL HAZARDS

3.2.1 NORMAL EVENTS

- Handling

Normal events include handling by transfer of the waste packages from the waste treatment plant to the storage, placing the packages in the storage, and removal from the storage for transportation to the respository (the risk of transportation is covered in the following chapter). The handling includes minor mechanical shocks and impacts that must normally be expected in similar industrial processes.

The tests of waste packages carried out in Finland /8/ have clearly shown that the packages are extremely resistant to mechanical forces. On the basis of these tests, none of the mechanical forces inflicted on the waste packages during normal handling are of sufficient magnitude to cause any damage to the packages that will result in release of radioactive materials, neither can they cause any damage that may have detrimental effects later on during transportation or disposal.

- Environment

Normal events also include influence from the environment, such as weather condition during transfer from the treatment plant to the storage, temperature and humidity during storage, including sub-zero temperatures. The influence of the weather can be ignored due to the short duration of the exposure.

Temperature and humidity in the storage environment may, however, have some influence on the waste packages. While a few periods of sub-zero temperatures will have no measurable influence on the reference waste packages, a large number of freeze-thaw cycles cause deterioration, in particular in the case of the packages with cementized waste. High water content in the waste packages will tend to increase this effect. The risk of freeze-thaw cycles is, however, dependent on the type of storage building and on the climatic conditions. In many cases, covering the ventilation openings during low temperature periods will be sufficient to limit the risk. If freeze-thaw cycles can not be limited to a minimum, the waste packages must be able to withstand such influence.

- Corrosion

For storage of waste incorporated in bitumen, the integrity of the drums is of major importance, as bitumen is not formstable. Two ways are available to counter this hazard:

- use of high quality materials for the drums, for instance stainless steel
- control of the air conditions in the storage facility, i.e. maintain low relative humidity and avoid large temperature variations

The latter solution was chosen for the reference design for the storage of bituminized waste (see Technical Part I).

For waste incorporated in cement, the drum is not of the same importance, and some corrosion of the drums can be tolerated.

Protection of the drums with a suitable paint will for most conditions provide sufficient protection for the periods the waste will be in storage (normally 5, maximum 50 years).

- Processes in the waste

Normal events include, to a certain degree, also processes originating in the waste packages themselves to the extent that such processes may have any significant influence within the maximum period of time that the waste is stored.

Possible damage or deterioration of the waste packages due to "internal" processes have been evaluated. One such process is radiation damage, in particular to bitumen. However, even the maximum nuclide content in the waste is insufficient to cause any damage /4/. Also the effect of micro-organisms have been evaluated, but any such effects can be excluded for the periods considered in this connection /5/.

Another process that could be of importance is swelling of the waste as this could cause breaching of the waste package. Therefore swelling, for any cause, should be limited.

For bituminized waste, swelling of the waste up to 5 % by volume can normally be tolerated as the free volume in the filled drums is about 8-10 %, and the viscosity of the bitumen is sufficiently low to permit swelling without causing excessive pressure on the drum. However, the results from the latest experiments indicate that swelling of bitumen immersed in water can exceed this limit, see Technical Part III, chapter 5. For waste incorporated in cement, swelling of the waste matrix is more serious, as even moderate swelling may cause cracking in the waste matrix as well as rupture of the steel drum or of the concrete mould. Although these effects will normally not represent any significant hazards while stored, affected packages will tend to increase the consequences if involved in abnormal events during handling and transportation, and the leach rate from such packages will be higher than normal when disposed. For cementized waste it has been shown /6,7/ that cracking or rupture can be avoided if swelling is kept below 0.3 % by volume. The reference waste packages (see Technical Part I) will fulfill this requirement. (The concrete moulds are equipped with 10 mm foam plastic mats that will absorb minor swelling).

3.2.2 ABNORMAL EVENTS

Abnormal events during storage and related handling operations include collision or drop of a waste package during handling and excessive heat due to fire. These accidents are analysed in more detail in sections 3.3 and 3.4.

3.3 DROP AND COLLISION

The maximum lifting height in the reference storage facilities is about 7 m for the two surface facilities and about 14 m for the rock cavern storage. A drop from 7 m corresponds to an impact velocity of 40 km/h, 14 m to 55 km/h, and it is very unlikely that the fork lift handling the waste packages will exceed a speed of 40 km/h within the controlled area (accidents at higher speeds are analysed in section 4.3).

The drop tests carried out in Finland /8/ have shown that although the impact will cause a considerable increase in the surface area of the waste matrix, no significant release should be expected from waste packages corresponding to the reference design (see Technical Part I).

Should a package of inferior quality be involved in such an accident, the event will take place within a controlled area, most probably inside the storage building, and can not cause any exposure of the general public. The Finnish experiments have alo shown that the fraction of dust-size particles produced by the impact is extremely small. Clean-up of the debris will therefore be a rather trivial operation, which the operators should be well prepared for.

3.4 BITUMEN FIRE

3.4.1 BASIC ACCIDENT ASSUMPTIONS

Although most bitumen types are difficult to ignite, a fire in the storage facility for bituminized waste may cause ignition of the waste/bitumen mixture. The fire is assumed to occur in one of the storage pits where the cover is removed, and that two layers, i.e. 8 drums are involved. Even if none of the fire fighting systems are working, the fire is likely to die out due to lack of oxygen, since it takes place in a very confined space.

The basic assumptions for the calculation of doses and risks are as follows:

- The 8 waste drums involved contain RWCS-resin shortly after solidification, with the following nuclide content:

Co-60:	800 GBq	(22	Ci)
Cs-137:	270 "	(7.2	Ci)
Cs-134:	270 "	(7.2	Ci)
Sr-90:	13 "	(0.36	Ci)

  The longlived radionuclides also included in the reference waste (Technical Part I) are not considered in these calculations due to the very small amounts involved.

- During the fire 100 % of the cesium is released as gas. Of the remaining nuclides 60 % is released as gases or particles smaller than 10 microns, 25 % is released as particles larger than 10 microns and 15 % will remain in the ashes /9, 10/. Based upon the results from the latest experiments /17, 18/, these assumptions are probably very pessimistic, see Technical Part III, chapter 3 and 6.

- The fire temperature is about 1000 degrees C, will last one hour and the energy release is 48 GJ /10/. However, most of the heat will be given off to structural material of the storage building, so that the temperature of the release will not significantly exceed the ambient temperature.

- Doses and risks have been calculated using the computer program CRAC /37/, originally developed for use in WASH-1400. For further details, see /12/, /13/, /14/ and /15/. All the calculated doses are effective committed doses.

- Data on meteorology and population distribution are specified in Technical Part I.

- The storage building is 10 m high and 30 m wide.

3.4.2 RESULTS OF CALCULATIONS

Two quite different types of results are presented: individual doses as a function of distance from the release point, and the collective dose commitment, used to describe the collective risk associated with a certain release, taking into account the probability of different weather conditions.

The individual doses as a function of distance are calculated assuming constant weather conditions. Calculations have been performed for two different weather conditions: Pasquill C and wind speed 3 m per second, and Pasquill F and wind speed 1 m per second. The calculations could have been performed for other weather conditions as well, but it is felt that these two weather conditions indicate how the doses may vary from one set of weather conditions to another.

The results of these calculations are shown in figures 3.1 and 3.2. Figure 3.1 shows the effective committed dose as function of distance from the release point for the two weather conditions. The maximum value is about 0.1 Sv (10 rem).

The contribution of the different radionuclides are shown in figure 3.2 for the Pasquill C condition. While the contribution from Sr-90 is rather insignificant, the other three nuclides have approximately equal contribution.

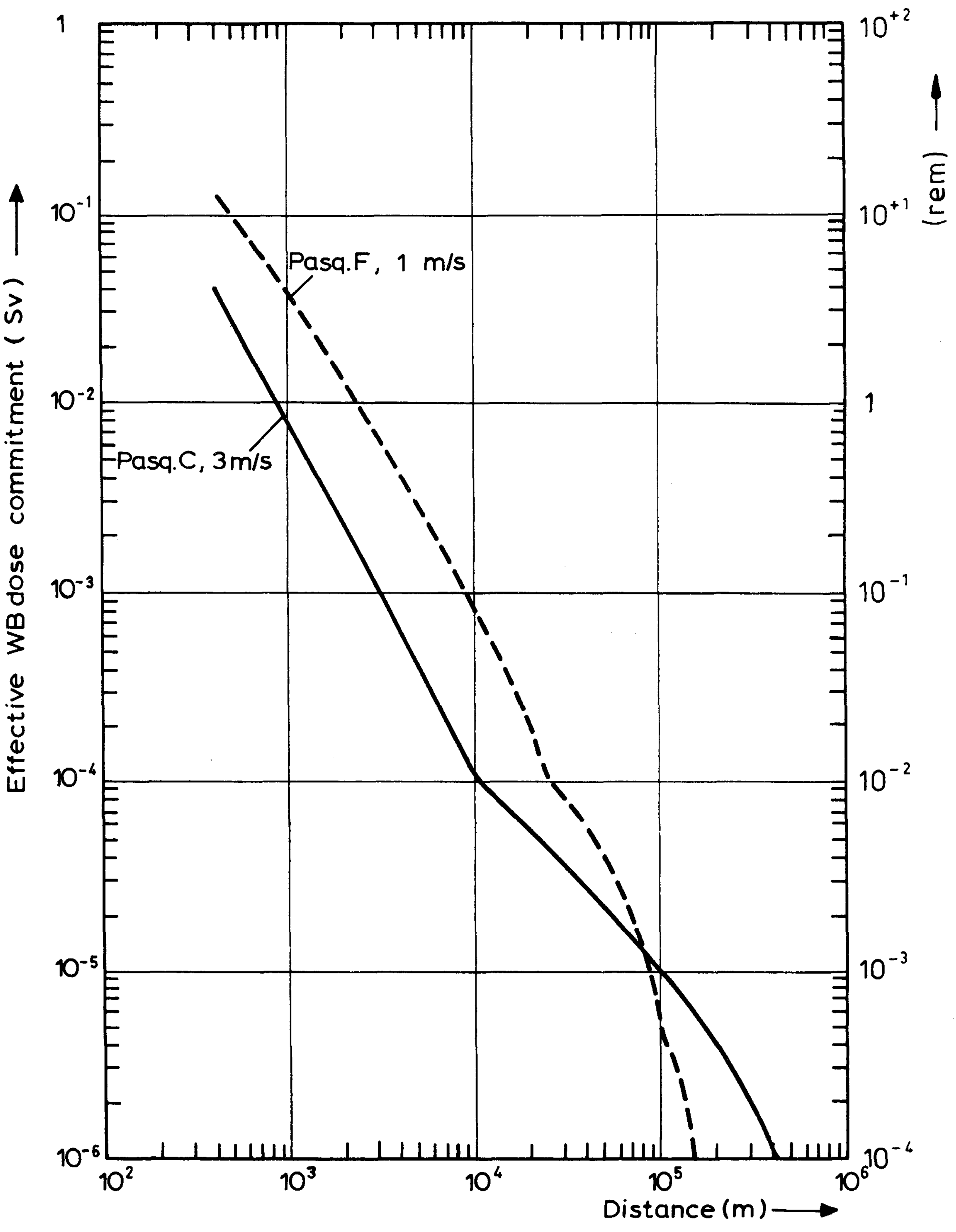
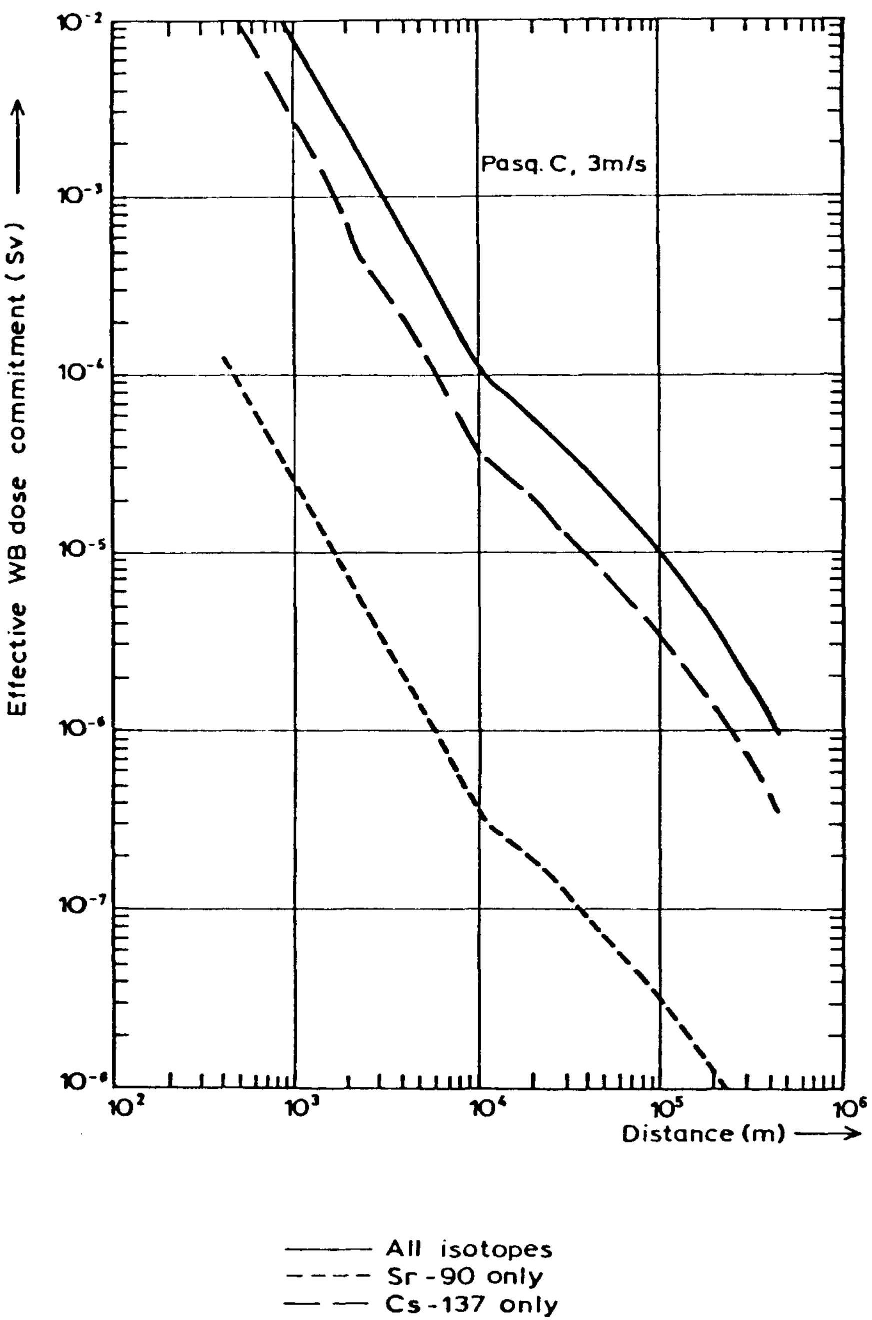


Figure 3.1 Doses resulting from bitumen fire in storage building.



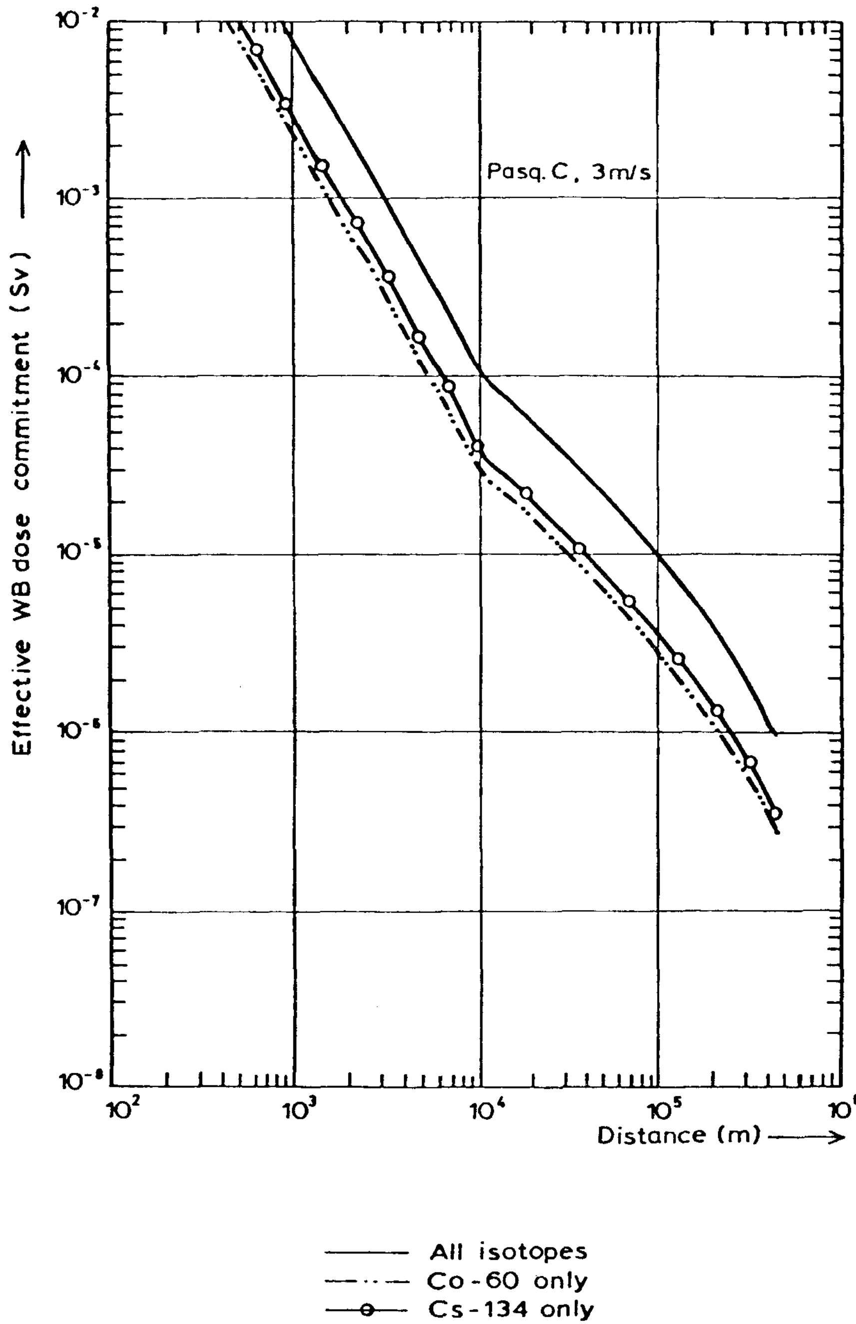


Figure 3.2 Release from bitumen fire. Dose-contribution from individual nuclides.

For one of the the selected weather conditions another breakdown of the results has also been performed, showing how the different exposure pathways contribute to the total dose. The results are shown in table 3.1, where the percentage of the dose received via each exposure pathway is shown.

Table 3.1 Percentage of dose received via each exposure pathway

External radiation from passing cloud	0.005 %
Inhalation from passing cloud	0.01 %
Inhalation of resuspended activity	0.2 %
External radiation from activity deposited in ground	44.0 %
Exposure via nutrition	56.0 %

The dose is received almost exclusively via two longterm exposure pathways. The relative contribution may be somewhat different for other weather conditions and release scenarios, but the difference is not expected to be large.

For calculation of the risk it is necessary to take a number of factors into account. In the present calculations based on the population distribution specified in Technical Part I, the consequences of a release are expressed as the resulting collective doses. The consequence might be expressed in other ways, e.g. the number of cases of cancer or genetic damage expected to result from the release (using certain assumptions regarding the relationship between cancer/genetic damage and the dose).

Besides population distribution and consequence calculations, the weather statistic may also be taken into account with a varying degree of refinement. In the present calculations a whole year of meteorological data on an hour-by-hour basis is stored in a data file. For each hour throughout the year the atmospheric stability, the wind speed and information on whether there is precipitation or not is stored. Wind direction information is included in a simpler manner, as four frequency distributions, one for each of the seasons.

It would be possible to calculate the consequence of the release (the collective dose) for each one of the 8760 hours in the year, and construct a probability distribution from the results. The computer time required is, however, in many cases prohibitive, and in the present case calculations have been performed for 91 different hours, distributed over the year in a semi-random manner, and the probability distribution is constructed from these results, where each result has a probability of (1/91). Such a probability distribution is often called a risk curve.

The results of these calculations are shown in figure 3.3, which shows the risk assuming that the fire has or will take place. The probability of the fire is discussed below.

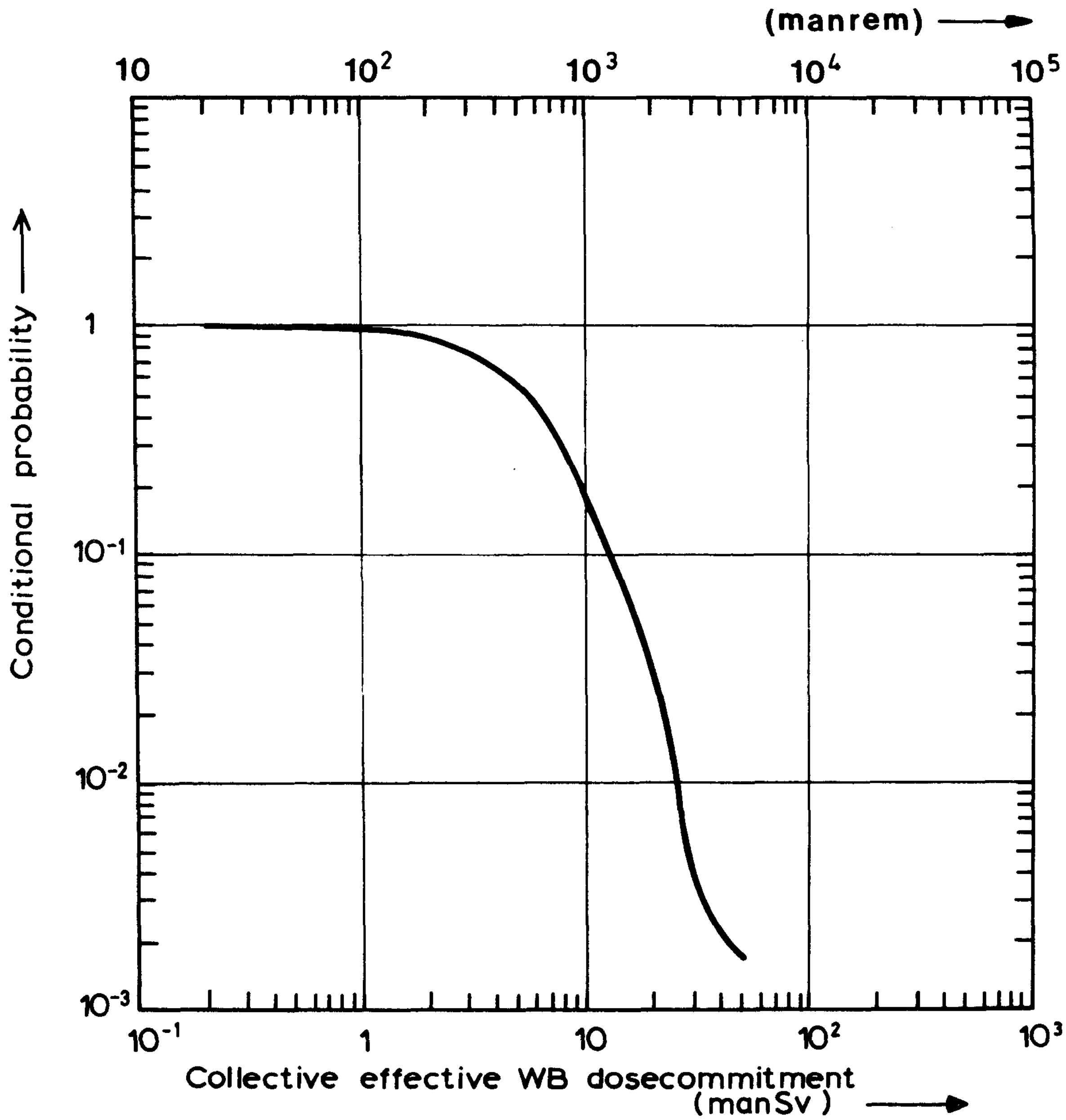


Figure 3.3 "Risk" curve. Bitumen fire in storage.

3.4.3 PROBABILITY OF BITUMEN FIRE

Attempts have been made to estimate the probability of a fire in the stored bituminized waste. The only causes for a fire in a storage that have been identified are:

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

The probability of sabotage is difficult to estimate, but with the security measures in force at nuclear power plants, such action seems unlikely, particularly when one considers the considerable amount of inflammable material that would be needed to ignite the bituminized waste, compared to the relatively small consequences of the action. The probability of an airplane crashing on the rector building can, in the Nordic countries, be assumed to be less than 10^{-7} per year /11/, and this risk is considered acceptable by the authorities. On this basis it must be assumed that the risk of a plane crash at the waste storage would also be acceptable.

A large forest fire could represent a potential risk at a few reactor sites. This risk should - also for other reasons - be considered during the site evaluation. Sufficient deforested belt and adequate fire fighting equipment can reduce or even eliminate this risk.

A severe failure of the overhead crane could roughly be estimated to have a probability of 10^{-4} per year, however, it seems extremely unlikely that such failure will lead to ingition of the stored waste.

3.4.4 FACTORS TO BE CONSIDERED

Although the probability of a severe fire in stored bituminized waste seems very remote, the consequences are sufficiently serious to justify countermeasures to reduce the risk. The arrangement of the reference storage facility for bituminized waste reflects such precautions:

- The storage pits physically limit the number of drums that could be involved in a fire.
- When the covers of the storage pits are in place, a fire will very soon die out due to lack of oxygen.
- Installation of smoke detectors ensures early alarm in case of fire.
- Fire extinguishing systems make an uncontrolled development of a fire very unlikely.
- Access control reduces the risk of sabotage.
- The risk of airplane crash is, for other and more serious reasons, always considered in the site evaluation for a nuclear plant.
- Risk of a large forest in the vicinity of the storage facility should be considered also for other reasons. This risk can easily be controlled by keeping sufficient distance between the storage facility and the forest.

The above discussion primarily relates to the surface storage facility. The underground storage facility is briefly discussed below.

A bitumen fire in the underground storage will most probably lead to a relatively small release of radionuclides to the environment, since the larger proportion of the nuclides will be deposited on all the cold surfaces that the smoke will pass before release to the environment. This will result in a very troublesome contamination of surfaces in the storage cavern and tunnels. One way of preventing the fire would be to store the drums in concrete containers (each housing 8 drums).

With no inflammable material available (except for the waste) the possibility of a fire in the underground storage can be ignored for causes other than sabotage. In addition, an effective access control should be easier and simpler to implement for the underground than for the surface storage.

3.5 FIRE IN A STORAGE FOR CEMENTIZED WASTE

A fire in the storage for cementized waste can not be totally excluded, although it is even less likely than in the storage for bituminized waste, and the consequences of a fire will be significantly less than for a bitumen fire. Experiments carried out in Finland /16/ have shown that the release from a drum with cementized waste exposed for one hour to a fire with a temperature of 800 degrees C. will be less than 13 % of the total radionuclide inventory in the drum. However, other examinations during the same test indicate that probably only a surface layer of a few centimeters was affected by the fire. This implies that for cementized waste packed in concrete moulds, no release at all could be expected, even if a fire lasted for several hours.

4 TRANSPORTATION, DESCRIPTION OF CALCULATIONS

4.1 GENERAL

This chapter discusses the risk related to transportation of the low and medium level radioactive wastes from nuclear power plant operation. The discussion includes normal as well as abnormal events that may occur during transportation and related handling of the waste packages, and which may result in release of radioactive material to the environment and, eventually, to radiation doses to the public.

The reference transportation system and the reference waste packages are defined in Technical Part I.

4.2 POTENTIAL HAZARDS

4.2.1 NORMAL EVENTS

- Handling

Normal events during transportation include handling during loading and unloading operations, vibration and minor mechanical shocks and impacts that must normally be expected during transportation, and movements during sea transportation in rough weather.

The tests of waste packages carried out in Finland /8/ have clearly shown that the packages are extremenly resistant to mechanical forces. On the basis of these tests, none of the mechanical forces inflicted on the waste packages during normal handling are of sufficient magnitude to cause any damage to the packages that will result in release of radioactive materials, neither can they cause any damage that may have detrimental effects later on. Furthermore, all waste packages will be transported in containers that will at least provide some protection of the waste packages. Transport containers for packages containing significant amounts of radionuclides will, for shielding purposes, have about 300 mm thick concrete walls or about 90 mm thick steel walls that will provide a significant physical protection of the waste packages. With the basic assumptions for the reference waste packages one cannot envisage that any normal handling, however rough, could cause radioactive release from the waste packages.

- Environment

Normal events also include influence of climatic conditions such as rain, temperature variations, sub-zero temperatures etc., during transportation or short-time storage integrated in the transportation schedule.

Of the influences caused by climatic factors, only frost may have any measurable effect on the waste packages. All waste products contain small amounts of water, and repeated freeze-thaw cycles could cause cracking of the waste matrix, in particular in cementized waste. However, the relatively short travelling times foreseen in the Nordic countries automatically limit the number of cycles that the waste could be subjected to, and furthermore, the container combined with the mass of the waste packages will have a significant smoothing effect. Tests carried out /19/ have shown that the reference waste packages are rather resistant to freeze-thaw cycles and for these or similar packages no special precautions need to be taken.

The high and low temperatures that may be experienced in the Nordic countries are also not so extreme that they can have any detrimental effect on the waste packages. Even extended exposure to rain or snow would have hardly any influence on the waste packages even if they were directly exposed. The tranportation containers provide further protection against this type of exposure.

4.2.2 ABNORMAL EVENTS

Some of the abnormal events may have sufficiently strong influence on the waste packages to cause smaller or larger releases of radioactive materials to the environment. The following events have been considered:

Road transportation	Sea transportation
Drop, fall	Drop, fall
Collision	Collision, wreckage
Fall-in-water	Sinking
Fire	Fire

- Drop

Drop of a single waste package or of a filled transport-container may occur during loading or unloading operations. A drop from sufficient height could, of course, cause damage to the waste packages. The reference transport system does not, however, foresee any lifting of the goods higher than 1 to 3 meters, which is less than during handling in the storage. The drop tests carried out in Finland /8/ have clearly shown that the reference waste packages will suffer little or no damage at all even if dropped from 9 m onto a solid bolt. The transport container will provide additional protection. It may be concluded that the minimum quality applicable to the waste packages, no reasonably conceivable event involving drop of the waste packages will damage the packages sufficiently to result in release of radioactive materials.

- Truck accidents

A truck collision, or similar accident, may be sufficiently violent to inflict significant damage to the waste packages and cause release of radioactive materials to the environment. A truck accident may also involve loss of the payload in a river or a lake and result in release of radionuclides to the water. A collision may involve a fire in the fuel or, more seriously, involve collision with another truck carrying inflammable material. A severe fire obviously has the potential of causing significant releases of radionuclides to the atmosphere, especially if bituminized waste is involved. These three types of accidents were, consequently, selected for further study.

- Wreckage of ship

Accidents during sea transportation could include damage to and loss of one or more loaded transport containers, or it could involve sinking of the ship including total payload, or it could involve a severe fire. All these accidents have the potential of significant releases of radionuclides to the seawater or to the atmosphere; hence these types of accidents were studied in more detail.

4.3 TRUCK COLLISION

4.3.1 BASIC ACCIDENT ASSUMPTIONS

Truck accidents may cause severe impact to the payload. This type of accident could be a collision with another vehicle, a train or a stationary object; the truck could run off the road, it could overturn on or off the road.

The damage to the waste packages were basically evaluated assuming an impact velocity of 22 m/s (80 km/h). The damage to the waste package is somewhat dependent on the type of package.

The drop tests carried out in Finland /8/ showed that for cementized and bituminized waste packed in steel drums, no release should be expected if the drums were intact at the time of the accident, as the drums provide an effective barrier against release of waste material at impact velocities of 22 m/s or less. No tests were carried out with packages of exactly the same design as the concrete mould, but some full scale tests were carried out with reinforced concrete cylinders of the same dimensions as the drums. These tests showed considerable crushing and release of materials near the surface. The moulds differ from the test specimens by having a high quality concrete shell completely separated from the waste matrix by sheets of foam plastic preventing propagation of cracks from the surface into the waste matrix. It is believed that the behaviour of the concrete moulds will not differ significantly from that of the drums.

4.3.2 ESTIMATED CONSEQUENCES

Conservatively one may assume that the drums deteriorate by corrosion during the storage period. The drop tests with uncovered concrete showed in general release fractions of small particles (less than 44 microns) well below 0.1 % at impact velocities of 20 m/s (corresponding to a drop height of 20 m). At velocities of 27 m/s (corresponding to a drop height of 43 m) the maximum release fraction of small particles was 0.2 %. Only particles smaller than 20 microns may be dispersed over longer distances, while particles larger than 0.5 mm will settle close to the accident location. The release fraction in the form of "lost" fine particles is thus about one thousandth of the total content of the packages involved. Only a rough evaluation of doses has been performed, and compared to the doses calculated for a bitumen fire, they will be quite neglectible.

The above considerations take no account of the protection provided by the transport container.

4.3.3 PROBABILITY OF ACCIDENT

The probability of a truck transport collision accident is evaluated in /20/ and /24/. Dominating parameters are the velocity change and the total over-the-road weight of the transport. A few examples of estimated probabilities are given below:

Velocity change — Probability of truck collision accidents per km for transports weighing:

km/h	m/s	30 tons	50 tons
48	13	$1.1\ 10^{-8}$	$6.4\ 10^{-9}$
80	22	$1.8\ 10^{-9}$	$7.5\ 10^{-10}$
96	27	$2.5\ 10^{-9}$	$1.3\ 10^{-10}$

Due to the transport containers, the probability of any release of radioactive material will be even lower. The probability of puncture of mild steel package walls of different thicknesses is also evaluated in /24/:

Steel with thickness	Probability of puncture
13 mm	$2.7\ 10^{-9}$ per km
25 mm	$2.0\ 10^{-9}$ "
38 mm	$3.6\ 10^{-10}$ "
51 mm	$2.9\ 10^{-11}$ "

The probability of a truck collision resulting in release of radioavtive materials can thus be estimated to be of the order of 10^{-9} per transport kilometer, or probably even lower.

The total transportation distance (excluding empty return for the reference waste (RWCS plus SFPCS resin) from 30 years operation of 6 reactors, is for

- Bituminized waste in drums: 192,000 km
- Cementized waste in drums: 360,000 km
- Cementized waste in moulds: 288,000 km

The corresponding probability of the truck accident discussed above is then

- Bituminized waste: 2×10^{-4}
- Cementized waste; drums 4×10^{-4}
- Cementized waste; moulds: 3×10^{-4}

4.4 FALL-IN-WATER ACCIDENT

4.4.1 BASIC ACCIDENT ASSUMPTIONS

Truck accidents may involve loss of the payload in a river or lake. Such an accident could occur during bridge crossing, roadside immersions resulting from rollover or cargo ejected from the trailer in a collision, or the truck might leave the road and end up in adjacent water. Immersed in water, radionuclides may be released from the waste and ultimately cause radiation to human beings either by intake of drinking water or via the food chains. 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 the water and prevent release for a considerable period of time. The total release will further depend on whether, and how soon, the waste packages are recovered. The study has analysed the potential consequences of this type of accident in some detail /21, 22/, and the results of this analysis are summarized in the following.

The drop test /8/ has shown that penetration of drums with solidified waste is not very likely, even when dropped on a steel bolt. An intact drum will prevent, limit or at least delay contact between water and waste. Although no drop tests with a concrete mould type package were performed, the results from the drop tests make it reasonable to assume that the concrete shell could be so severely damaged in a violent accident that the water would gain easy access to the surface of the waste product inside.

The following assumptions form the basis for the analysis of consequences of this type of accident:

- One concrete mould with 5 years old reference RWCS resin is involved (see Technical Part I).
- The package is lost in a river with a flow of 10 cubicmeters/s (a relatively small river).
  The shell of the package is "lost" as a consequence of the accident, but the waste product remains undamaged.
- The following somewhat pessimistic, but not unrealistic, initial leach rates are assumed:

  Cs: 10^{-2} per day
  Sr: 10^{-3} per day
  Co: 10^{-4} per day

- These leach rates and the shape of the waste block give the following initial releases of radionuclides to the river water:

  Cs-134: $1x10^{7}$ Bq/d ($3x10^{-4}$ Ci/d)
  Cs-137: $4x10^{7}$ Bq/d ($1x10^{-3}$ Ci/d)
  Sr-90: $2x10^{5}$ Bq/d ($6x10^{-6}$ Ci/d)

  (Preliminary calculations have shown that the remaining nuclides do not make a significant contribution to the radiation doses).

- Three main scenarios were considered:

  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.

In case 1 the initial release rate was maintained throughout the 30 days. In case 2 the initial release rate was maintained throughout the first 10 days, and then the release rate was assumed to decrease with the square-root of 1/t.

While release rates significantly larger than assumed for cases 1 and 2 seem very unlikely, they can not be totally excluded. Very high release rates might for instance result from fall from a high bridge or down a steep slope causing excessive damage to the waste package, or might result if the package involved in the accident was of inferior standard. For this scenario, case 3, it was assumed that 10 % of the cesium content was released during the first 24 hours. The leach rate from a crushed cementmatrix will be larger than that from an intact package, but the increase in strontium will be much smaller, and strontium is accordingly not included in these calculations.

4.4.2 RESULTS OF CALCULATIONS

In order to calculate the radiation doses resulting from above releases, an actual or a model site for the accident must be chosen. As no actual location was defined, the dose calculations were made utilizing an already existing model in the BIOPTH computer program /23/. This model assumes a river with a constant water flow from the location of the accident to the estuary in the Baltic Sea. The local population consists of 1200 persons, the regional population is 18,000 persons, and the population around the Baltic Sea is about 10 million persons. The results of the calculations are summarized in table 4.1. They are reasonably representative for many rivers flowing into the Baltic Sea, Kattegat, Skagerak, or the North Sea. For cases 1 and 2, the doses are shown graphically in figures 4.1 - 4.4, as a function of time.

In many cases the radiation doses to a critical group are of particular interest, and those doses are also calculated by the BIOPATH program. In this particular case the individual as well as the collective doses are so small that doses to a critical group can be of hardly any interest.

Some factors influencing the radiation doses are discussed in the following:

- The maximum number of concrete moulds per transport is two if shielding is required. If shielding is not required, more moulds can be transported on the same truck, but total inventory of radionuclides will be significantly less than for the calculated examples.

- The radiation doses are dominated by Cs-137, Cs-134 and Sr-90 in that order. The inventory per mould of the cesium isotopes could be up to 10 times higher than the values used in the dose calculations, while inventory of Sr-90 could be 4 times higher. Moulds with this maximum inventory will most probably require additional shielding; hence only two moulds can be transported on the same truck. The additional shielding will also provide additional protection of the package.

- The maximum individual doses are proportional to the activity concentration in the river water. A river with a flow of 10 cubic meter per second is a fairly small river. If the river flow was much less than 10 cubicmeter per second it is very likely that the waste packages could be recovered within a day or two.

- The collective dose does not directly depend upon the water flow, but rather on the total release. The reason for this is the assumption (in the calculations model) that 2 % of the water flow (containing of course 2 % of the activity released) is used for irrigation.

An extremely conservative estimate has been made, using the following and purely hypothetical assumptions:

- the mould involved contained the maximum nuclide inventory
- all cesium was released during the first 24 hours
- the river flow was only 1 cubic meter/s.

The resulting maximum individual dose would increase by a factor 10 x 10 x 10 to 7.5 mSv (0.75 rem) during the first year (if no countermeasures were taken).

Table 4.1 Fall-in-water accident - summary of dose calculations

	Nuclide	Case no.		
		1	2	3
Individual dose: Max.annual dose, x) Sv:	Cs-137	8.0×10^{-7}	1.2×10^{-6}	6.0×10^{-6}
	Cs-134	3.5×10^{-7}	4.4×10^{-7}	1.5×10^{-6}
	Sr-90	3.6×10^{-7}	4.7×10^{-7}	-
	Σ	$1.5 x^{10-6}$	2.1×10^{-6}	7.5×10^{-6}
Collective dose: Total collective dose, manSv	Cs-137	5.7×10^{-3}	5.0×10^{-2}	4.0×10^{-2}
	Cs-134	1.1×10^{-2}	3.4×10^{-2}	6.6×10^{-3}
	Sr-90	1.7×10^{-3}	1.6×10^{-2}	-
	Σ	1.8×10^{-2}	6.9×10^{-2}	4.7×10^{-2}

x) Based on the (very unlikely) assumption that one single person throughout a whole year each day is receiving the maximum individual daily dose.

Note: 1 Sv = 100 rem.

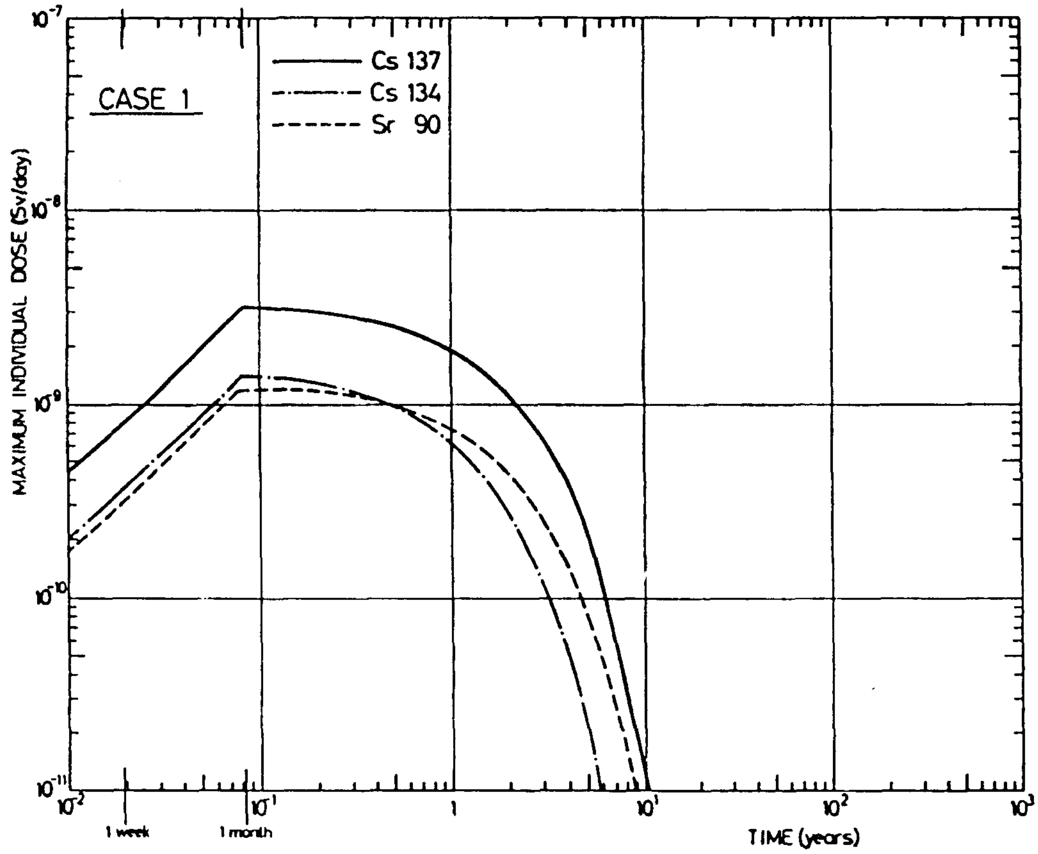


Figure 4.1 Fall-in-water accident

Case 1: Maximum individual daily dose

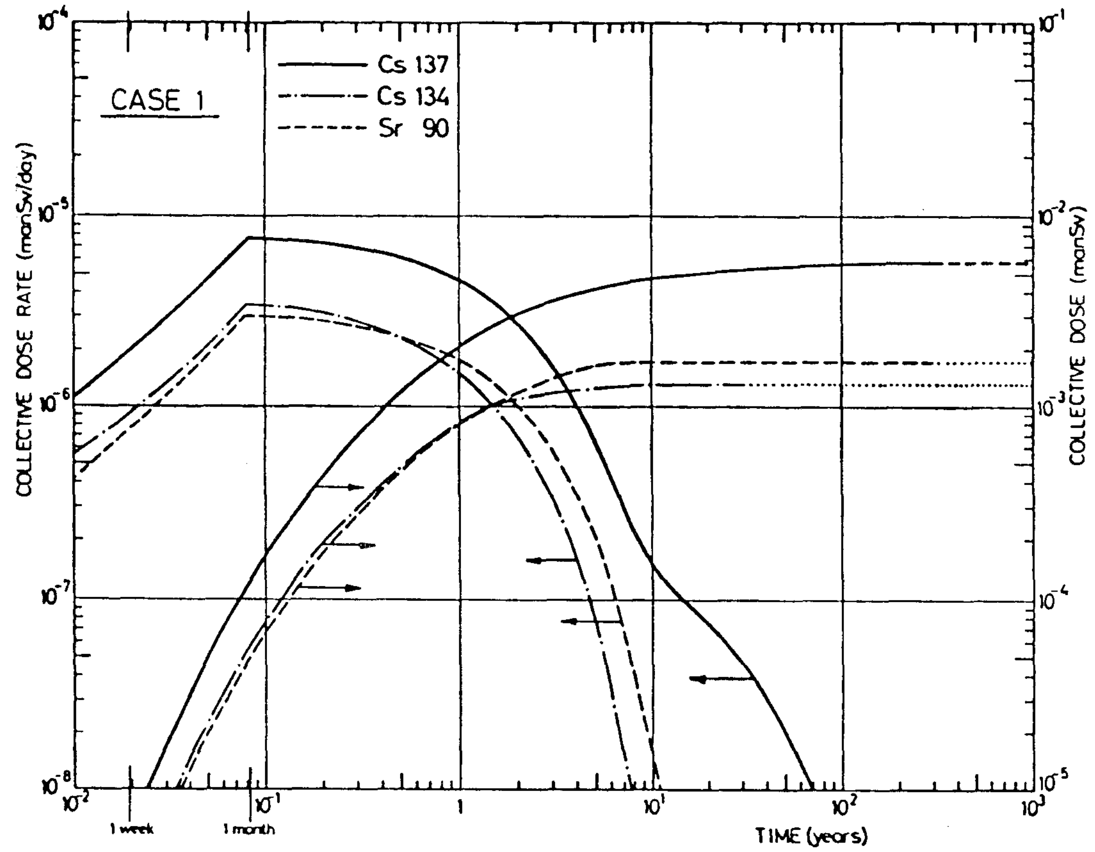


Figure 4.2 Fall-in-water accident

Case 1: Collective dose rate and collective dose

Note: 1 SV = 100 rem

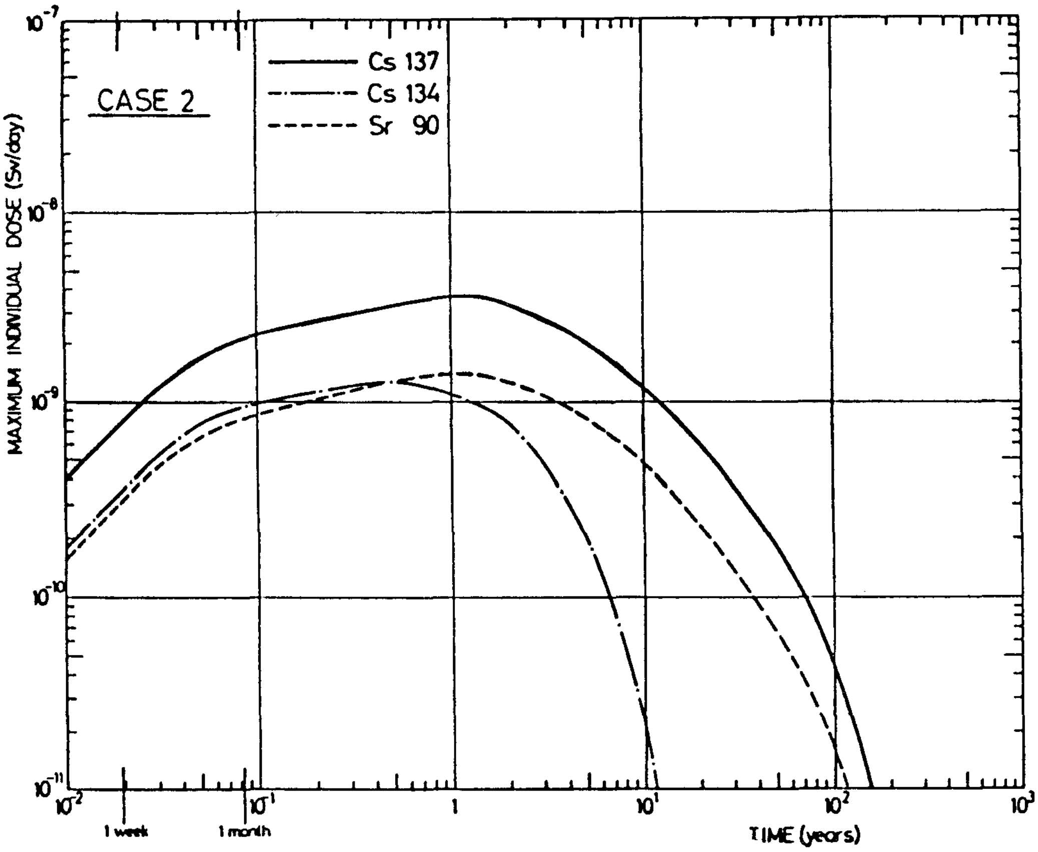


Figure 4.3 Fall-in-water accident
Case 2: Maximum individual daily dose

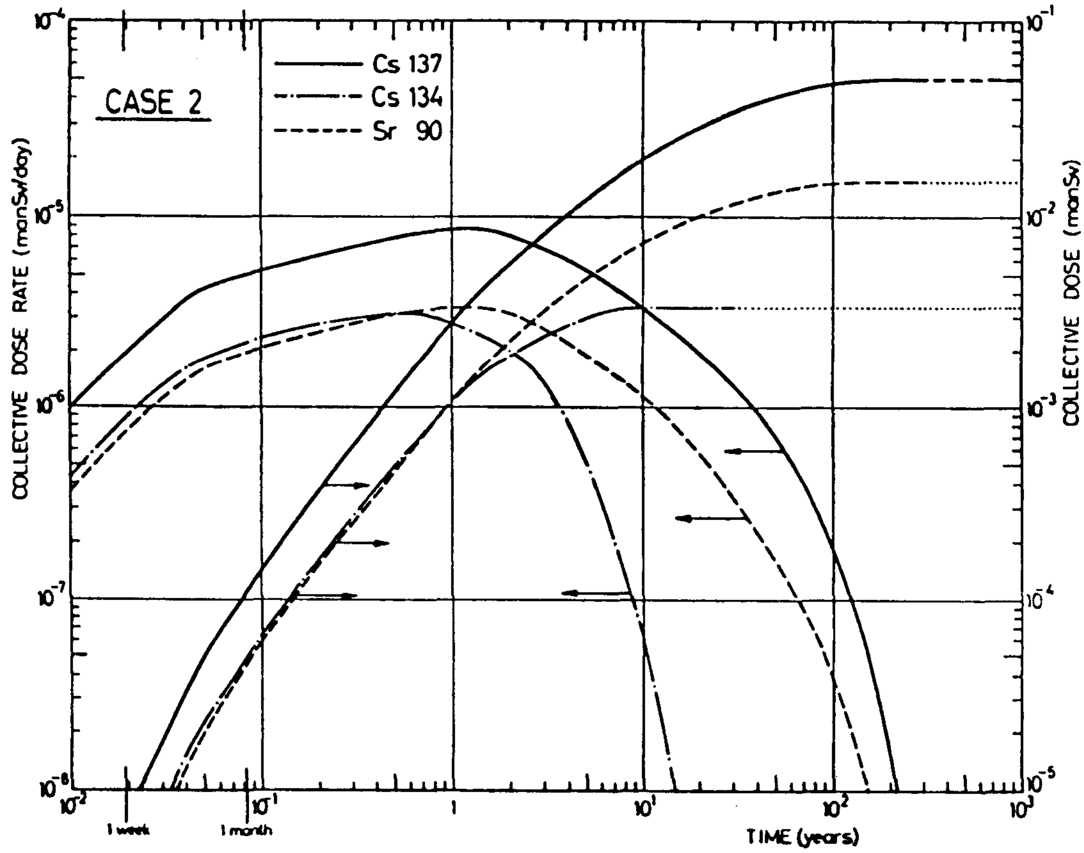


Figure 4.4 Fall-in-water accident
Case 2: Collective dose rate and collective dose
Note: 1 Sv = 100 rem

4.4.3 PROBABILITY OF ACCIDENT

The probability of a truck accident resulting in immersion of the payload in different water depth is evaluated in /20/:

Depth more than	1 meter	Probability 1×10^{-10}
"	3.5 meters	" 5×10^{-12}
"	10 "	" 1×10^{-12}

The probability is accordingly less than 10^{-10} per transport kilometer. The probability of a fall-in-water accident at more than 2 meters of water depth combined with breaching of the transport container is estimated to be less than 10^{-11} per transport kilometer.

The total transport distance for the reference waste (RWCS plus SFPCF resin) from 30 years operation of 6 reactors varies from 192,000 to 360,000 km for different types of packages. The corresponding probability of the truck accident discussed above is then, for all types of waste packages less than 10^{-6}.

The resulting doses are very moderate. Combined with the very low probability of this type of accident it does not seem reasonable to further evaluate the differences in the behaviour of the various waste packages if subjected to such an accident.

4.5 BITUMEN FIRE

4.5.1 BASIC ACCIDENT ASSUMPTIONS

A traffic accident involving a fire may cause ignition of the waste/bitumen mixture. The accident is assumed to involve the content of one 8-drum container. The container is severely damaged, a fire (for instance in the fuel) ignites the waste/bitumen mixture, and the content of the

8 drums burns out without any effort to extinguish the fire. The basic assumptions for the calculation of doses and risks are as follows:

- The 8 waste drums involved contain 5 year old RWCS-resin with the following nuclide content:

Co-60:	350 GBq	(9.5 Ci)
Cs-137:	240 GBq	(6.4 Ci)
Cs-134:	48 GBq	(1.3 Ci)
Sr-90:	10 GBq	(0.27 Ci)

The longlived radionuclides also included in the reference waste (Technical Part I) are not considered in these calculations because of the very small amounts involved.

- During the fire, 100 % of the cesium is released as gas. Of the remaining nuclides, 60 % is released as gases or as particles smaller than 10 microns, 25 % is released as particles larger than 10 microns, and 15 % will remain in the ashes /9, 10/. Based on the results from later experiments /17, 18/, these assumptions are probably very pessimistic, see Technical Part III, chapter 3 and 6.

- The fire temperature is about 1000 degrees C., will last 1 hour, and the energy release is 48 GJ /10/.

- Doses and risks have been calculated using the computer program CRAC /37/, originally developed for use in WASH-1400. For further details, see /12/, /13/, /14/ and /15/. All the calculated doses are effective committed doses.

- Data on meteorology and population distribution are specified in Technical Part I.

- The wreck of the truck is 3 m high and 10 m long.

Many of the parameter values used here are identical to the ones used in the calculations made for a bitumen fire in a storage building.

4.5.2 RESULTS OF CALCULATIONS

As for the bitumen fire in the storage building, two different types of results are presented. The individual doses as a function of distance from the release point, and the risk associated with a certain release, taking into account the probability of different weather situations.

The calculation of individual doses is shown in figure 4.5 for two weather conditions. The maximum value is about 0.01 Sv (1 rem).

This is about 10 times less than the corresponding dose calculated for a fire in the storage (see section 3.4). The difference is partly due to the lower nuclide inventory (an additional decay time of 5 years), but the main reason is the warm release from the traffic accident, giving thermal buoyancy to the release.

The risk is calculated for the actual Oslofjord-site, chosen as an example, as well as for a hypothetical city location (see Technical Part I). The results are shown in figures 4.6 and 4.7.

4.5.3 PROBABILITY OF BITUMEN FIRE

The most credible cause for a fire during truck transportation is the truck being involved in some sort of accident that subjects the payload to sufficiently high temperatures long enough to cause ignition of the waste/bitumen mixture.

Sabotage by means of fire does not seem very likely when one considers the considerable efforts required to first breach the transport container and then ignite the bituminized waste.

The probability of a road transportation accident resulting in a fire of a certain duration is evaluated in /24/.

Duration	Probability
more than 10 min.	1×10^{-8}/km
" " 20 min.	3×10^{-9}/km
" " 30 min.	1×10^{-9}/km
" " 60 min.	4×10^{-10}/km

Experiments carried out in Finland /10,16/ and in Sweden /25/ have shown that a waste/bitumen mixture does not ignite very easily, and that the drums must be exposed to 800-1000 degrees C. for 10 to 20 minutes before the bitumen starts to burn. Reference waste is transported in shielded transport containers. With the container intact or only moderately damaged it will form an effective protection of the waste packages against high temperatures, and even a fire lasting several hours will not be sufficient to ignite the bituminized waste.

If, however, the transport container is so severely damaged that the waste packages are directly exposed to the fire, a duration of about 10 minutes may be sufficient to cause ignition of the bitumen. A traffic accident with such severe damage to the containers has an even lower probability than a 10 minutes fire. Reference /24/ has thus given the probability of puncture of a mild steel package wall of different thicknesses in a traffic accident as follows:

Wall thickness	Probability
13 mm	2.7×10^{-9}/km
25 mm	2.0×10^{-9}/km
38 mm	3.6×10^{-10}/km
51 mm	2.9×10^{-10}/km

No statistics have been identified that correlate or combine mechanical damage and fire duration, but the same report /24/ states that only 24 % of collision accidents also involve fire.

Based on the above discussion, it seems reasonable to assume that the probability of a traffic accident leading to a fire in bituminized waste is less than 10^{-8} per km for waste packed in an unshielded transport container only. For transportation of waste of the reference composition, a shielded transport container will be used. The probability of a fire in the bituminized waste under these circumstances can be assumed to be less than 10^{-10} per km.

The total transport distance for the reference waste (RWCS plus SFPCS resin) from 30 years operation of 6 reactors corresponding probability of the transport accident discussed above is then about 2×10^{-5}.

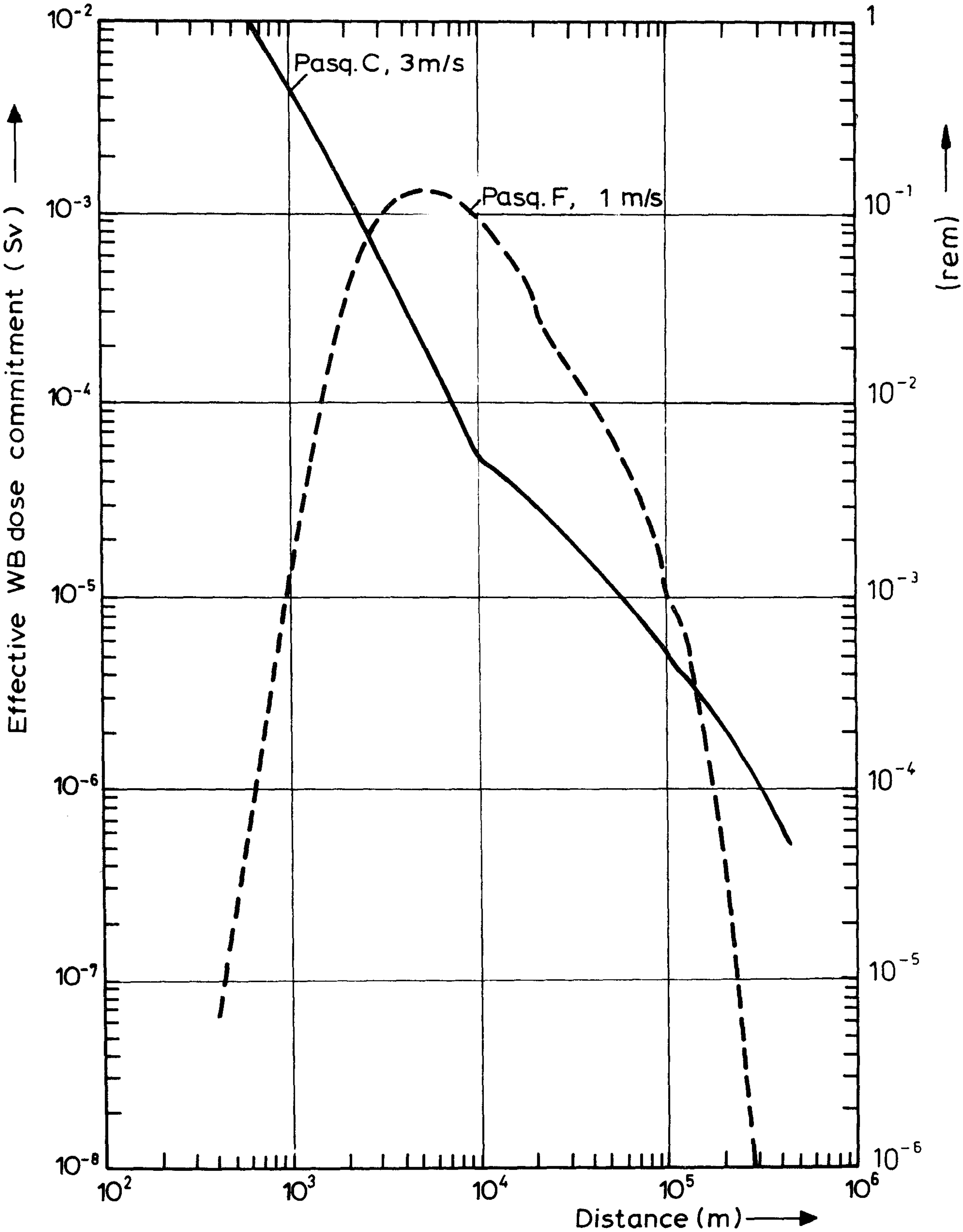


Figure 4.5 Doses resulting from bitumen fire after traffic accident.

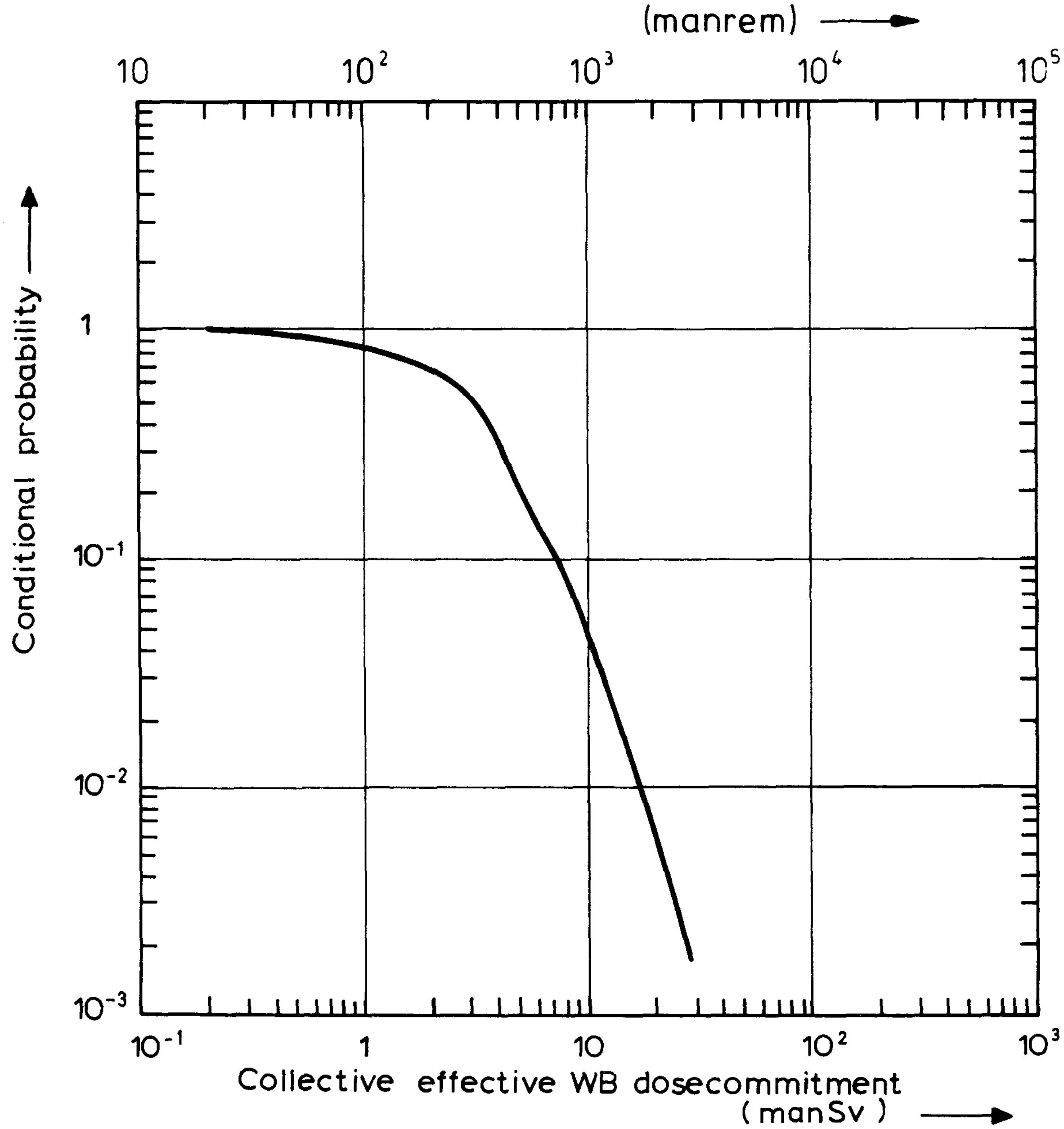


Figure 4.6 "Risk" curve. Bitumen fire after traffic accident, in low population area.

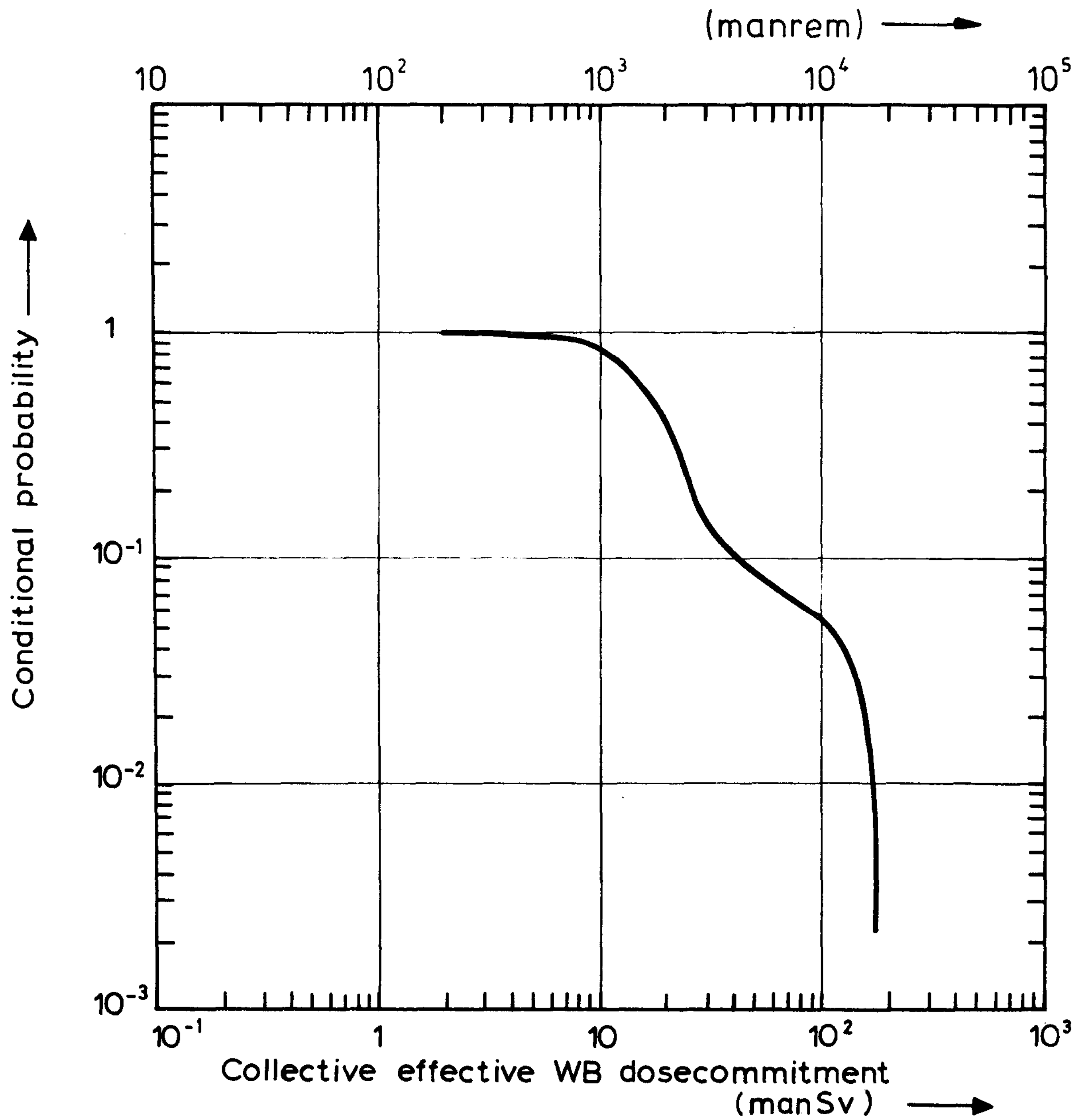


Figure 4.7 "Risk" curve. Bitumen fire after traffic accident, in city.

4.6 SHIP WRECKAGE

4.6.1 THE ALMA-STUDY

The wreckage of a ship carrying radioactive waste may results in a loss at sea of anything from a single waste package up to the total payload.

This study has made no evaluation of its own of the risks related to sea transportation of radioactive waste, as a comprehensive study of that subject was performed within the ALMA-study. The doses given in the following are based on the same scenarios as used in the ALMA-study, but the doses have been recalculated to match with the reference waste, which is slightly different from the ALMA-waste with respect to nuclide inventory.

The documentation of the ALMA-study is given in a number of PRAV reports, /26, 27, 28/ and, in particular, /29/.

4.6.2 BASIC ACCIDENT ASSUMPTIONS

Of the many accident scenarios considered in the ALMA-study, the following two were selected for presentation here:

- one concrete mould is lost and left undamaged on the sea floor. 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 released at a steady rate over 6 months (a pessimistic and rather unrealistic assumption).

It is furthermore assumed that:

- the concrete mould contains 5 year old waste of reference composition (Technical Part I), which is:

Cs-137:	75 GBq	(2.0 Ci)
Cs-134:	17 GBq	(0.46 Ci)
Sr-90:	3.8 GBq	(0.10 Ci)

  Co is known to give a smaller dose than any of the above 3 nuclides, and is not considered in the dose calculations.

- the accident take place in the Baltic Sea about 5 km of the Southern part of the Swedish East-coast.

4.6.3 RESULTS OF CALCULATIONS

The calculated doses resulting from release from one concrete mould, resting undamaged on the sea floor are given in table 4.2. The calculated doses resulting from release from one severely damaged concrete mould resting on the sea floor are given in table 4.3. It should be noticed that these calculations assume total damage of the transport containers. Undamaged containers will effectively prevent release from the waste packages for a very long period, permitting recovery of the waste.

The calculated doses resulting from the accident refer, as specified, to one package. Because of the special design of the waste transport ship, the rigid transport containers and the strong concrete moulds, a loss and direct exposure to sea water of more than a few packages seems extremely unlikely, even if the transport ship was involved in a violent collision with a high speed passenger ship.

Table 4.2 Calculated individual dose commitment and collective dose commitment as a result of release into the sea by diffusion from one undamaged concrete mould. (Based on /29/).

Nuclide	Initial inventory GBq	Total release GBq	Time for maximum release; years after accident	Max. individual dose commitment * µSv	Collective dose commitment manSv
Sr-90	3.7	0.067	45	0.0040	0.00019
Cs-134	17.0	0.16	4	0.013	0.000039
Cs-137	74.0	25.0	10	0.72	0.024

* Integrated over 50 years

Note: 1 Sv = 100 rem

1 Ci = $3.7 \cdot 10^{10}$ Bq = 37 GBq

Table 4.3 Individual dose commitment and collective dose commitment as a result of continuous release of the content of one concrete mould into the sea over 6 months. (Based on /29/).

Nuclide	Total release GBq	Maximum individual dose commitment* µSv	Collective dose commitment manSv
Sr-90	3.7	0.63	0.01
Cs-134	17.0	1.5	0.0038
Cs-137	74.0	4.0	0.07

* Integrated over 50 years

Note: 1 Sv = 100 rem

1 Ci = $3.7 \cdot 10^{10}$ Bq = 37 GBq

4.6.4 PROBABILITY OF ACCIDENT

The probability of different types of events during sea transportation is evaluated in /29/ and summarized in table 4.4.

Table 4.4 Events arising from collisions, their probabilities and mechanical consequences. All probabilities are based on time at sea carrying radioactive cargo.

Event	Estimated probability per year	Mechanical consequences
Collision	$< 10^{-3}$	Not necessarily any consequences.
Relatively serious collision	$< 10^{-4}$	Relatively serious damage to the vessel. Cargo not necessarily damaged.
Total loss of vessel	$< 10^{-5}$	The vessel either irrecoverable or repair not profitable.
Vessel foundered	$< 10^{-5}$	Steel containers start to leak. Vessel recovered within 6 months.
Vessel foundered at a depth greater than 80 m	$< 10^{-6}$	Concrete containers start to leak. Vessel recovered within 6 months.
Concrete blocks lost at sea	$< 10^{-6}$	Blocks seriously damaged when hitting the bottom if hard. Leaching of the waste fairly fast.
Concrete block in steel container damaged by collision	$< 10^{-7}$	Blocks not recovered. Straight through crack in blocks. Superficial cracks in the other blocks.
Concrete block in concrete container damaged by collision	$\leq 10^{-7}$	Same as above.
Loss of concrete containers with concrete blocks	$< 10^{-6}$	Containers seriously damaged if hitting hard bottom, but remain a unit which can be lifted within 6 months. Concrete blocks inside the containers cracked and leaching resistance decreases.
Fire in bitumen drums	Inconceivable	

(Ref./29/, Table 4.1).

4.7 BITUMEN FIRE ABOARD SHIP

No specific analysis of a bitumen fire aboard a ship was analysed within the Nordic project, as this type of accident was thoroughly dealt with in the ALMA-project. The discussion in this section is primarily based on /29/.

As discussed in Technical Part I, all drums with bituminized waste will be transported in concrete shielded containers. The containers have very good fire resistant 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. Secondly, the oxygen content in the tight containers will allow only about 0.1 % of the bitumen in the container to burn.

The only way to ignite the bitumen would be a very serious collision which damages the container severely, followed by a violent fire. Such a fire is only credible in connection with a collision with a tanker carrying crude oil or gas. Such an accident is deemed to be inconceivable under the prevailing conditions. The doses calculated for a bitumen fire clearly show that a bitumen fire aboard a ship would mean no severe radiation doses to the public.

5 DISPOSAL, DESCRIPTION OF CALCULATIONS

5.1 GENERAL DESCRIPTION OF CALCULATIONS PERFORMED

Various combinations of the following concepts, parameters and specific data have been examined in preliminary and/or more elaborate calculations /41, 42, 43, 44, 45, 46, 47/.

- Disposal concepts: Shallow land burial
Concrete bunker
Rock cavern

- Geology: Sandy till
Clayey till
Crystalline rock

- Waste type: Bitumen drum
Concrete drum
Concrete mould

- Nuclides: C-14, Co-60, Ni-63
Sr-90, Tc-99, I-129
Cs-135, Cs-137

- Events:
  1. Normal release from the repository to a well outside the site area, and to the sea.
  2. Release from fractured repository to a well outside the site area and to the sea.
  3. Dwelling on the repository area.
  4. Release from the repository to a well close to the repository.
  5. Release from a fractured repository to a well close to the repository.

6 Excavation at the site of the repository.

7 Farming around the site of the repository.

- Release parameters: Diffusion coefficient, variation of numerical values

- Geological and hydrological parameters: Hydraulic conductivity, kinetic and diffusion porosity, retention factor, variation of numerical values

5.2 ADMINISTRATION AND RELEASE SCENARIOS

There are different periods in the history of a disposal site. In the present waste management system these time periods have been defined as follows, and as indicated in figure 5.1.

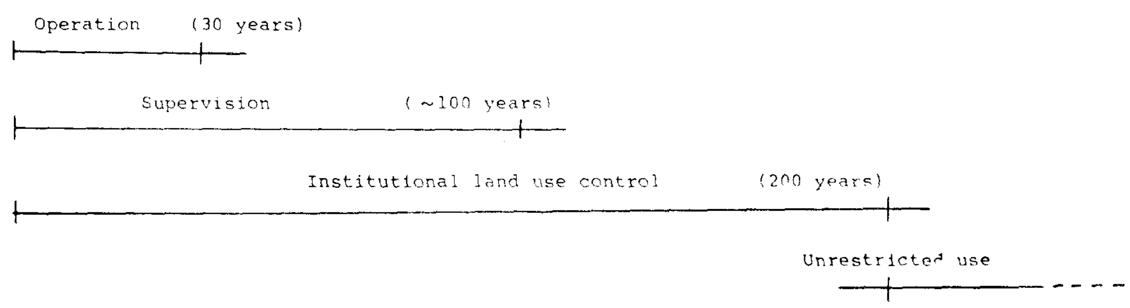


Figure 5.1 Definitions of the periods for a disposal site

5.2.1 PERIOD OF OPERATION

During this period waste is supplied to the storage, and this period will probably be followed by an observation period before the storage is finally closed.

Normal events:

- normal waste handling
- handling of broken packages

Abnormal events:

- dropped package
- fire in disposal area

All these events will be more or less the same as those described for storage and transportation. The environmental conditions will probably be more favorable than those for storage and transportation, or at least as good, since the disposal site will probably be chosen either near a reactor site, or in a remote area. Accordingly no separate calculations for this period have been performed.

5.2.2 PERIOD OF SUPERVISION

During this period an active control and surveillance is maintained. Unauthorized access to the fenced-off area is not allowed. The first part of this period comprises the operational period. After the repository has been closed, the surveillance may reveal any abnormal behaviour of the storage, and remedial actions can be taken, if necessary.

When the repository is closed, the groundwater will return to its natural level. If the repository is situated below this level, it will slowly be filled with water. In the calculation it is conservatively assumed that the diffusion of radioactive nuclides starts immediately (meaning that the nuclides are assumed to be completely dissolved as soon as the water fills the repository), later followed by release from the repository and migration in the ground.

If no abnormal behaviour occurs, the influence of the site on the environment will not appear until after an extensive time period.

5.2.3 PERIOD OF INSTITUTIONAL LAND USE CONTROL

It is assumed that during the period of institutional land use control no human activity is allowed that involves any intrusion into the ground. The interesting exposure pathway for this period is drinking water from a well hypothetically assumed to be located just outside the controlled area. It is assumed that there is a distance of at least 100 meters from the repository to the well (or to the border of the repository area). This is clearly a pessimistic assumption. If the controlled area is adjacent to a lake or the sea, a well would not be placed on that side of the repository, while ground water movement would as a rule be in that direction. Such sites are not difficult to find in the Nordic countries, making the scenario unlikely. This case is however used as a reference case in the study.

After the period of supervision is completed, it is assumed that building of industrial or residential areas would only take place if careful evaluations showed the hazards to be acceptable for the situation in question. It is however not foreseen that there will be restrictions on recreational use of the area. There would accordingly be a theoretical possibility of a person camping on top of the repository during a whole summer vacation; this would probably be the worst case one could imagine. Simple calculations performed in the study show that the dose would still be acceptable with a very wide margin.

As an additional assurance that administrative control would be effective, passive marking of the area could be used. For example marking could be done by placing an inscription on stone or metal above the repository. Such marking would significantly reduce the probability of instrusion by mistake.

5.2.4 PERIOD OF UNRESTRICTED USE

When the period of institutional control is completed, there is free access to the area, and a number of man-initiated activities or events could take place. Various natural events may also damage the repository, but in the Nordic countries it is possible to find sites where the importance of natural events will be small compared to the importance of man-caused events /48/. Scenarios analysed are described in the following. In the study it is assumed that the period of unrestricted use starts after 200 years.

When there is free access to the area, a person or persons may settle on top of the repository. Dwelling on top of the repository is one of the cases analysed in the present study.

Drilling for water is another reference case. It is assumed that drilling right into the repository will not take place, as a person hitting a concrete structure while drilling, will probably try again in another place. It is assumed however that a well may be drilled directly outside of the repository. For shallow land burial and concrete bunker there will not be enough inflow of water to a well to supply even one family with water. A family needs roughly 300 cubicmeters per year, and a higher value is incompatible with the values used for the hydrological parameters of soil, used in the calculations. In clayey till wells are not drilled, and in sandy till the capacity of a well will at 10 meters below undisturbed ground water level be only about 30 cubicmeters per year /49/. In crystalline rock the capacity might be sufficient. Accordingly use of water from a well drilled just outside the repository, in crystalline rock, is used as a reference case.

Construction work on top of the repository might take place, and have such a character that the shielding soil layer on top of the repository might be removed or reduced. Such activity, also referred to as intrusion, is analysed as a reference case.

Farming could be a permanent human acitivity at the site when it is left for unrestricted use. That would not involve any hazard unless some of the contaminated earth is digged up and spread over the field. One could select an area that is not suitable for farming and with unfertile deeper layers of ground. Then the probability that the soil could be used for agriculture would be very low.

For a reference case in this study it is however assumed that an agricultural area is contaminated.

5.3 CALCULATIONS AND RESULTS

After a number of years (30 years in this study) in the operational state, the repository is sealed. When the repository is filled with water, it is impossible to keep some of the very longlived nuclides permanently confined inside the barriers. Even if the repository has very low hydraulic conductivity, there will be diffusion out of the repository, though the rate of diffusion is low for most nuclides.

Calculation of release and dispersion has been carried out in three consequtive steps:

- Migration of nuclides out of repository
- Migration of nuclides in ground and their dilution in ground water
- Transport of nuclides in the biosphere resulting in doses to man

It must be pointed out that the simplified model used for calculating the diffusion of nuclides from the repository requires that one assumes that the nuclides are homogeneously distributed over the whole repository volume. For a rock cavern and a concrete bunker this is the volume inside the walls; and for shallow land burial it is the volume inside each drum or mould. This volume is in the following referred to as the matrix.

During this study many calculations have been performed, using various combinations of parameters. In this way certain cases have been found to give higher potential doses than others.

It has been decided to present as numerical results mainly the results obtained from calculations on steel drums filled with cementized waste. The reason is that with the calculation models used this type of package gives the highest doses.

As a result of the simplifications in the calculation models, the cases of concrete mould and concrete drums give the same release rate from the repository, since the repository volume is in both cases assumed to be one compact concrete block with homogeneous distribution of the waste. This is true for both concrete bunker and rock cavern. For shallow land burial however each mould and drum is regarded separately, and as a result the drums will have a higher surface/volume ratio, giving higher release rate.

For the dose-dominating nuclides the diffusion coefficients for bituminized waste can be assumed to be 100 times smaller than for cementized waste. As the release rate is proportional to the square root of the diffusion coefficient, the release rate will be about 10 times smaller for this type of waste.

Four nuclides have been selected for analysis of release from the repository: C-14, I-129, Cs-135, Cs-137. Their properties are presented in table 5.1. Apart from having various properties of interest to this study, they give the dominating dose contribution with the assumptions used in the calculations. Some other nuclides could have been included in the study. One of these is Tc-99, which behaves like C-14. The amount of this nuclide in waste is however much smaller. Ni-63 and Sr-90 may be compared to Cs-137, as they are also relatively short-lived, but again the content of these nuclides in the waste is much smaller, and so are the diffusion coefficients.

Table 5.1 Nuclide properties

Nuclide	Properties
I-129	A very long-lived nuclide, rapidly migrating
C-14	A long-lived nuclide, rapidly migrating
Cs-135	A very long-lived nuclide, slowly migrating
Cs-137	A nuclide of medium half-life, slowly migrating

In scenarios involving generation of dust in the air, and inhalation, Sr-90 has however been included, because the dose conversion factor for this nuclide is higher than for Cs-137.

The content of Pu-239 in waste from reactor operation is low. The diffusion and migration rate is also low. For this nuclide, as for Sr-90, inhalation is the critical exposure pathway. Early in this study a calculation was performed, involving inhalation of dust resulting from plasting of the repository itself. The results showed that the inhalation dose from Pu-239 was much lower than the one from Sr-90, and comparable to the inhalation dose from C-14.

5.3.1 WATER FROM WELL NEAR INTACT REPOSITORY

It is assumed that there is a well 100 meters from the repository, and that the water after consumption runs into the sea, which in the present calculations is the Baltic Sea. This case refers to the period of supervision.

The release of radionuclides from the intact repository is assumed in the calculations to be governed by diffusion, according to equation (1), chapter 6. An important parameter in these calculations is the diffusion coefficient, and the numerical values employed are presented in table 5.2.

As an example all parameter values used in the case of release from a concrete bunker containing concrete drums, are given in table 5.3.

In the case of shallow land burial, the individual drums or moulds are assumed to leak out directly into the soil refilled into the trench. This soil is expected to be more porous than the undisturbed soil, and rain water will easily seep through the soil in the trench, carrying the nuclides that have leaked out down to ground water level.

Within the repository the mean migration distance is assumed to be one meter (of soil). The vertical velocity of rain water in the trench is assumed to be 5 meters per year. This velocity is usually called the percolation rate. It must be pointed out that it has not been possible to account for the delaying action of the steel drums, which may be considerable.

Table 5.2 Diffusion coefficients

Nuclide	Diffusion coefficient (m^2/a)		
	Concrete	Bitumen	Clay
Carbon	$3 \cdot 10^{-4}$	$2 \cdot 10^{-6}$	$3 \cdot 10^{-2}$
Iodine	$6 \cdot 10^{-4}$	$2 \cdot 10^{-6}$	$6 \cdot 10^{-2}$
Cesium	$3 \cdot 10^{-4}$	$1 \cdot 10^{-6}$	$2 \cdot 10^{-4}$

Table 5.3 Parameter values, concrete bunker

	C-14	I-129	Cs-135	Cs-137
a_o = contents of nuclide at time zero (GBq)	558	0.54	0.9	$1.8 \cdot 10^5$
S = surface area of the repository (m^2)	7200			
V = volume of the repository (m^3)	12000			
D'_o = diffusion coefficient for the nuclide in the matrix (m^2/a)	$3 \cdot 10^{-4}$	$6 \cdot 10^{-4}$	$3 \cdot 10^{-4}$	$3 \cdot 10^{-4}$
X_1 = the thickness of the wall (m)	0,5			
D'_1 = diffusion coefficient for the nuclide in the wall (m^2/a)	$3 \cdot 10^{-4}$	$6 \cdot 10^{-4}$	$3 \cdot 10^{-4}$	$3 \cdot 10^{-4}$
λ = decay constant for the nuclide (1/a)	$1 \cdot 10^{-4}$	$4 \cdot 10^{-8}$	$2 \cdot 10^{-7}$	$2 \cdot 10^{-2}$

In table 5.4 the calculated maximum release rates from the repositories are given, and so is the time at which this maximum occurs.

Table 5.4 The release from the repository

Repository	Maximum release rate (MBq) and time (a)			
	C-14	I-129	Cs-135	Cs-137
Shallow land burial	$5.2 \cdot 10^4$ (1)	7.4 (1)	81 (80)	$2.5 \cdot 10^6$ (80)
Concrete bunker	93 (400)	0.19 (200)	0.16 (400)	$1.3 \cdot 10^3$ (85)
Rock cavern	52 (400)	0.11 (500)	$3.3 \cdot 10^{-2}$ (300)	$8.9 \cdot 10^{-4}$ (320)

The model used for calculating diffusion is one-dimensional, and accordingly equation (1), chapter 6, can be used. For most cases with homogeneous barriers this model is satisfactory. But for a rock cavern, in the case where there is rock with discrete cracks surrounded by clay, a two-dimensional model is needed to evaluate the effect of the concentration of the diffusion field in front of the cracks. In reference /50/ such a calculation is performed, using a computer program. The two-dimensional model gives a much slower transport than would have been calculated using equation (1) in chapter 6. If this equation is used with data from the ALMA study, a comparison between one- and two-dimensional models can be made, as shown in table 5.5. The ALMA study has a clay barrier of 1.5 meters, which gives a maximum much later than for the present study.

Table 5.5 Comparison

Nuclide	Time for maximum leakage (a)		Release ratio
	One dimensional	Two dimensional	(One dim./two dim.)
C-14	1000	2000	5
I-129	600	1000	2
Cs-135	10000	15000	2
Cs-137	500	800	1000

The nuclides released from the repository will migrate with the ground water flowing past the repository, and might eventually reach a well outside the site area, assumed to be 100 meters from the repository. The migration time is calculated with a simple model and, for some of the parameters, using very simplified data. The parameter values used in the calculations are shown in table 5.6. The retention factor is calculated using equation (10), and for rock using equation (11), both in chapter 6. It should be pointed out at this point that the value used for the distribution coefficient for carbon and iodine assumes that all carbon and iodine is dissolved all the time during migration. For a large number of possible carbon and iodine compounds this is not true, and for these compounds the calculated migration rates and radiation doses may be grossly overestimated.

Table 5.6 Parameters for calculation of migration

	Sandy till	Clayey till	Rock
K_p = hydraulic conductivity m/s	$1 \cdot 10^{-7}$	10^{-9}	10^{-8}
i = hydraulic gradient m/m	$5 \cdot 10^{-2}$	$5 \cdot 10^{-2}$	$2 \cdot 10^{-2}$
ε_k = kinetic porosity	$3 \cdot 10^{-2}$	$1 \cdot 10^{-2}$	$3 \cdot 10^{-3}$
ε_d = diffusion porosity	0.4	0.2	-
ρ_{TP} = dry particle density of soil (m^3/kg)	2,650	2,650	-
α_2 = geometric surface area of rock (m^2/kg)			30
s = fracture spacing (m) in rock			0.4
K_d = distribution coefficient (m^3/kg)			
- carbon	0	0	0
- iodine	0	0	0
- cesium	0.1	0.5	0.1
K_i = retention factor			
- carbon	1	1	0
- iodine	1	1	0
- cesium	400	5,300	420

Table 5.7 shows transport or migration times for ground water through the geological formations considered. These transport times are calculated disregarding the disturbing effect on ground water movement of the presence of the well. As can be seen from these values transport in clayey till is very much slower than in sandy till are presented, but for clayey till all doses will accordingly be much lower than the ones calculated for sandy till. The dose from Cs-137 will in all cases be negligible, because of the relatively short half-life and high retention factor of this nuclide.

Table 5.7 Ground water migration, 100 meters

	Velocity (m/a)	Migration time (a)
Sandy till	5.2	19
Clayey till	0.16	634
Rock	2.1	48

The maximum inflow to the well over a one-year period is shown in table 5.8.

Table 5.8 Inflow of activity into the well 100 m below the repository

Repository	Maximum inflow (MBq/a) and time (a) to reach maximum			
	C-14	I-129	Cs-135	Cs-137
Shallow land burial	$5.2\cdot10^4$ (25)	74 (25)	81 $(1\cdot10^4)$	0
Concrete bunker	$9.3\cdot10^7$ (420)	$1.9\cdot10^5$ (220)	$1.5\cdot10^5$ $(1\cdot10^4)$	0
Rock cavern	$5.2\cdot10^7$ (450)	$1.1\cdot10^5$ (250)	$3.3\cdot10^4$ $(2.3\cdot10^4$	0

It is assumed that all nuclides are collected and diluted in the well water (300 cubic meters), and that one person consumes 0.44 cubic meters per year of this water. The dose is then calculated using equation (12) in chapter 6. The dose conversion factors used are listed in table 5.9. These factors give the dose resulting from the intake of one unit if radioactive material. In table 5.10 the results of the calculations are given.

Table 5.9 Dose conversion factors for intake with water and food

Nuclide	Dose factor (Sv/Bq)
C-14	$2.6 \cdot 10^{-10}$
Sr-90	$3.2 \cdot 10^{-8}$
I-129	$1.0 \cdot 10^{-7}$
Cs-135	$2.0 \cdot 10^{-9}$
Cs-137	$1.4 \cdot 10^{-8}$

Table 5.10 Individual doses from drinking water (intact repository)

	Dose (Sv/a) and time (a) for maximum			
	C-14	I-129	Cs-135	Cs-137
Shallow land burial	$2 \cdot 10^{-2}$ (25)	$7.9 \cdot 10^{-3}$ (25)	$3.3 \cdot 10^{-4}$ $(1 \cdot 10^{4})$	0
Concrete bunker	$3.6 \cdot 10^{-5}$ (420)	$2.6 \cdot 10^{-5}$ (220)	$4.3 \cdot 10^{-7}$ $(1 \cdot 10^{4})$	0
Rock cavern	$2.0 \cdot 10^{-5}$ (450)	$1.4 \cdot 10^{-5}$ (350)	$9.5 \cdot 10^{-8}$ $(2.3 \cdot 10^{4})$	0

The doses in table 5.10 from Cs-137 have been compared to doses obtained when using the computer program DIFMIG (described in chapter 6). This program is in principle based upon the same equation as that used to get the results in table 5.4 and 5.8, but DIFMIG can in addition take into consideration the effect of time-dependent changes in concentrations caused by factors other than diffusion. The two programs in this case produced practically identical results.

After the water from the well has been used, it will go with waste water to some water recipient; in the present calculations the Baltic Sea. Also it might happen that instead of reaching a well, ground water might migrate directly into the sea. Doses resulting from release to the sea are calculated using, the computer program BIOPATH with data meant to represent the Baltic Sea. The results of these calculations are given in table 5.11. These results are also annual doses, though collective. In this case there is however a build-up of radioactive materials in the Baltic Sea, and the calculation is performed for the last of 500 consequtive years in which the release to the sea each year has been the calculated maximum annual release. The doses are effective committed doses, and are caused by the collective intake during "year 500".

Table 5.11 Collective doses for release to the sea

Repository	Dose (manSv/a)			
	C-14	I-129	Cs-135	Cs-137
Shallow land burial	$3 \cdot 10^{-1}$	$1 \cdot 10^{-2}$	$2 \cdot 10^{-6}$	0
Concrete bunker	$6 \cdot 10^{-3}$	$4 \cdot 10^{-4}$	$5 \cdot 10^{-8}$	0
Rock cavern	$3 \cdot 10^{-3}$	$2 \cdot 10^{-4}$	$1 \cdot 10^{-8}$	0

5.3.2 WATER FROM WELL NEAR FRACTURED REPOSITORY

Like in 5.3.1 it is assumed that there is a well 100 meters from the repository. As a way of judging the consequences from all imaginable types of damage to the repository, the release from a repository with a crack traversing both the waste and the concrete walls has been calculated. The way in which such a crack might occur has not been specified, and it may be caused by natural as well as man-caused events. Consequently there is a wide range in the probability of a crack occurring, depending upon the origin.

Release rate and nuclide migration have been calculated using the same models as in the calculations for an intact repository. For shallow land burial the case does of course not apply.

For the concrete bunker the crack is supposed to expose a fracture area twice the cross-section of the repository (2x100 squaremeters). This area is small compared to the total surface area of the bunker (7,200 squaremeters). In the calculations it is assumed, as previously, that the radioavtive nuclides initially are homogeneously distributed throughout the whole bunker volume. As time proceeds, diffusion will result in a reduction of concentration of radioactive nuclides near the surfaces through which release takes place, while the concentration of long-lived nuclides will be practically unchanged in the parts of the bunker volume far from such surfaces. The consequence of this is that the release from the crack surfaces will be practically the same whether the crack occurs right after the repository has been sealed, or several hundred years later. The release through the intact bunker surfaces will however change with time, and because of the delaying effect of the surrounding concrete wall, the maximum release rate will occur after 200-400 years. If the crack also occurs at this time, the total release will be higher than if the crack occurred at any other time.

For a rock cavern it is assumed that even if the repository is fractured, the surrounding clay still represents an unbroken barrier, as wet clay is a plastic substance. It is implicit in the other assumptions for the scenario that the clay is wet. Calculations using the one-dimensional model indicate that the crack will result in a slight increase in release rate. The individual doses calculated using this model are given in table 5.12. For this case the one-dimensional model is actually rather unsuitable (this which is also true for the bunker case), as the area through which the release from the crack must pass into the clay is quite small. The increase in doses caused by the presence of the crack would actually be much smaller than indicated by the values in the table. The increase in collective doses would also be quite insignificant.

Table 5.12 Individual doses from drinking water (fractured repository)

Repository	Dose (Sv/a) and time (a) for maximum			
	C-14	I-129	Cs-135	Cs-137
Concrete bunker	$3.6 \cdot 10^{-5}$ (20)	$2.6 \cdot 10^{-5}$ (20)	$4.2 \cdot 10^{-7}$ $(1 \cdot 10^{4})$	0
Rock caveren	$2.8 \cdot 10^{-5}$ (80)	$2.3 \cdot 10^{-5}$ (58)	$1.3 \cdot 10^{-7}$ $(2.3 \cdot 10^{4})$	0

5.3.3 DWELLING ON TOP OF THE REPOSITORY

Only shallow land burial could possibly give external radiation exposure to a person staying on top of the repository. As mentioned in chapter 5.2.3 even camping on top of the repository over extensive periods would only result in negligible doses. After commencement of the period of unrestricted use, however, a dwelling might be erected on top of the repository.

After about 100 years the gamma radiation from the repository is dominated by Cs-137. The dose rate directly over the repository is calculated using standard shielding calculation methods, assuming an earth cover of 2 meters, and the dose rate is found to be about 10^{-13} Sv/h (10^{-8} mrem/h). This corresponds, assuming continuous exposure of the person, to 10^{-9} Sv/a (10^{-4} mrem/a) or $5x10^{-8}$ Sv ($5x10^{-3}$ mrem) over a lifetime.

5.3.4 WATER FROM WELL DIRECTLY OUTSIDE INTACT REPOSITORY

When the area is left for unrestricted use, a well might be drilled directly adjoining the repository. It is assumed that this may take place at the earliest 200 years after the repository was sealed. The doses are calculated using the same models as when the well is at a 100 meter distance.

As mentioned in section 5.2.4 it is only in crystalline rock that a well may supply sufficient water for the needs of one family, if the hydrological parameter values used for sandy and clayed till are valid. In spite of this it is however assumed that a well in sandy till is drilled, in order to supply drinking water only. The water volume will of course be smaller in this case, and the concentration of radioactive nuclides correspondingly larger.

Another effect which may be important when a well is drilled right beside a repository, is the draining effect of the well itself. The well might lower the ground water level sufficiently to leave the repository partially or completely dry. In that case the release from the repository will be reduced and possibly eliminated. This effect will usually be more pronounced in the case of concrete bunker than shallow land burial. In the latter case surface water on its way from earth surface down to ground water level will run directly past the waste, while in the former case it will not enter the concrete bunker at all.

In table 5.13 are given individual doses calculated on the basis of the well supplying only drinking water, and disregarding the dose-reducing effects mentioned in the above. The values should accordingly be regarded as upper limits, and would probably never be reached in reality.

From the table it is seen that Cs-137 gives by far the highest dose for shallow land burial and concrete bunker. It should be remembered, however, that the half-life of this nuclide is only about 30 years, and that more realistic calculation assumptions easily may reduce the dose from this nuclide to an insignificant level.

Table 5.13 Individual doses from drinking water (intact repository)

Repository	Dose (Sv/a) and time (a) for maximum			
	C-14	I-129	Cs-135	Cs-137
Shallow land burial	$1.3 \cdot 10^{-2}$ (200)	$6.9 \cdot 10^{-3}$ (200)	$1.7 \cdot 10^{-4}$ (200)	$4 \cdot 10^{-1}$ (200)
Concrete bunker	$3.6 \cdot 10^{-4}$ (400)	$2.6 \cdot 10^{-4}$ (200)	$4.5 \cdot 10^{-6}$ (400)	$5.6 \cdot 10^{-2}$ (200)
Rock cavern	$2.0 \cdot 10^{-5}$ (400)	$1.4 \cdot 10^{-5}$ (200)	$9.5 \cdot 10^{-8}$ (3000)	$1.9 \cdot 10^{-8}$ (320)

5.3.5 WATER FROM WELL DIRECTLY OUTSIDE FRACTURED REPOSITORY

Most of the discussion in section 5.3.2 applies to this case also, and this is also true for much of the discussion in section 5.3.4. The main difference is that the earliest data when the fracture may take place is 200 years after closure of the repository. This is near the time when the release rate from the repository is at a maximum. It is found that compared with an intact repository the release rate from the fractured repository may maximum be doubled.

Calculated individual doses, assuming fracture at 200 years, are presented in table 5.14. Like in all the chapters concerning water from well, the simplifications one has been forced to employ lead to conservative results.

Table 5.14 Individual doses from drinking water (fractured repository)

Repository	Dose (Sv/a) and time (a) for maximum			
	C-14	I-129	Cs-135	Cs-137
Concrete bunker	$5.4\cdot10^{-4}$ (400)	$4.5\cdot10^{-4}$ (200)	$8.2\cdot10^{-6}$ (400)	$1.5\cdot10^{-1}$ (200)
Rock cavern	$3.0\cdot10^{-5}$ (400)	$2.4\cdot10^{-5}$ (300)	$1.3\cdot10^{-7}$ (3000)	$4.6\cdot10^{-6}$ (250)

5.3.6 INTRUSION BY EXCAVATION

When the area is left for unrestricted use, exacavations for some reasons or other might take place. This may involve radiation exposure via inhalation of dust containing radioactive materials and by external exposure.

In the external exposure scenario it is assumed that the radioactive materials are in the repository, but that the earth cover may be reduced or removed. The doses will decrease with time, as radioactive decay is the only process of importance in connection with this scenario, and the highest doses will accordingly be encountered at 200 years after closure of the repository, i.e. the time at which the area is assumed to be left for unrestricted use. At this time Cs-137 is still the dominating gammaemitter, and is the nuclide for which the calculations have been performed.

Calculations using standard radiation shielding methods have been performed for earth cover varying from 2 meters to zero, and the results are shown in figure 5.2. If the earth cover is removed completely, the doses might become unacceptable. In order to keep the dose for shallow land burial about equal to the natural background radiation (roughly 10^{-7} Sv/h), the earth cover should not be much less than 1 meter. If a single waste package is removed from the repository, the dose rate at 1 meter's distance is found to be about 10^{-6} Sv/h (10^{-4} rem/h). This is valid for concrete drums. In this case the bitumen drums will give higher doses, because of the higher content of nuclides in the drums. But the dose will only be approximately a factor two higher.

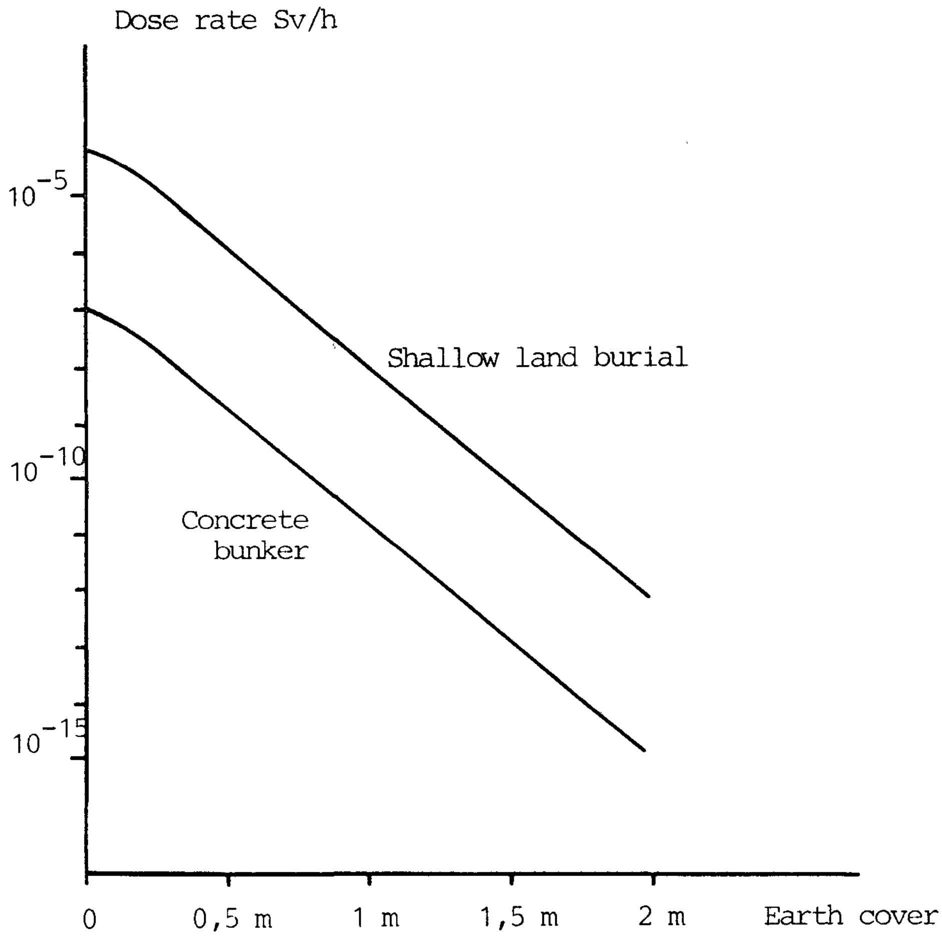


Figure 5.2 The external gamma dose rate as function of the earth cover over the repository, time = 200 years.

There are two inhalation scenarios; blasting and digging. In these scenarios only the leakages from the repositories are assumed to be involved, not the radioactive materials still in the repositories.

Blasting is supposed to give exposure only for a short time (assumed to be 1 hour in the calculations). Digging is supposed to give exposure over a longer time (20 hours), but the dust concentration in air will be lower. The following values for dust concentration are judged to be representative:

- concentration 10^{-3} g/m^3 for blasting
- concentration 5 10^{-4} g/m^3 for digging

In addition the breathing rate is assumed to be 0.91 cubicmeters per hour, and the dose rate conversation factors (dose per unit intake of radioactive nuclide) are listed in table 5.15. The concentration of radioactive nuclides in the soil are calculated using equations given in chapter 6.

The calculated doses are given in tables 5.16 and 5.17, and they are quite low compared to the ones from direct exposure when the earth cover over the repository is reduced or removed.

Table 5.15 Dose conversion factors for inhalation

Nuclide	Dose factor Sv/Bq
C-14	$1.8 \cdot 10^{-10}$
Sr-90	$5.7 \cdot 10^{-8}$
I-129	$6.2 \cdot 10^{-8}$
Cs-137	$8.9 \cdot 10^{-9}$

Table 5.16 Doses from inhalation, blasting

Nuclide	Doses (Sv)	
	Shallow land burial	Concrete bunker
C-14	$3.8 \cdot 10^{-13}$	$4.4 \cdot 10^{-15}$
Sr-90	$4.2 \cdot 10^{-9}$	0
I-129	$2.0 \cdot 10^{-13}$	$5.3 \cdot 10^{-15}$
Cs-137	$5.2 \cdot 10^{-7}$	$6.0 \cdot 10^{-9}$

Table 5.17 Doses from inhalation, digging

Nuclide	Doses (Sv)	
	Shallow land burial	Concrete bunker
C-14	$3.8 \cdot 10^{-12}$	$4.4 \cdot 10^{-14}$
Sr-90	$4.2 \cdot 10^{-8}$	0
I-129	$2.0 \cdot 10^{-12}$	$5.3 \cdot 10^{-14}$
Cs-137	$5.2 \cdot 10^{-6}$	$6.0 \cdot 10^{-8}$

5.3.7 FARMING

When the area is left for unrestricted use, it may also be used for agricultural purposes. There is then a possibility of exposure via inhalation of dust suspended in the air and via agricultural products. In this scenario only materials that have leaked out of the repository are assumed to be involved. The doses will be largest at 200 years. At 200 years the various nuclides have migrated a certain distance. It is assumed that the total amount of the nuclide outside of the repository is homogeneously distributed in an earth volume of length equal to the migration distance, width equal to the width of the repository, and depth equal to the distance from earth surface to the level of the lowest part of the repository. In connection with this scenario, the rock cavern is irrelevant.

The earth volumes that are assumed to be contaminated are given in table 5.18, and the corresponding concentrations of the various nuclides are given in table 5.19.

Table 5.18 Contaminated earth volume

Nuclide	Contaminated earth volume (m^3)	
	Shallow land burial	Concrete bunker
C-14	$5.4 \cdot 10^5$	$4.4 \cdot 10^5$
Sr-90	$1.2 \cdot 10^4$	0
Cs-137	$6.0 \cdot 10^3$	$4.2 \cdot 10^3$

Table 5.19 Concentration in ground

Nuclide	Concentration in ground (Bq/m^3)	
	Shallow land burial	Concrete bunker
C-14	$2.4 \cdot 10^5$	$1.7 \cdot 10^4$
Sr-90	$2.2 \cdot 10^5$	0
Cs-137	$6.8 \cdot 10^7$	$5.6 \cdot 10^6$

It has been assumed in the calculations that the dust concentration is 10^{-4} grams per cubicmeter for this scenario. Furthermore it has been assumed that only 1/10 of the total field working time of 500 hours per year is spent in the area which is contaminated, since this area will not be large. For calculation of exposure via agricultural products, only the vegetable pathway has been included, and it is assumed that the annual intake of vegetables is 28 kilos. Though this value is not identical to the value for the same parameter used in the calculation of doses from bitumen fire, it is not very different. The fact that the other nutrition pathways are not included may have lead to a certain underestimation of the dose, but probably not by more than a factor two. This underestimation is probably more than compensated by overestimations e.g. of leakage from the repository.

The doses calculated are given in tables 5.20 and 5.21. The table giving the doses via vegetables also contain the uptake factor used in the calculations. The uptake factor is the ratio between the concentration per weight unit of the nuclide in the vegetables (wet weight) and soil /51.23/.

Table 5.20 Inhalation doses from farming

Nuclide	Doses (Sv/a)	
	Shallow land burial	Concrete bunker
C-14	$1.3 \cdot 10^{-13}$	$9.2 \cdot 10^{-15}$
Sr-90	$3.8 \cdot 10^{-11}$	0
Cs-137	$1.8 \cdot 10^{-9}$	$1.5 \cdot 10^{-10}$

Table 5.21 Doses from intake of vegetables

Nuclide	Uptake factor	Doses (Sv/a)	
		Shallow land burial	Concrete bunker
C-14	5.5	$6.4 \cdot 10^{-6}$	$4.5 \cdot 10^{-7}$
Sr-90	$8 \cdot 10^{-2}$	$1.1 \cdot 10^{-5}$	0
Cs-137	$5 \cdot 10^{-3}$	$8.9 \cdot 10^{-5}$	$7.3 \cdot 10^{-6}$

6 DESCRIPTION OF MODELS

Supplementary information about models used in various parts of the safety analysis is given in this chapter. It is not necessary to read this chapter in order to gain a general understanding of the calculations performed or the meaning of the results. It is rather provided to supply additional information to persons with special interest in one or other of the models. The descriptions given in this chapter are brief, and do not fully explain the models. For further information consult the references. It has not been found necessary to give more than a very brief explanation of models or methods that are well established.

6.1 BITUMEN FIRE

The consequences of a bitumen fire, whether in storage or during transportation, are calculated using the computer program CRAC /37, 38/. This program was developed for use in the American Reactor Safety Study, and CRAC or programs largely based upon CRAC are widely used, and there is no need to give details here. In addition to the references previously mentioned, references /55, 56/ are recommended. They give a more comprehesive description of calculations using CRAC.

6.2 FALL-IN-WATER ACCIDENT

The consequences of the fall-in-water accident are calculated using the computer program BIOPATH /52/. This program is designed for calculation of distribution of radioactive nuclides and their decay products released in an ecological system and for calculation of individual and collective doses. The ecological system is divided into a number of connected areas to simulate the gradually increased dispersion from the release point to the whole global zone. Each area consists of different compartments like soil, surface water, sediment, ground water. Dynamic exchange of the nuclides in the biosphere is simulated by compartment theory, and the exchange between the different compartments is described by nuclide specific transfer factors. In the present calculations doses have been calculated for a critical population group and to the whole population around the Baltic Sea.

6.3 RELEASE RATE FROM WASTE AND REPOSITORY

The transport of radionuclides from the moulds or drums is assumed to be governed by diffusion. This means that the nuclides are assumed to be completely in solution the moment water is present. For dissolved compounds the mass transport flux is at all points proportional to the concentration gradient. This is called Fick's first law of diffusion. The following equation is derived from Fick's law, and is used for calculating release of the single radionuclides through the matrix and its walls.

$$A_1 = \frac{a_0 \cdot s}{V} \left(\frac{D_0'}{\pi \cdot t}\right)^{\frac{1}{2}} \exp - \left\{\sum_{n=1}^{N} \frac{X_n^2}{4\, D_n' t} + \lambda t\right\} \quad (1)$$

where

- A_1 = release rate of nuclide in question (Bq/a)
- a_o = nuclide content at time zero (Bq)
- S = surface area of waste or of repository (m^2)
- V = volume of waste or repository (m^3)
- D_o' = diffusion coefficient for the nuclide in the matrix (m^2/a)
- t = time since time zero (a)
- X_n = thickness of the nth wall outside the matrix (m)
- D_n' = diffusion coefficient of nuclide in the nth wall (m^2/a)
- λ = radioactive decay constant (per year)

There are two computer programs that are developed in Scandinavia for such calculations; DIFMIG /30/, which was developed in Denmark, and GEOPATH /53/, which was developed in Sweden.

6.3.1 DIFMIG

The computer program DIFMIG /30/ is an improved version of an earlier program called COLUMN. It calculates one-dimensionally (i.e. column) the diffusive migration of single substances through arbitrary multi-barrier systems, according to the following equation.

$$\frac{\partial C}{\partial t} = D'(t)\,\frac{\partial^2 C}{\partial t^2} + F'(C,t) \qquad (2)$$

where

$D'(t)$ = time dependent effective diffusion coefficient (m^2/a)

$F'(C,t)$ = function expressing time dependent changes in concentration other than dispersion/diffusion, e.g. slow dissolution of a compound, radioactive decay and/or build-up of daughter products

The method takes the possible time dependent changes in the effective dispersion coefficient into account, the latter being defined as the product of the (possibly time dependent) retention factor (equation (3)), and the diffusion coefficient in the liquid phase (equation (4)).

$$R_\rho(t) = \left(1 + \rho \frac{1 - \varepsilon_d(t)}{\varepsilon_d(t)} K_d\right)^{-1} \tag{3}$$

$$D'(t) = R_\rho(t)\, D \tag{4}$$

where

R_ρ = retention factor
$\varepsilon_d(t)$ = time dependent porosity
ρ = density of the solid phase
K_d = distribution coefficient (m^3/kg), which is assumed to be constant throughout the time period investigated

The diffusion equation (2) is solved by a finite difference implicit method; and the resulting trigonal matrix equation is solved by standard methods.

6.3.2 GEOPATH

GEOPATH is a computer program developed for the analysis of radionuclide migration in saturated porous and fracture media /31/. Man-made barriers and waste products can be modelled with the code. The algorithm is two-dimensional, but GEOPATH can also be applied to three-dimensional axisymmetric geometries. The aqueous flow pattern must be specified by the user. The code is capable of handling linear chain decay, convective flow, anisotropic dispersion, molecular/ionic diffusion, equilibrium sorption, reversible and irreversible sorption kinetics, and leach processes. In principle GEOPATH can be applied to groundwater transport of any kind of contaminants, not necessarily radioactive ones.

The output from GEOPATH is to be made compatible with BIOPATH. A geohydrological model is presently under development, and will eventually complete the code package.

GEOPATH is an improved version of the earlier program GETOUT.

6.4 NUCLIDE MIGRATION IN GROUND

From the repository the radionuclides leak out with different rates into the ground and migrate with the ground water flow.

The ground water flow velocity, caused by the natural hydraulic gradient only, is calculated according to Darcy's law:

$$v_w = \frac{K_p \cdot i}{\varepsilon_k} \tag{5}$$

where

v_w = ground water flow velocity (m/s)
K_p = hydraulic conductivity (m/s)
i = hydraulic gradient (m/m)
ε_k = kinetic porosity

The corresponding transport times (T) are calculated dividing the distance (L) by the flow velocity:

$$T = \frac{L}{v_w} \tag{6}$$

The disturbance that a well causes to the natural flow of the ground water has been analysed using a computer program FEFLOW /32/ based upon a finite element method solution of the flow equations.

The flow in porous media can be described with Darcy's law:

$$v = \frac{k \cdot \nu}{\varepsilon_k \cdot \mu} \cdot \nabla\Psi = - \frac{K_p}{\varepsilon_k} \nabla\Psi \qquad (7)$$

where

k	=	intrinsic permeability (m^2)
ν	=	specific weight of the liquid (g/cm^3).
K_p	=	hydraulic conductivity (m/s)
Ψ	=	potential function
ε_k	=	kinetic porosity

In a two-dimensional case, assuming the liquid and the solid to be non-compressible, the potential function can be solved from the following equation:

$$\frac{\partial}{\partial x_1} \left(K_{11} \frac{\partial \Psi}{x_1}\right) + \frac{\partial}{\partial x_2} \left(K_{22} \frac{\partial \Psi}{\partial x_2}\right) = 0 \qquad (8)$$

The computer program FEFLOW solves the above equation by the finite-element method and gives as its output the principal directions and velocities of ground water flow as well as stream lines and flow times.

The combined effect of natural flow and well-induced flow can be found by simple superposition.

6.5 NUCLIDE MIGRATION IN TILL

The migration of radionuclides in porous media is calculated using the following formula:

$$V_n = K_i^{-1} \cdot v_w \qquad (9)$$

where

v_n = velocity of nuclide (m/s)
v_w = velocity of ground water (m/s)
K_i = retention factor

The retention factor can be calculated by the following equation:

$$K_i = 1 + K_d \cdot \rho_{TP} \cdot \frac{1-\varepsilon_d}{\varepsilon_d} \tag{10}$$

where

K_d = distribution coefficient (m^3/kg)
ρ_{TP} = dry particle density of the ground (kg/m^3)
ε_d = diffusion porosity

6.6 NUCLIDE MIGRATION IN ROCK

The transport velocity of a radionuclide in rock is calculated using equation (9), but the retention factor is calculated using the following equation /54/:

$$K_i = 1 + \frac{2\ K_d}{0.01\ a_2\ (K_p \cdot s)^{1/3}} \tag{11}$$

where

a_2 = geometric area of rock sample (30 m^2/kg)
s = average fissure spacing (m)
K_p = hydraulic conductivity (m/s)

6.7 WATER CONSUMPTION FROM A WELL

The maximum dose resulting from intake of contaminated water from the well is calculated using the following equation:

$$R = C_{max} \frac{I_v \cdot D}{V} \tag{12}$$

where

C_{max} = maximum inflow of nuclide into the well (Bq/a)

I_v = drinking water volume per person (m^3/a)

D = dose conversion factor for the nuclide (Sv/Bq)

V = dilution in the well (m^3/a)

The dilution volume is equal to the total volume of water flowing into the well.

6.8 SOIL CONTAMINATION

The release of radionuclides from the repository causes soil contamination around the repository. The following equation gives the concentration of radionuclides in the soil:

$$C_n = A_n/V \tag{13}$$

where

V = volume of contaminated soil (m^3)

The release of a certain nuclide from repository is given by

$$A_n = \frac{a_0 S}{V} \sqrt{D_0'} \left\{ \sqrt{\frac{4t}{\pi}} \exp\left[-\left(\sum_{n=1}^{N} \frac{X_n}{\sqrt{D_n'}} \right)^2 \cdot \frac{1}{4t} + \lambda_n t \right] - \left(\sum_{n=1}^{N} \frac{X_n}{\sqrt{D_n'}} \right) \operatorname{erfc} \left[\left(\sum_{n=1}^{N} \frac{X_n}{\sqrt{D_n'}} \right) \cdot \frac{1}{\sqrt{4t}} \right] \right\} \tag{14}$$

where

$erfc$ = complimentary error function
A_n = amount of nuclide in question (Bq)
a_o = amount of the nuclide at time zero in the repository (Bq)
S = surface area of the repository (m^2)
V = volume of the repository (m^3)
D'_o = diffusion coefficient of nuclide in matrix (m^2/a)
t = time since time zero (a)
X_n = thickness of the nth wall outside the matrix (m)
D'_n = diffusion coefficient of nuclide in the nth wall (m^2/a)
λ_n = radioactive decay constant (per a)

6.9 UPTAKE IN VEGETABLES

The following equation is used in the calculations:

$$U_{veg} = \frac{C_n \cdot B \cdot \exp(-\lambda t)}{\rho_s} \tag{15}$$

where

U_{veg} = concentration in vegetables (Bq/kg)
C_n = concentration in the soil (Bq/m^3)
B = biological uptake factor
ρ_s = soil density (kg/m^3)

6.10 INGESTION OF VEGETABLES

The following equation gives the doses from ingestion of vegetables:

$$R = U_{veg} \cdot I_{veg} \cdot D \tag{16}$$

where

U_{veg} = concentration in vegetables (Bq/kg)
I_{veg} = annual consumption of vegetables (kg/a)
D = dose conversion factor (Sv/Bq)

6.11 INHALATION OF CONTAMINATED DUST

The following equation gives the doses from inhalation of contaminated dust:

$$R = \frac{C_n \cdot K}{\rho} \cdot b \cdot D \tag{17}$$

where

R = dose rate (Sv/h)
C_n = concentration in soil (Bq/m^3)
K = dust content in the air (kg/m^3)
b = breathing rate of exposed individual (m^3/h)
D = dose conversion factor (Sv/Bq)
ρ = soil density (kg/m^3)

7 PARAMETER VARIATIONS

The sensitivity of calculated results to variation in parameter values or calculational conditions has been investigated in this study to a limited extent, and these parameter variations are described in the present chapter.

7.1 NUTRITION PARAMETER VARIATIONS

For some of the calculated accident scenarios, doses are dominated by exposure via nutrition pathways. It was felt that marked local variations in the values of many of the nutrition pathway parameters were to be expected. Members of the NKA radioecology group were asked to comment upon the parameter values used in the reference calculations of the present study (the American parameter values from WASH-1400 /37/, as well as the model itself, and also to recommend parameter values typical of each of the four Nordic countries. It was,however, possible to do this only for the bitumen fire scenarios during storage or transportation.

These comments are described in detail in /15/. The model was generally judged as being acceptable Various specific alternative parameter values were however recommended. The most important of the suggested alternative values are summarised in the following.

- Fraction of deposited activity that is initially intercepted by the plants is in the reference set to 50 %. Alternative value (suggested by Norwegian representative) is 25 %.

- Vegetable consumption per day is (ref. calc.) 0.12 kg. Alternative value (Sweden) 0.8 kg.

- A cow eats grass from 45 square meters per day (ref. calc.). Alternative values 200 square meters (Norway) and 100 square meters (Sweden).

- Vegetables grown per unit area is (ref. calc.) 2.4 kg per square meter. Alternative value is 3 kg per square meter (Sweden).

- Percentage of cesium intake (direct deposition) that is via milk is (ref. calc.) 33.3 %. Alternative value is 16 % (Denmark).

- In addition the Danish and Norwegian representatives specified alternative values of certain of the concentration factors mentioned in the next paragraph.

- The calculations are actually carried out using so-called concentration factors derived from the parameters discussed above. From the information gathered, four sets of concentration factors were constructed, one for each of the Nordic countries, and four new sets of calculations were carried out. The results of these calculations are shown in figures 7.1 and 7.2. The calculations were carried out for one specific weather condition, but the range of variation will be very similar for all weather conditions.

The calculations show that the results are only moderately sensitive to the assumptions for nutrition pathway calculations.

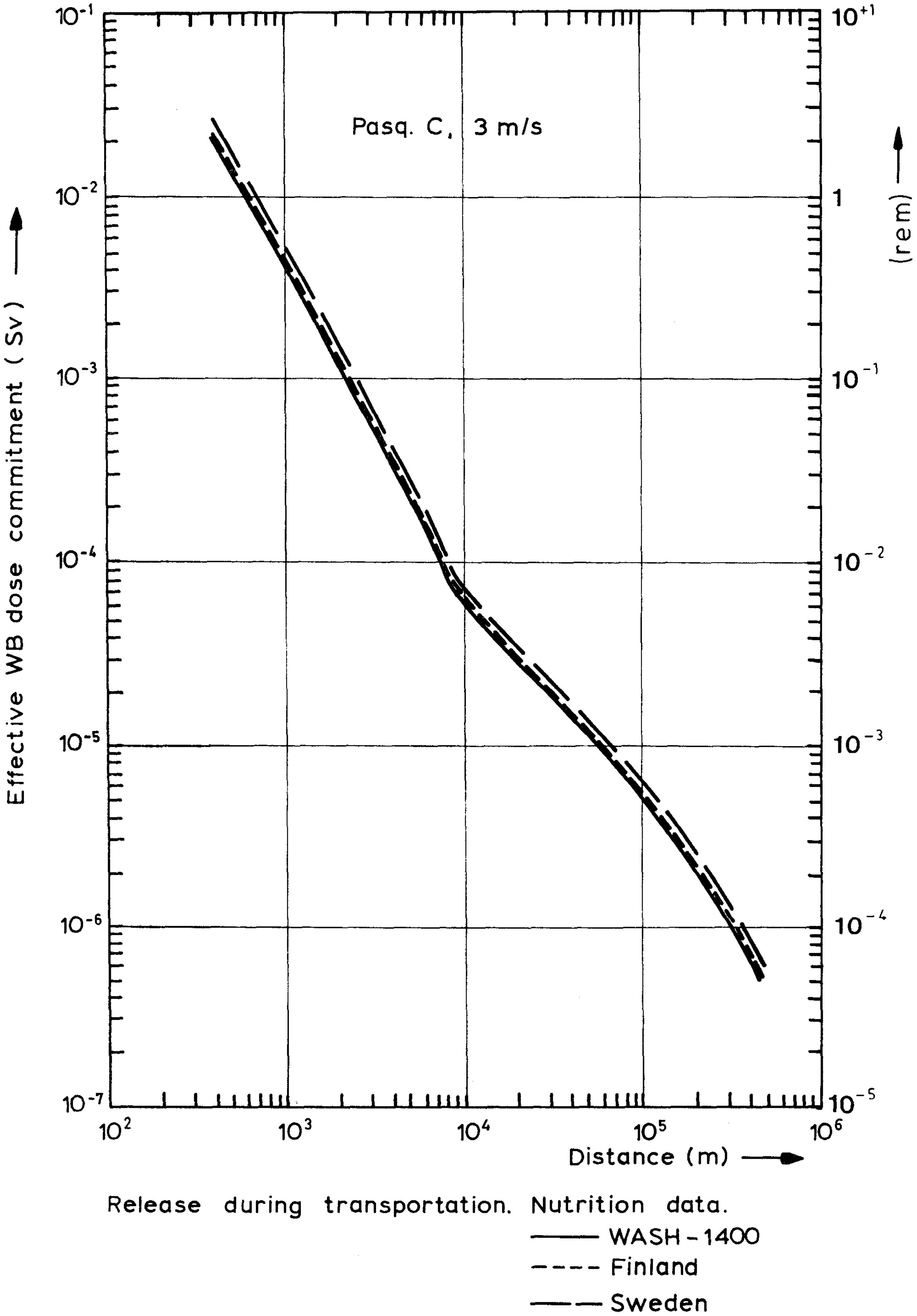
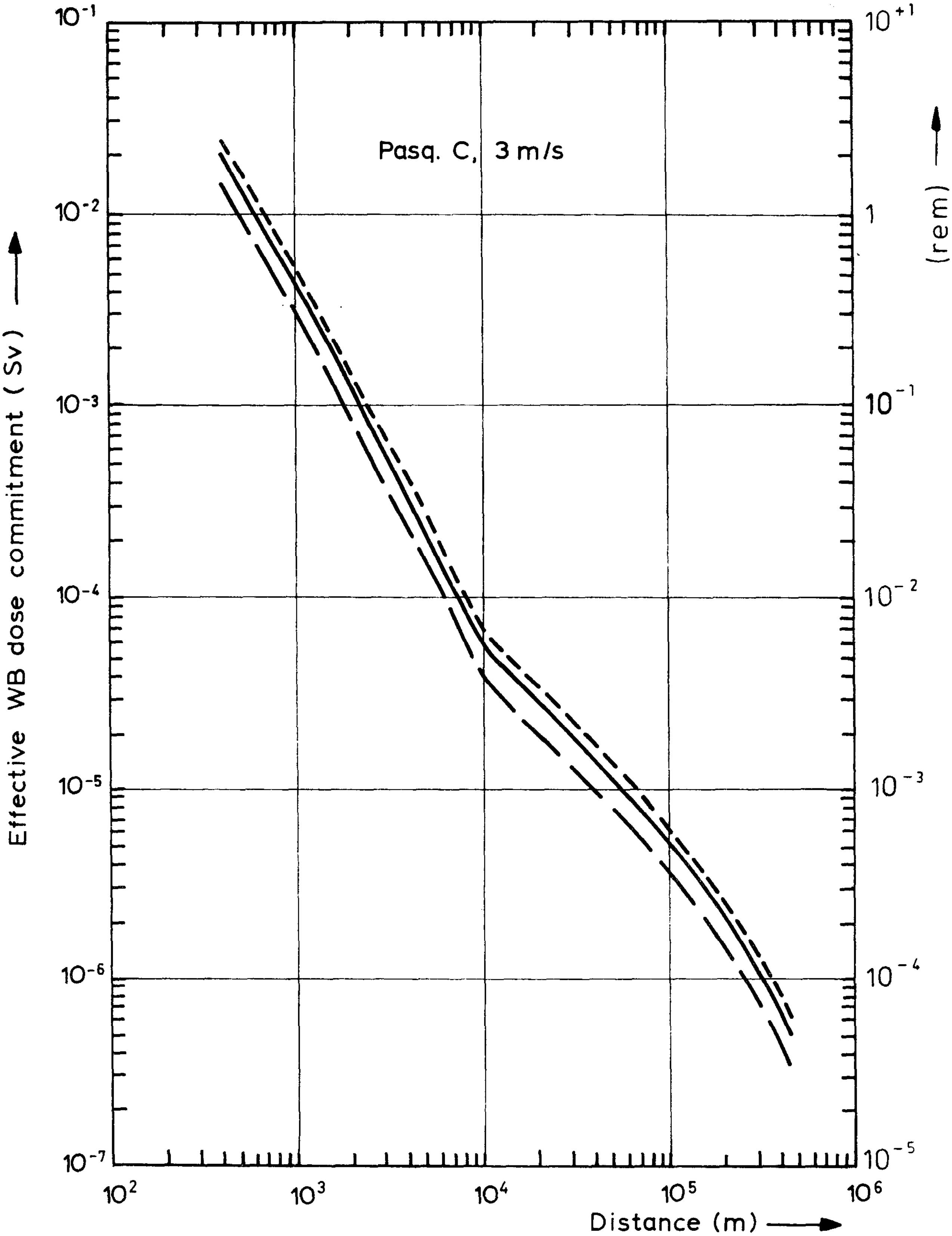

Pasq. C, 3 m/s
Effective WB dose commitment (Sv)
(rem)
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
10^{+1}
1
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{2}
10^{3}
10^{4}
10^{5}
10^{6}
Distance (m)
Release during transportation. Nutrition data.
WASH - 1400
Finland
Sweden

Figure 7.1



Release during transportation. Nutrition data.

—— WASH-1400

- - - - Norway

— — Denmark

Figure 7.2

7.2 DIFFUSION COEFFICIENT

The release rate from the repository is assumed in the present calculations to be governed by diffusion. Variation of the diffusion coefficient (or the leach rate) acts upon calculated doses from all nuclides.

Down to a value of the diffusion coefficient of abo EE-6 m2/a there is essentially linearity with the leach rate for long-lived nuclides, while short-lived nuclides like Sr-90 and Cs-137 will deviate from linearity, due to radioactive decay, and the leach rate will decrease more rapidly. This is shown in figure 7.3 /39/.

As an example, by replacing the leach coefficient for Cs-137 in concrete of 10^{-14} m^2/a by 10^{-5} m^2/a, the individual dose will be reduced by a factor of 150. In the case where the Cs-contribution is dominant, where doses from a well close to a repository are calculated, this would make the dose from Cs-137 equal to that of C-14. The same will be the case in the calculation for a fractured repository.

A further reduction of the leach coefficient y a factor of 10 would entirely eliminate the dose contribution from Cs-137. In Technical Part III is recommended to lower leach coefficient values for C-14 and I-129 by a factor of 100. This would reduce the final doses to maximum 10^{-5} Sv/a (10^{-3} rem/a) for shallow land burial in the case of water consumption, and correspondingly lower (10^{-6} and 10^{-5} Sv/a) (10^{-4} and 10^{-3} rem/a) for concrete bunker and rock cavern respectively.

If these reduced leach coefficients, indicated by laboratory experiments, are applied in the reference case, then the maximum individual dose, calculated for water consumption from a well drilled close to the repository, will be at an extremenly low level.

7.3 DISTRIBUTION COEFFICIENT

For long-lived nuclides, where the distribution coefficient in the present calculations has been assumed to be zero, a considerable increase in the value of the distribution coefficient (and corresponding decrease in migration velocity) would be required in order to significantly reduce the calculated doses. This is quite obvious, and is a result of the calculational conditions used. An increase in distribution coefficient in these calculations will only increase migration time, while all other conditions are kept unchanged; and the only reduction will be due to radioactive decay. In a realistic case, longer migration time would probably be accompanied by other effects, like increased traverse migration, change of chemical form, etc.

Short-lived nuclides (Sr-90 and Cs-137) are very sensitive to variations in value of the distribution coefficient. Increase by a factor of 10 of the assumed values for rock, results in a 300 times lower individual dose. In sandy till, where the water flow is slow, a 10-fold increase in the value, reduces the individual dose by a factor of 10^{12} /40/.

7.4 NUCLIDE COMPOSITION

In chapter 3 of this report are given the results of calculations for the individual nuclides, for many scenarios. Sensitivity to nuclide composition can be read from these results. In all the present calculations, there is direct proportionality between nuclide content and dose from that specific nuclide.

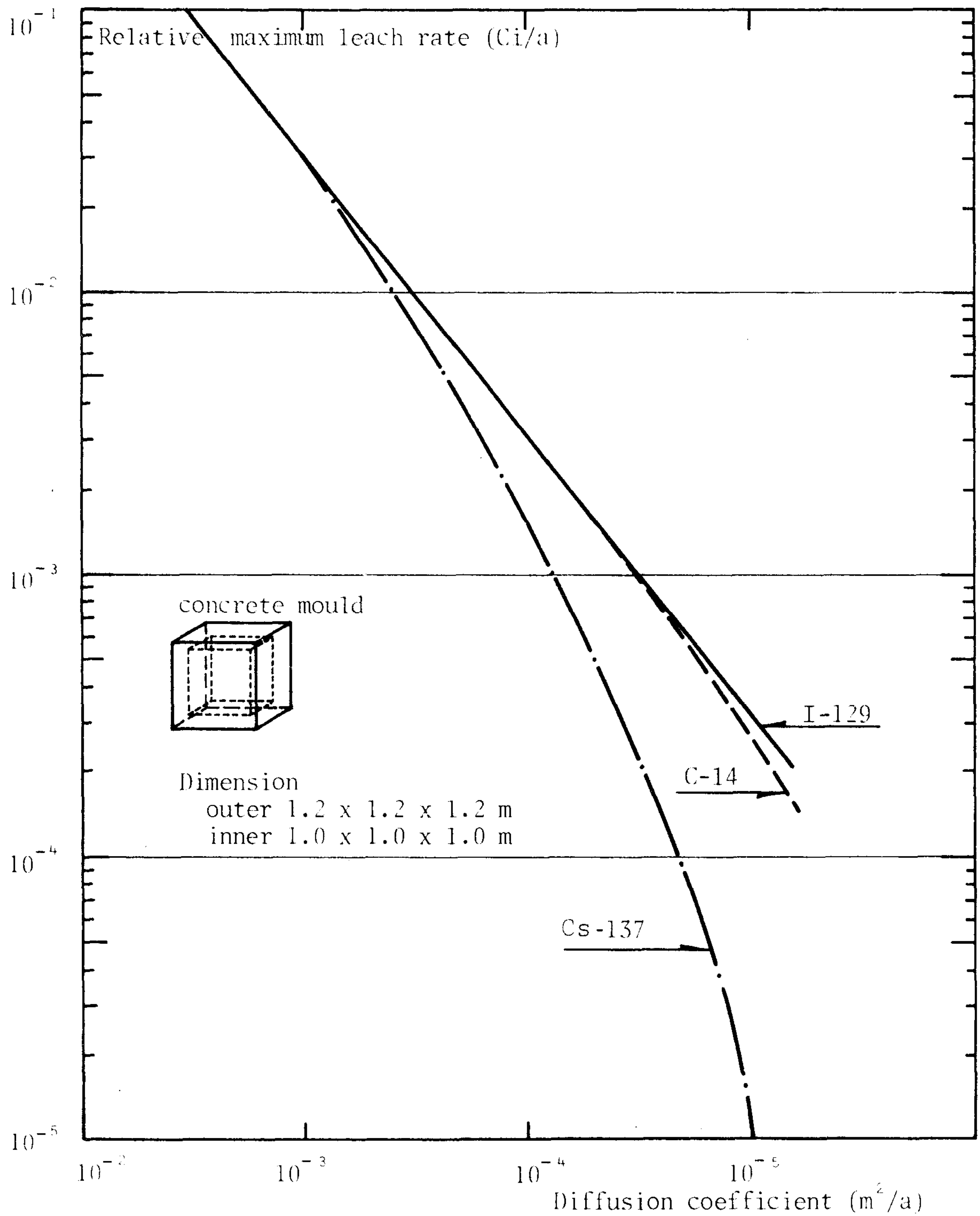


Figure 7.3 Variation of the diffusion coefficient in concrete mould

7.5 POPULATION DENSITY

Sensitivity of the results to population density is of course relevant only in connection with collective doses. In the case of transportation accident, two alternative population distributions were used; one of moderate to low population density, and one where the release is assumed to take place in the center of a city. The collective doses are of course higher in the latter case, but not extremely much higher. The effect on doses can be examined by comparing figures 4.6 and 4.7 in section 4.5.

7.6 RESPONSE SURFACE METHOD

One of the more systematic methods available for performing sensitivity analyses is the response surface method.

In reality this is a combination of simple parameter variation analyses and interpolation methods. The result (e.g. the dose) is calculated for a number of combinations of values of the various parameters; using the methods generally used to calculate the dose for the specific problem analysed. It is then postulated that the results for intermediate values of the parameters are positioned on a multi-dimensional "surface" connecting all the points (one specific value of each parameter plus the result defines a point in multi-dimensional space). If only two parameters are varied, and the rest are kept constant, the surface will be two-dimensional, and it can be plotted. An example is shown in figure 7.4. For this case (the two-dimensional) the response surface method may serve to visualize parameter variation quite effectively.

In the present study a fragment of a response surface analysis /57/ has been performed, to demonstrate the method. The case analysed is a rock cavern disposal facility positioned so that the distance to the nearest well in the ground water flow direction is 100 meters. The three parameters varied express the leaching from the bitumen matrix, diffusion through repository walls, and ground water transport time. The result from variation of two of the parameters (the first two) for one specific nuclide is shown in figure 7.4.

It is concluded in the case of long-lived nuclides that

- the effect of ground water transport time is negligible.

- the effect of leaching from bitument matrix and dissusion through repository walls is small. Changing the relevant parameters by a factor 100 changes the result by a factor of 10. Changing both at the same time, gives a change in result by a factor 100.

In the case of a short-lived nuclide it is concluded that

- the effect of ground water transport time is very large (due to radioactive decay).

- the effect of diffusion through repository walls is very large. A change of parameter value by a factor of 10 gives a change in result by a factor of 1000.

- the effect of leaching from waste matrix is small as a change in parameter value by a factor of 100 gives a change in result by a factor of 10.

These results agree well with what is presented in the previous sections of this chapter. As always, it is important to remember that the results of an analysis depends upon the assumptions made.

If the full potential of the surface response methods is to be used, the distributions of the parameter values should be available. This has not been the case in the present study, and as mentioned, there is in that case little difference between the response surface method and parameter variations as usually performed. If parameter value distributions are available, however, an uncertainty analysis could be performed, and one could obtain a probability distribution of e.g. the dose value.

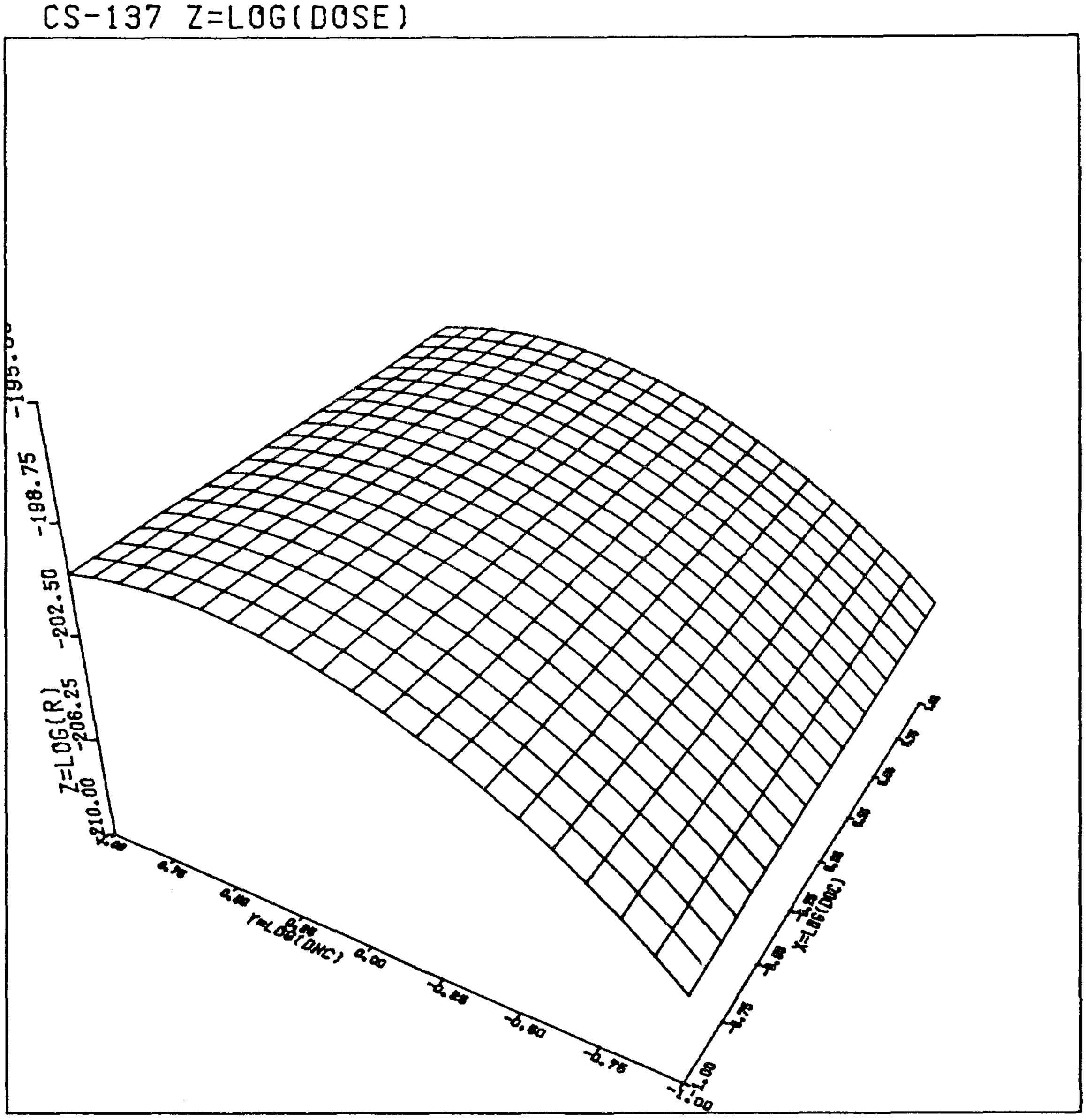


Figure 7.4 The responce surface in case of Cs-137 when leaching from the bitumen matrix and diffusion through the repository walls have been varied. In the figure X = log ($D_o c$), and Y = log ($D_n c$) where

c = variation coefficient

D_o = diffusivity in bitumen

D_n = diffusion through repository walls

X = Y = 0 corresponds to parameter values used in the reference case calculations.

8 COMPARISON OF RISKS

The ultimate aim of a safety analysis is to compare the risk in the different parts of the managemant system, or between alternate solutions of the same part of the system. Critical part 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 experessed in identical manner in the different parts of the system. This may be very difficult, as also in the present case.

8.1 RESULTS FROM THE PRESENT STUDY

The results calculated in the present safety analysis vary much in type, due to differences in methods, data and general approach. In the following it is attempted to summarize what types of results have been calculated for the various parts of the management system:

- Storage, bitumen fire.
  Individual life-time committed doses as function of distance, and for different weather conditions. Total collective dose commitments.
  Weather- and population-distribution-related probability distributions.
  Probability distributions.
  Probability of bitumen fire.

- Transportation, fall-in-water accident.
  Maximum individual committed doses based upon quite conservative assumptions.
  Total collective dose commitment.
  Probability of transportation accident followed by water immersion.

- Transportation, bitumen fire.
  Individual life-time committed doses, as function of distance, and for different weather conditions.
  Total collective dose commitments.
  Weather- and population-distribution-related probability distributions.
  Probability of transportation accident followed by bitumen fire.

- Transportation, ship wreckage.
  Maximum individual committed doses.
  Maximum collective dose commitments.
  Probability of collision followed by various types of events.

- Disposal, well scenarios.
  Maximum annual individual committed doses (this means committed dose from the maximum release during a one-year period).
  Maximum annual collective committed doses (not dose commitments). This maximum is assumed to occur after 500 years, where the release each year is the maximum annual release.
  When respository is intact, the probability is assumed to be one; when not it can not be assessed.

- Disposal, dwelling on top of the repository.
  Annual and life-time individual doses, using methods expected to give quite realistic results.
  Probability can not be assessed.

- Disposal, intrusion by excavation.
  Individual dose rates, using methods expected to give quite realistic results.
  Probability can not be assessed.

- Disposal, farming
  Maximum annual individual committed doses.
  Probability can not be assessed.

It is seen from the above that the doses calculated for the different parts of the management system are 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 long-lived nuclides, combined with the fact that the doses are calculated using simplified methods. The well scenarios may serve as illustration: it is assumed that the well just gives sufficient supply of drinking water, which means that there will be no accumulation of radioactive materials in the well. They are consumed as they seep into the well. For long-lived nuclides and a constant leak rate into the well, the individual doses will be about the same from year to year. After consumption the radioactive nuclides will go with the 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 on comparisons may be carried out: comparisons 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.

Some specific results have been chosen to serve as basis for the comparisons, as carried out here. They are summarized in the following, rounded off to nearest 5 or 10. Comparisons along other lines might have been chosen, and other parts of the results presented in chapters 3, 4 and 5 might then have been more relevant.

- Storage, bitumen fire.

Individual lifetime committed dose at 500 meters distance from the site of the accident.	0.1 Sv **
Dose in more favourable weather condition.	0.05 Sv
Total collective dose commitment	
maximum.	50 manSv
most probable range.	1-10 manSv
Probability * of airplane crash against building.	10^{-5}

* Probabilities in the storage and transportation cases refer to the total reference system, i.e. 6 reactors operated for 30 years. Distance reactor-repository = 200 km.

** 1 Sv = 100 rem

- Transportation, fall-in-water accident.

Maximum annual individual committed dose.	10^{-5} Sv/a
Total collective dose commitment.	0.05 manSv
Propability of tranport accident followed by water immersion	10^{-6}

- Transportation, bitumen fire

Individual lifetime committed dose at 500 meters distance from the site of the accident.	0.01 Sv
Dose in more favourable weather condition (maximum, which occurs at 1500 meters distance).	10^{-3} Sv
Total collective dose commitment	
maximum.	50 manSv
most probable range.	0.5-10 manSv
Probability* of transport accident followed by fire	10^{-5}

- Transportation, ship wreckage.

Maximum individual committed dose	
(release to sea from one undamaged concrete mould) (occurs 10 years after accident).	0.5 microSv
(release to sea from one severely damaged concrete mould).	5 microSv

* See foot-note, previous page.

Maximum collective dose commitment
(release to sea from one undamaged concrete mould) (occurs 10 years after accident). 0.01 manSv
(release to sea from one severely damaged concrete mould). 0.05 manSv

Probability* of concrete block lost as sea, and severely damaged when hitting bottom. $< 10^{-5}$
Probability* of vessel foundered at depth greater than 80 meters, and recovered after 6 months. $< 10^{-5}$

- Disposal, water from well near intact repository.

Maximum annual individual committed dose.

Shallow land burial (C-14- and I-129-dominated component, max. occurs 25 years after closure). 0.05 Sv/a
Shallow land burial (Cs-135-dominated component, max. occurs 10 000 years after closure). 5×10^{-4}

Concrete bunker (C-14- and I-129-dominated component, max. occurs 200-400 years after closure). 5×10^{-5}
Concrete bunker (Cs-135-dominated component, max. occurs 10,000 years after closure). 5×10^{-7} Sv/a

Rock cavern (C-14- and I-129-dominated component, max. occurs 300-500 years after closure). 5×10^{-5} Sv/a
Rock cavern (Cs-135-dominated component, max. occurs 20,000 years after closure). 5×10^{-7} Sv/a

* See foot-note under storage.

Maximum annual collective committed dose (Committed dose from the exposure in year 500 after the year of maximum release to well, see above).

Shallow land burial (I-129-dominated component).	10 manSv/a
Shallow land burial (Cs-135-dominated component).	1 manSv/a
Concrete bunker (I-129-dominated component).	0.05 manSv/a
Concrete bunker (Cs-135-dominated component).	$5x10^{-3}$ manSv/a
Rock cavern (I-129-dominated component).	0.05 manSv/a
Rock cavern (Cs-135-dominated component).	10^{-3} manSv/a

Probability can not be assessed, but it should be remembered that there is assumed a period of institutional land use control of 200 years after closure of the repository.

- Disposal, water from well near fractured repository.

The case does not apply to shallow land burial. All individual doses are very similar to the ones calculated for the preceding scenario. All collective doses are practically identical to the ones calculated for the preceding scenario.

- Disposal, water from well directly outside intact repository.

This and all the following cases do not apply during the period of institutional land control; which implies that the earliest possible time at which doses can be received in these scenarios is 200 years after closure.

Maximum annual individual dose.

Shallow land burial (Cs-137-dominated).	0.5 Sv/a
Concrete bunker (Cs-137-dominated).	0.05 Sv/a
Rock cavern (C-14- and I-129-dominated).	5×10^{-5} Sv/a

Collective doses will be almost identical to the values in the previous two scenarios.

- Disposal, water from well directly outside fractured repository.

This case does not apply to shallow land burial.

Maximum annual individual dose.

Concrete bunker (Cs-137-dominated).	0.1 Sv
Rock covern (C-14- and I-129-dominated).	5×10^{-5} Sv

Collective dose, as above.

- Disposal, dwelling on top of repository.

Annual individual dose.	10^{-9} Sv/a
Life-time individual dose.	5×10^{-8} Sv

- Disposal, intrusion by excavation.

Maximum dose-rate. No earth cover.

Shallow land burial.	10^{-4} Sv/h
Concrete bunker.	5×10^{-7} Sv/h

Dose-rate with one meter earth cover

Shallow land burial.	10^{-8} Sv/h
Concrete bunker.	10^{-11} Sv/h

Individual committed dose via inhalation (Cs-137-dominated).

Shallow land burial, blasting.	$5x10^{-7}$ Sv
Shallow land burial, digging.	$5x10^{-6}$ Sv
Concrete bunker, blasting.	$5x10^{-9}$ Sv
Concrete bunker, digging.	$5x10^{-8}$ Sv

- Disposal, farming.

Maximum annual individual committed dose via vegetables.

Shallow land burial (Cs-137-dominated).	10^{-4} Sv/a
Concrete bunker (Cs-137-dominated).	$5x10^{-6}$ Sv/a

Maximum annual individual committed dose, via inhalation (Cs-137-dominated).

Shallow land burial.	10^{-9} Sv/a
Concrete bunker.	10^{-10} Sv/a

8.2 BACKGROUND RADIATION

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 8.1 are given typical values of annual doses, gathered from numerous references (among them NCRP,1975, UNSCEAR,1977, EPA,1977, BEIR,1979). In figure 8.1 are given the same type of information, but there a ranges of possible values are indicated.

Table 8.1. Average value of equivalent dose to population /58/.

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

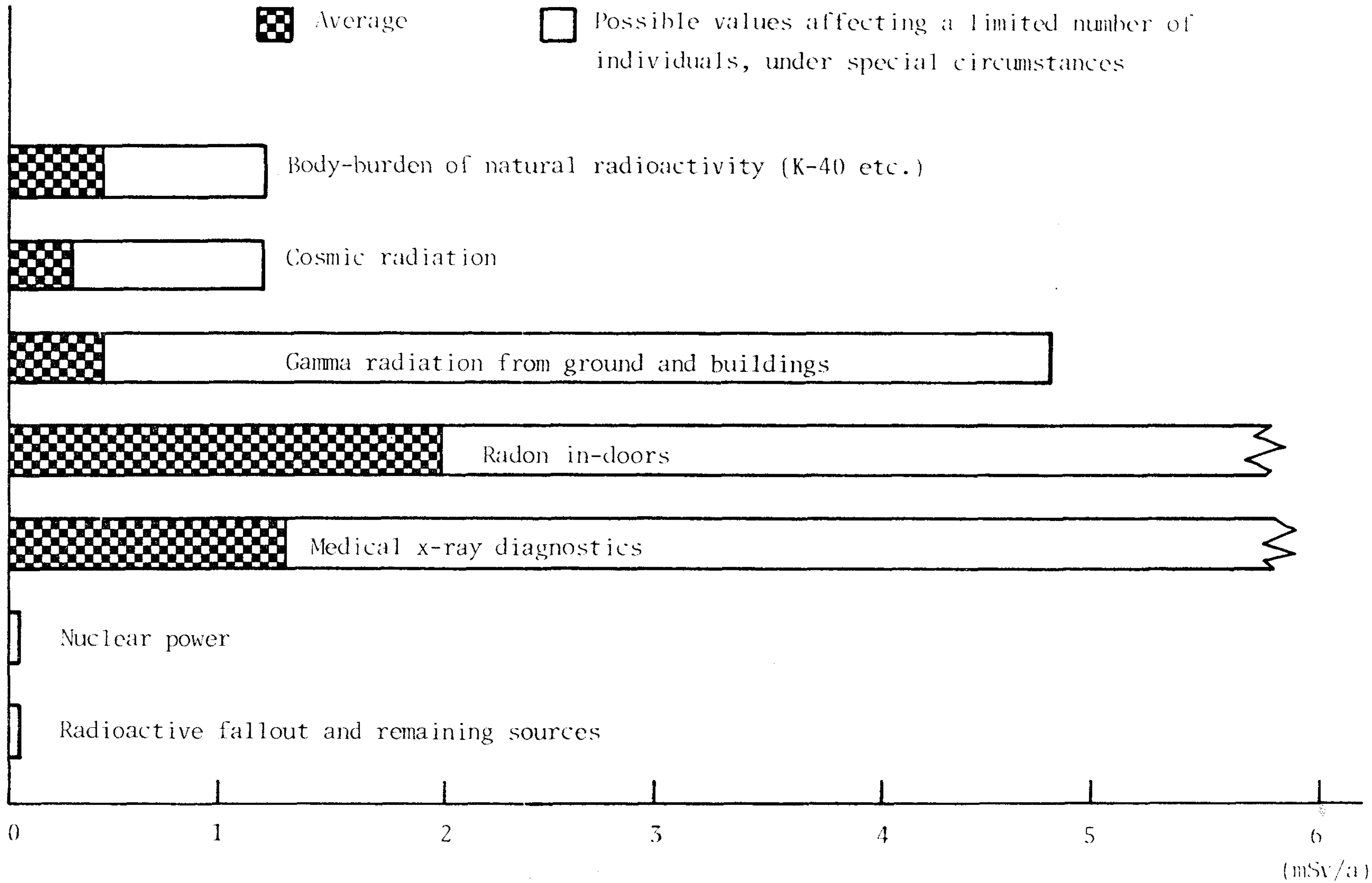


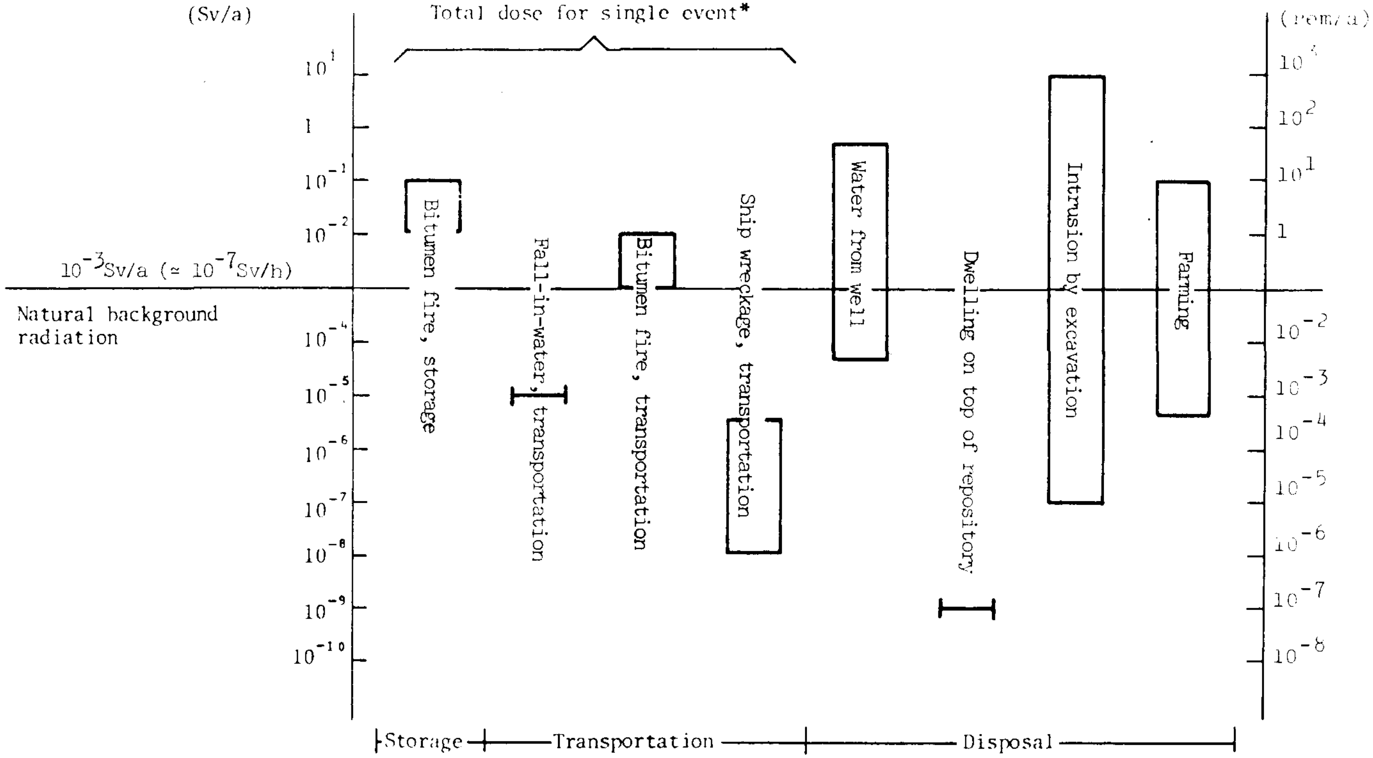
Figure 8.1 Variation of effective dose from various sources in society

8.3 COMPARISON

The results of the calculations, given in tabular form in section 8.1, are summarized here in figures 8.2 and 8.3, and compared to the natural background radiation level. The rectangles do not indicate actual ranges of uncertainty, but are only meant to show the range between the highest and lowest doses or dose rates calculated in this study for one scenario or a class of related scenarios.

It is important to remember, when evaluating the results as shown here, that the results for the different scenarios are in most cases not directly comparable. How can one e.g. compare a dose received during the period of supervision to a dose received 10,000 years later? And how can an annual individual committed dose, which will remain approximately constant for hundreds of years, be compared to a dose commitment from a single accidental release?

The two figures 8.2 and 8.3 should be used together with constant reference to the more complete information in section 8.1, and in the chapters describing the calculations. But used with great care, comparisons of the type shown in the two figures may be informative.



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

Figure 8.2 Individual doses and dose rates

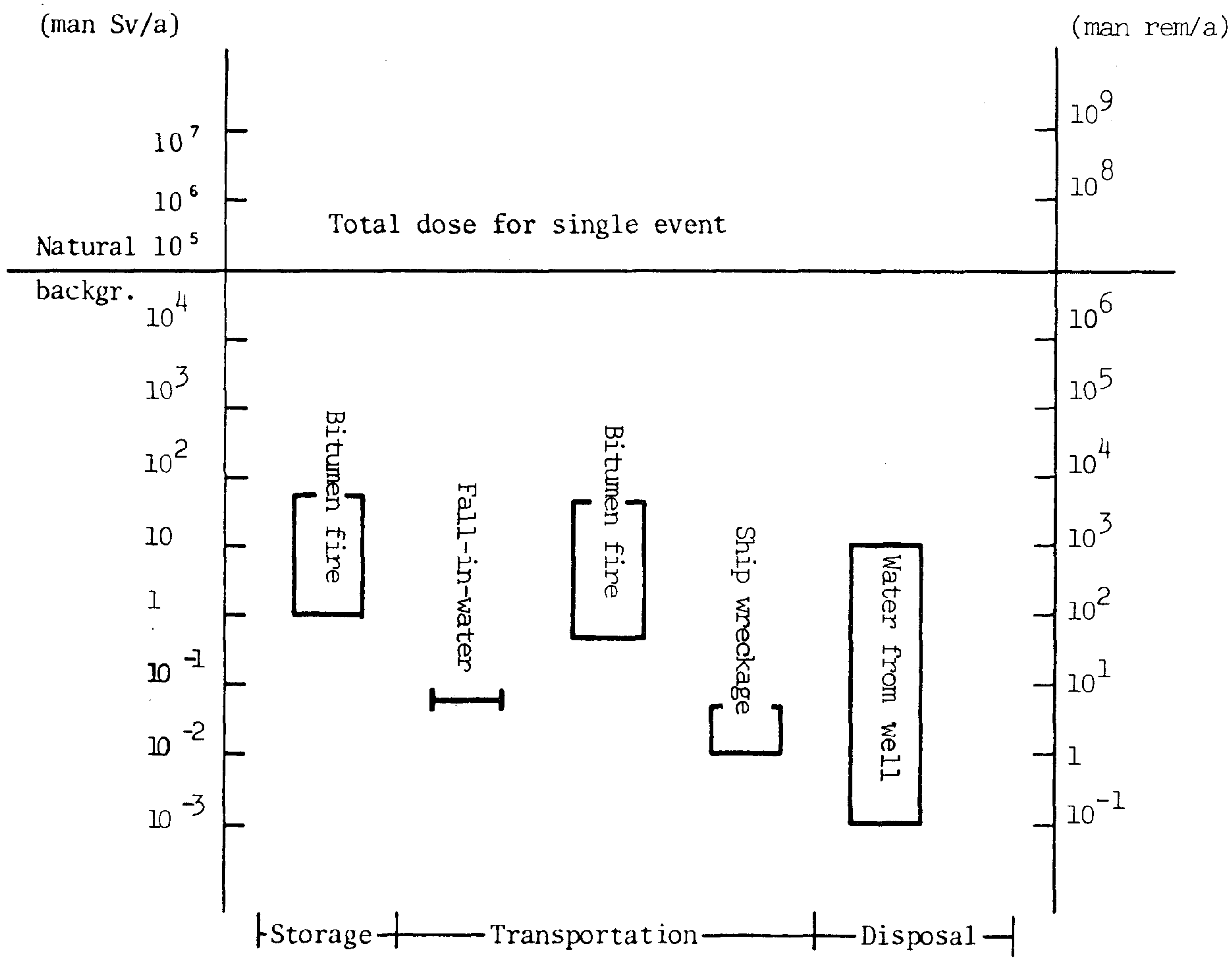


Figure 8.3 Collective doses and dose rates

REFERENCES

1. Annals of the ICRP Vol. 1 No. 3. ICRP Publication No. 26. Recommendations of the International Commission on Radiological Protection. Pergamon Press, Oxford 1977.

2. Adams, N., Hunt, B.W., and Reissland, J.A., Annual limits of intake of radionuclides for workers. National Radiological Protection Board. NRPB-R82. 1978.

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

4. Aittola, J.P., Valkiainen, M., Autio, D., Gamma-radiation stability of different Finnish bitumen types. Paper presented at the 6th conference of Nordic society for radiation research and technology, Hirtshals, Denmark, August 1977.

5. Brodersen, K., Influence of microorganisms on the leaching of activity from radioactive waste solidified in bitumen. A literature survey and some general considerations. (In Danish). NKA/AO(79), May 1979.

6. Bonnevie-Svedsen, M., Tallberg, K., Aittola, J.P., Tollback, H., Studies on the incorporation of spent ion exchange resins from nuclear power plants into bitumen and cement. IAEA-SM-207/78.

7. Aittola, J.P., Chyssler, J., Ringberg, H., Thermal stability of ion exchange resins, KBS report, unpublished, April 1981.

8. Rämö, E., Pulkkinen, U., Aittola, J.P., Damage of reactor waste packages in transport accidents. NKA/AO (79)17, October 1979.

9. Hahkala, M., Mehtonen, A., Aittola, J.P., Incineration of the ion exchanger resin/bitumen mixture. Particle size distribution in the off-gas. NKA/AO(79)21.

10. Nykyri, M., Valkiainen, M., Incineration properties of ion exchange resin. (In Finnish). NKA/AO(79)22, August 1979.

11. Killerud, K., Influence from outside on nuclear power plants, Scanpower A/S report 2.22.09 - D9, 1974.

12. Tveten, U., Radioactive waste transportation risk and temporary storage fire risk. NKA/AO(80)34.

13. Tveten, U., Decription of dose calculations via the nutrition pathways. (In Norwegian). NKA/AO(80)10.

14. Tveten, U., On calculation of individual and collective doses. (In Norwegian). NKA/AO(80)16, January 1980.

15. Tveten, U., Calculation of doses via nutrition NKA/AO(80)17, January 1981.

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

17. Larsen, V.V., Vinther, A., Brodersen, K., Smallscale firetest of ion exchange resin solidified by bitumen or cement. NKA/AO(80)45, January 1981.

18. Aittola, J.P., Chyssler, J., Ringberg, H., Fire experiments with bituminized ion exchange resin, KBS report, unpublished, April 1981.

19. Pihlajavaara, S.E., Aittola, J.P., Radwaste concrete: Solidification of nuclear wastes with Portland cement, NKA/AO(79)3, April 1979.

20. Nielsen, P.O., Probability of truck transport accidents. NKA/AO(80)33.

21. Nielsen, P.O., Fall-in-water accident (Releases). NKA/SO(80)15, Rev. 2, June 1980.

22. Nielsen, P.O., Fall-in-water accident - Dose calculations. NKA/AO(80)32, Rev.2, August 1980.

23. Bergman, R., Bergström, U., Evans, S., Dose and dose commitment from ground water carried radioactive materials by final disposal of used reactor fuel. (in Swedish). KBS Technical Report no. 100. Studsvik/K2-79/92, February 1979.

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

25. Jonsson, E., Fire test with bituminized waste (in Swedish). Bofors Nobel Chematur AB, PRAV report 1.8, April 1978.

26. Grundfelt, B., Eriksson, E., Shielded transport containers for reactor waste - damage scenarios and container failures. (In Swedish). PRAV report 1.30, April 1980.

27. Ericsson, A.M., Grundfelt, B., Brandberg, S., Determination of feasible accidents during transportation and their frequency. (In Swedish). PRAV report 1.16, June 1978.

28. Kjellbert, N., Appelgren, A., Bergman, R., Consequences of accidents by sea transportation of reactor waste: Radiation doses from activity released from a waste package. (In Swedish). PRAV report 1.17, July 1978.

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

30. Bo, P., Carlsen, L., DIFMIG - A computer program for calculation of diffusive migration of radionuclides through multi-barrier sysstems. Risö National Laboratory. Report M-2262.

31. Marklund, J.E., Kjellbert, N., GEOPATH - A computer program for the analysis of radionuclide migration in the geosphere. Studsvik. Report T2-80/67 and K2-80/393 1980.

32. Puttonen, J., Salo, J.P., Groundwater flow calculations as a part of the safety analysis waste. (In Finnish), 12th Finnish Rock Mechanics Meeting, 1979.

33. Bo, P., COLUMN - Numerical solution of migration equation involving various physicochemical processes. Risö National Laboratory. 1978. Unpublished report.

34. DeMier, W.W., Cloninger, M.O. Burkholder, H.C., Liddel, P.J., GETOUT - A computer program for predicting radionuclide decay chain transport through geologic media. Report PNL-2970. 1979.

35. Tveten, U., Thomassen, D., ARCON 3 - A computer program for calculating air relative concentration for radioactive materials released to the atmosphere. Instute for Energy Technology, Kjeller, 1975. Unpublished report.

36. Fryer, L.S., TIRION 4 - A computer code for calculating the consequences of releasing radioactive material to the atmosphere. Report SRD R 120. 1978.

37. Reactor Safety Study Appendix VI: Calculation of reactor accident consequences. WASH-1400 (NUREG 75/014), U.S. Nuclear Regulatory Commission. 1975.

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

39. Stenquist, C., Variations in the diffusion constant in the waste packages. (In Swedish), NKA/AO(80)28, June 1980.

40. Stenquist, C., Waste repository - damage analysis. (In Swedish), NKA/AO(79)31 B.I., October 1979.

41. Brodersen, K., Some system analysis concerning the philosophy of barriers and the design of a waste storage facility. (In Danish). NKA/AO(78)19. 1978.

42. Brodersen, K., Some general conditions concerning the siting of a near-surface repository for reactor waste in Denmark. (In Danish). NKA/AO(78)19. 1978.

43. Stenquist, C., Diffusion migration and dose calculation. (In Swedish). NKA/AO(79)23 rev. 1979.

44. Brodersen, K., Circumstances that can lead to release from reactor waste in a concrete bunker. (In Danish). NKA/AO(79)36. 1979.

45. Stenquist, C., Calculation of release and doses. (In Swedish). NKA/AO(79)37. 1979.

46. Stenquist, C., Calculation of doses from disposal facilities. (In Swedish). NKA/AO(80)18 rev. 1. 1980.

47. Heiskanen, K., Release of radionuclides from the repository. NKA/AO(80)29. 1980.

48. Nielsen, P.O., System- and safety analyses of reactor waste. Notes about earthquake, erosion, and hydrology. (In Norwegian). NKA/AO(80)12. 1980.

49. Heiskanen, K., Härkönen, H., Peltonen, E., Puttonen, J., The effect of a well on the radionuclide migration. NKA/AO(80)14 with appendix. 1980.

50. Kjellbert, N. et al., Central repository for low- and intermediate-level waste (ALMA), Conceptual design, siting and safety. Report PRAV 1.29.1980.

51. Bergman, R. et al., Ecological transport and radiation doses from radioactive substances carried by groundwater. KBS-40. 1977.

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

53. Marklund, J.E., Kjellbert, N., GEOPATH - A computer program for the analysis of radionuclide migration in the geosphere. Studsvik/T2-80/67. 1980.

54. Grundfelt, B., Transport of radionuclides with groundwater from a rock repository. (In Swedish). KBS-43. 1977.

55. Tveten, U., Thomassen, D., Kvaal, E., Accident analysis performed for the Norwegian Government Committee on Nuclear Power, KR-156, Kjeller, February 1979.

56. Nuclear power and safety, Norges offentlige utredninger, NOU 1978:35C, Oslo, 1978.

57. E. Peltonen and R. Sairanen, Disposal of reactor waste in bedrock - sensitivity analysis of dose by response surface method, NKA/AO(81)6.

58. Bertil Persson, Comments on calculation of collective doses and presentation of risks. (In Swedish), NKA/AO(80)43.