Danish Decommissioning

Pre-feasibility study for final disposal of radioactive waste. Disposal concepts

Main Report

May 2011

COWI A/S

Parallelvej 2
DK-2800 Kongens Lyngby
Denmark

Tel +45 45 97 22 11
Fax +45 45 97 22 12
www.cowi.com

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Prepared	Lizzi Andersen, Carsten Skov, André Küter, Lene Schepper, Hans Gottberg Rømer, Anders Refsgaard, Magdalena Utko
Checked	Torben Kristiansen,
Approved	Carsten Skov

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1 Introduction

This prefeasibility study is part of the overall process related to the decision on placement and design of a repository for the Danish low and medium level radioactive waste primarily from the facilities at Risø. The prefeasibility study encompasses the preliminary design of a number of repository types based on the overall types set out in the "Parliamentary decision" together with a preliminary safety assessment of these repository types based on their possible placement in a set of typical Danish geologies.

The study has been carried out for Danish Decommissioning by COWI A/S in cooperation with Studsvik AB and Hasløv & Kjærsgaard A/S.

The basis for the prefeasibility study has been the "Parliamentary Decision for a final repository for low and medium level radioactive waste, November 2008" (Ministry of Interior and Health, 2008) together with a number of other reports and memos that together made up the background for the political decision to carry out the prefeasibility study, see references in Chapter 12. This study is a part of the overall prefeasibility study, which also encompasses a study regarding possible location of a repository (carried out by GEUS[1]), and a study concerning transport issues (carried out by SIS). These studies jointly form the basis of a further clarification and decision process leading to a final choice of repository site and design. Since specific locations are not yet decided upon, the present study is carried out based on generic geologies, topographies, etc.

The alternative repository concepts to be considered in this study are:

- A near surface repository (above or below the surface to a depth of 30 m)
- A near surface repository (above or below the surface to a depth of 30 m) in combination with a borehole.
- A medium deep repository placed at 30 to 100 m below the surface.

The repositories should be designed to have a life time of at least 300 years after initial filling.

The prefeasibility study comprises:

[1] Geological Survey of Denmark and Greenland

- Preliminary design of repository types and related waste conditioning
- Preliminary safety assessments
- Description and estimation of costs related to the suggested solutions

The preliminary safety assessments have encompassed assessments of likelihood and consequence of potential accidental incidents related to the different periods of the overall life time of the repository. The main activities include filling and operation, possible retrieval of waste and the passive period after closure of the repository. It has also encompassed an assessment of the potential long term impact on a reference person due to long term release of nuclides from the repository with time.

The basis for the preliminary safety assessments has been the suggested repository designs, including option for total or partial reversibility that is the possibility to retrieve some or all of the waste from the repository at a later time. In the "Parliamentary Decision" it was described that a large part of the waste is of a character that does not make it desirable to retrieve (e.g. mixed waste consisting of gloves and working clothes, etc. concrete from decommissioning, and tailings) also due to the low activity content. The special waste may be relevant to retrieve although the total amount is small and it has not yet been sorted with the aim of retrieval.

Another part of the basis for the preliminary safety assessments are a set of generic geologies:

- Fat, tertiary clay
- Moraine clay and similar clay types
- Limestone
- Rock (granite).

These geologies and the relevant receiving water recipients for groundwater potentially that are impacted by the repository are used in the preliminary safety assessment with parameters and conditions typical for Denmark. In accordance with the recommendations from the International Committee on Radiological Protection, ICRP, the possible impact is assessed for a so called *reference person*. The potential exposure routes for this reference person are also based on typical Danish conditions, including a typical Danish diet.

As part of the preliminary safety assessment, the variability of the relevant parameters is assessed together with the influence of this variability on the results. It should be noted that the existing knowledge and amount of data related to the different parameters varies, which causes some parameters to have a more uncertain basis than others. The fact that the safety assessment is carried out on generic locations also adds to the variability of the relevant parameters compared to the variability that will be present at an actual site.

It is thus important that supplementary safety assessments are carried out for the potential specific locations, where the actual parameters can be assessed, and that a better data basis also can be established for some of the less site specific data, where these are assessed to have a substantial influence on the result of the safety assessment.

Apart from impact from long term leaching of nuclides from the repository, also gaseous release of nuclides from the repository is assessed.

The safety assessment modeling has calculated consequences of releases until 10,000 years after the initial filling of the repository and the following spreading of these releases for up to 1,000,000 years. On this basis, supplementary estimates of potential impact from releases at a later stage due to nuclides still containing activity above the clearance criteria suggested by the International Atomic Energy Agency, IAEA, have also been carried out.

1.1 Content of this report

This report consists of three parts. Part I is the descriptive part containing information on the waste to be disposed of, the potential conditioning (packaging) possibilities for the waste before placement in a repository, the suggested preliminary design of the different repository types, and the suggested visual appearance of the repository. Part I comprises the chapters 2 to 5.

Part II is the assessment part. It contains an introduction to the concepts used in the preliminary safety assessment, which encompasses: the assessment of potential long term impact and the assessment of possible accidental incidents. The division of the preliminary safety assessment in to these two categories has several reasons. One is that the criteria to which impact is to be compared are different for the two types of impact, another is that while the possible variation in the long term impact is primarily due to the possible variation in the parameters influencing the impact, the impact from accidental incidents is governed by the probability of the occurrence of these incidents and the potential consequence of the impact, which calls for a different assessment approach.

Since the suggestions for packaging of the different waste types is a result of both types of assessments, part II also contains a description of these suggestions based on the preliminary safety assessments. Finally part II contains the costs related to the different types of repositories and the suggested packaging.

Part II encompasses the chapters 6 to 10.

Part III of the report contains the recommendations that are the result of the preliminary design and the preliminary safety assessment. These are divided into three subparts: General recommendations with respect to site selection etc., recommendations that are a result of the preliminary safety assessment (other than the suggestions for waste conditioning described in part II), and recommendations for the process forward including supplementary investigations. Part III is contained in Chapter 11.

At the end of the report, there is a reference list (Chapter 12) and a glossary (Chapter 13).

The main report is supplemented by a number of annexes (A to L) containing details about specific issues, data basis and more detailed results and outcomes of the prefeasibility study.

2 Waste to be disposed

The waste to be disposed of originates from a number of places and is of very varying type and activity. Part of it comes from the nuclear research previously carried out at Risø, while the rest comes from other Danish users of radioactive material such as the health sector, research institutes and the industry. The types of waste comprise:

- Compressed low level radioactive solid waste such as paper, plastic, working clothes glass and metal contaminated with radioactive nuclides
- Discarded equipment contaminated with radioactivity nuclides
- Residue from water treatment from The Waste Management Plant at Risø (evaporation residue contained in bitumen and ion exchange waste)
- Discarded radioactive sources (from the health sector, research and industry)
- Decommissioning waste from the nuclear facilities
- Special waste (will be further described in the following chapter).

Some of the waste already exists from both external sources and the decommissioning of some of the nuclear facilities already carried out at Risø. The rest of the waste will also come partly from external sources and from the decommissioning of the remaining the nuclear facilities at Risø.

Danish Decommissioning has provided available information about both the existing and the expected waste to be placed in a repository, see Chapter 12 for references. The waste has been classified in accordance with the recommendations given by the International Atomic Energy Agency (IAEA) and EURATOM with the aim of determining appropriate conditioning and storage before final disposal. Both IAEA and EURATOM have given recommendations regarding the classification of radioactive waste. In the Commission Recommendation of September 15, 1999 (EURATOM, 1999), three classes are given:

1) Transition radioactive waste: will decay during storage to levels within clearance.
2) Low and intermediate level waste (LILW): has a concentration of radioactive isotopes low enough to ensure that the heat development will be sufficiently low during disposal. This will depend on a specific evaluation.
3) High level waste (HLW): will generate significant thermal energy and consists mainly of nuclear fuel.

LILW is divided into two classes - short-lived and long-lived waste. Short-lived waste, LILW-SL, contains nuclides with half lives no longer than Cs-137 and Sr-90 (app. 30 years) and a limited concentration of long-lived α-emitters (less than 4,000 Bq/g in a single package and less than 400 Bq/g overall). Long-lived waste, LILW-LL, contains long-lived isotopes and α-emitters in higher concentrations than accepted for short-lived waste.

In IAEA[2] (2009) Figure 2.1 is presented as a suggested classification of radioactive waste. Classification can be carried out with different aims. It is envisaged that a detailed classification of all waste will be carried out before the conditioning prior to final disposal and before the final choice of disposal option is chosen (near surface repository, medium deep repository or borehole).

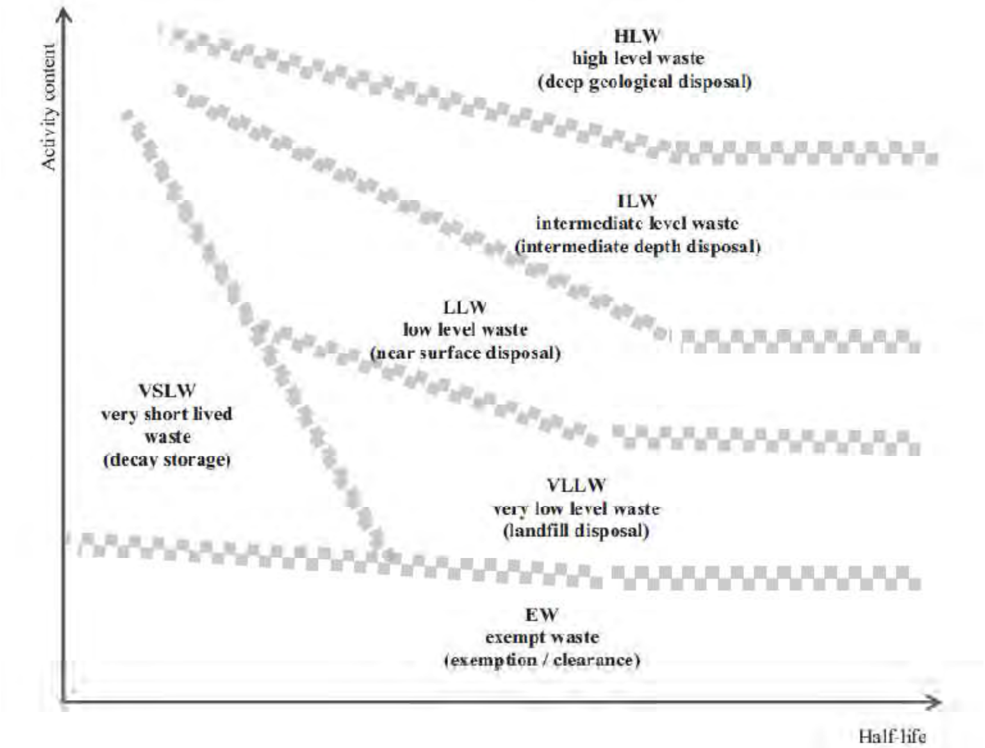

Figure 2.1 Schematic presentation of general classification of radioactive waste, IAEA (2009)

Short-lived nuclides (nuclides with a half-life less than 30 years) are with respect to possible long term impact expected to be of minor importance from a storage safety point of view since the repository will be designed to be intact for a long period of time (> 300 years). A nuclide with a half-life of 30 years will decay to 10 % of the initial activity in 100 years. After 300 years less than 0.1 % of the initial activity remains.

[2] International Atomic Energy Agency

Long-lived nuclides will in some cases not decay more than fractions of a percent in 300 years. For material containing significant amounts of long-lived isotopes, it is of great importance that the waste is correctly classified so that the long-term safety issues can be correctly addressed.

According to IAEA (2003), a waste characterization record should contain the following information pertaining to the waste:

- the source or origin
- the physical and chemical form
- the amount (volume and/or mass)
- the radiological characteristics (the activity concentration, the total activity, the radio nuclides present and their relative proportions)
- the classification in accordance with the international waste classification system IAEA (2009)
- any chemical, pathogenic or other hazards associated with the waste and the concentrations of hazardous material
- any special handling necessary due to criticality concerns, the need for the removal of decay heat or significantly elevated radiation fields, if present.

2.1 Waste types

The current waste stored at the Risø site is described in Danish Decommissioning (2009) and Danish Decommissioning (2002). In both reports the waste is classified as low level and intermediate level waste. The overview of the waste types given below is taken from Danish Decommissioning (2009).

Decommissioning waste

The decommissioning waste originates from the decommissioning of nuclear facilities at Risø. The facilities encompass, among others, research reactors, hot cells for material examination, waste treatment, and storage facilities. The radioactivity arises from both contamination and due to activation of materials. This waste includes both existing waste and expected waste from planned decommissioning activities in the future. Table 2.1 presents the distribution of the types of waste.

The present waste from the decommissioning of Danish Reactor 1 (DR1) makes up a total of 9 tonnes, where the majority is graphite and concrete (3.5 and 3.4 tonnes, respectively). The waste also consists of lead and cadmium (338 and 44 kg, respectively).

The present waste from decommissioning of Danish Reactor 2 (DR2) makes up a total of 166 tonnes where the dominant fraction is concrete (138 tonnes). The remaining waste mainly consists of metals such as aluminium, steel and lead. The amount of lead is 7 tonnes.

Low level waste

In most cases, low level waste (LLW) mainly contains short-lived nuclides. In the actual waste there is a significant amount of long-lived alpha activity. According to information from previous studies and Danish Decommissioning representatives, some drums contain sealed sources such as smoke detectors with Am-241 which is a long-lived alpha-emitting nuclide.

The average specific long-lived α activity is hundreds of Bq/g compared to the general clearance levels of 0.1 - 1 Bq/g for α-emitting nuclides in the European Commission Recommendations RP 113 (building rubble) and RP 122 (materials). Part of the waste packages classified as low level waste cannot fulfil the clearance criteria for millions of years and should be kept safe for a very long period of time.

The dominating short lived isotopes in drummed waste are C-60 and Cs-137. The very long term α-activity originates primarily from U-238 and U-234. Table 2.2 presents the distribution of types of waste.

Intermediate level waste

Intermediate level waste comprises both waste items with high short lived β-activity and spent fuel with the very long lived α-emitters U-238 and U-234. The waste requires special attention both during handling of the waste and from a long-term safety point of view.

Special waste

The nuclide inventory is dominated by nuclides in irradiated fuel which can cause high dose radiation rates if not properly packed and handled. The irradiated fuels have, in addition to the high activity of short-lived nuclides, a high content of long-lived alpha-nuclides. This waste should be treated individually and will require special attention, see Chapter 9 and 11.

Tailings and contaminated concrete

Tailings produced during experiments with extraction of uranium from uranium ore in the 1970's and 80' have been stored at Risø in large concrete basins covered by water. The concrete making up the basins has been contaminated by the activity originating in the tailings. Both tailings and concrete include considerable amounts of long lived α-emitters (although at low concentrations) and should therefore be kept safe for a very ling time.

The amounts and volumes of the different types of waste are summarised in Table 2.1 and Table 2.2 together with the type of radiation related to the decay of the radioactive nuclides present in this waste type.

The amounts etc. listed in these two tables do not include the tailings and the related concrete, which combined are numbered waste type 21. The overall amount of tailings is 1130 tonne. The radiation related to the tailings is of very low level, for some nuclides very close to the clearance level, but are all long-lived α-nuclides.

Table 2.1 Estimated waste amounts from decommissioning, approximate disposal volume, and activity in the waste per 1 June 2008.

Type	Waste	Weight/ units	Volume, conditioned (m^3)	Short-lived ß/γ (GBq)	Long-lived ß/γ (GBq)	Long-lived α (GBq)
Waste from decommissioning of DR1, DR2, and DR3						
1	Graphite	17 t	39	4,000	120	
2	Aluminium	17 t	75	20,400		0,7
3	Stainless steel and lead	345 t	732	66,600	18,000	1,5
4	Heavy concrete and concrete	1313 t	1,129	570	38,000	108
Waste from decommissioning of Hotcell						
5	Stainless steel, steel and lead	3 t	5			
6	Concrete	20 t	40			
7	Various components	3 t	5			
8	Secondary waste	100 drums = 20 t	57	3,000	1	160
	SUM		**2,082**			

Table 2.2 includes information about waste type 20, Heavy water. Danish Decommissioning has during the pre-feasibility study handled this type of waste separately and it will thus not be disposed of in a future repository. The waste type is therefore not included further in the preliminary safety assessment.

Table 2.2 Existing waste amounts, approximate disposal volume, and activity in the waste per 1 June 2008. "C" refers to container for "CC-blade". "A" is "A-bin".

Type	Waste	Weight/ units	Volume, conditioned (m^3)	Short-lived ß/γ (GBq)	Long-lived ß/γ (GBq)	Long-lived α (GBq)
Low-active waste						
9	Waste from wastewater treatment	1100 drums	920	1,800	0,5	130
10	Compacted waste and soil	4400 drums	1,100	2,600	0,6	170
Medium-active waste						
11	Waste from DR3	17 C + 40 drums	80	5,400	18,000	
12	Waste from Hotcell	180 drums+ 40A + various	430	33,000	147	1,300
13	Radioactive sources	18 drums + various	30	370,000	300	1,500
Special waste						
14	Approx. 20 larger sources	Various	35			1,000
15	1,2 kg irradiated, dissolved uranium	3 drums	5	4,000	9	400
16	12 kg irradiated fuel	20A	20	23,000	55	1,500
17	222 kg irradiated fuel	13A	45	730,000	5200	31,000
18	Core solution from DR1	3 bottles (flasker)	10	120	1	4
19	Non-irradiated uranium	2 t	60			50
20	Heavy water	0.1 t	3	5.7		
SUM			**2,735**	1,264,490	79,835	37,324

Heavy metals

Apart from the nuclides present in the waste it also contains: the following major amounts of metals:

- 2 tons of uranium
- 50 to 70 tons of lead
- 200 kg of cadmium
- 80 kg of beryllium.

2.2 Specific nuclides in the waste

Knowledge about the specific radionuclides and their activity level versus time forms an important part of the basis for the preliminary safety assessment in the pre-feasibility study.

The present description of the waste shall only be considered as preliminary and must only be used within this pre-feasibility study. The information available on types and amount of specific radionuclides is in many cases of a qualitative nature and not sufficient for a final classification.

The radionuclides in the waste are of the following origin:

- Radiation sources
- Nuclear fuel
- Neutron activated building materials and structures
- Operational waste from Risø's activities
- Contaminated building materials and structures
- Waste of external origin

Radiation sources

The waste includes radiation sources both originating from Risø's activities and from external waste generator. The radionuclides in these sources are primarily Co-60, Sr-90, Cs-137 and Am-241.

Nuclear fuel

The fuel in Risø's reactors was until the late 1970's high and thereafter medium enriched in U-235. The waste includes both non-irradiated and irradiated fuel.

Initially the nuclides giving the largest contribution to the radiation from the fission products are mostly due to very short lived isotopes, later the short lived Sr-90 and Cs-137 ($T_{½}$~30 years) are the main radioisotopes, being succeeded after 10^5 years by Tc-99 and the long lived actinides Np-237 and Pu-242.

Considering that the last Risø reactor DR3 was closed down in year 2000, the "age" of the waste today is 20 years or more. The very short lived nuclides have by now decayed, and at present and in the years ahead nuclides like Sr-90 and Cs-137 dominate. But the most critical nuclides for the long term risk will be the transuranic elements.

Neutron activated building materials and structures

Certain nuclides in the decommissioning waste from the Risø reactors have been activated by neutron radiation, and active nuclides are incorporated in the building materials and instruments. Among these, important nuclides are: Co-60 (from the stable alloy Co-59 in steel), Ca-41 (from stable Ca-40 in concrete), Fe-55 (from natural very long-lived Fe-54), Ni-63 (from stable Ni-62 in alloys and coating), Ba-133 (from natural Ba-132 in barite concrete).

Contaminated items and structures

Some of the waste consists of contaminated items and structures where the nuclides are located on the surfaces. The radionuclides may be any nuclide handled at the facilities or sent to Danish Decommissioning by external users.

Waste from external sources	This waste originates from the industry, hospitals and universities. The list of nuclides provided indicates the diverse kind of origin.
Tailings	Tailings originating from a uranium pilot plant at Risø. A fraction of U-238, U-235 and U-234 have been removed. The radionuclide content is close to the clearance levels.

In order to carry out the preliminary safety assessment and give recommendations as to the conditioning of the different waste types, it has been necessary to estimate in further detail, which nuclides are related to which waste types. This has been carried out as a part of the pre-feasibility study on the basis of the information about waste types and overall activity given by Danish Decommissioning. The basis for the estimates is primarily the information given in the following documents:

- Danish Decommissioning (2009e). The document includes a list of nuclides required to be considered in an assessment of the location of the repository: H-3 C-14 Ca-41 Fe-55 Co-60 Ni-63 Sr-90 Ba-133 Cs-137 Eu-152 Eu-154 Rn-222 U-235 U-238 Pu-239 Pu-241 Am-241.

- Danish Decommissioning (2010a). DD informs that the tables are of an early date and not complete. However, the information on specific nuclides may support the information retrieved from other sources.

- Danish Decommissioning (2010b), including a list indicating “Activity of tailings".

Radionuclides considered in the pre-feasibility study are listed in Table 2.3 to Table 2.5. The tables include some very short lived nuclides. This is because these nuclides contribute significantly to the indicated present total activity level of waste of external origin and they are required for calculation of nuclide distribution only, based on the information of the overall activity.

Waste of external origin includes other nuclides than indicated in the tables, some in small or trace amounts. All nuclides in considerable amounts are listed; nuclides with an activity (per 2008) of less than 1 GBq are omitted.

Based on the above, estimates have been made for nuclide contents for each of the 20 waste types. Details on this can be found in Annex A.

Table 2.3 Short lived β/γ-nuclides (half life < 30 years) considered in the preliminary safety assessment

Element	Nuclide	Half-life in years	Decay radiation	Daughter
Tritium	H-3	12.3	β	He-3 stable
Cobalt	Co-60	5.27	β & γ	Ni-60 stable
Selenium	Se-75	0,33	EC & γ	As-75 stable
Strontium	Sr-90	29.1	2xβ	Y-90 → Zr-90 stable
Caesium	Cs-137	30	β & γ	Ba-137 stable
Barium	Ba-133	10.7	EC & γ	Cs-133 stable
Europium	Eu-152	13.3	β & γ	0.72 Sm-152 stable
			β & γ + 3x α	0.28 Gd-152 → Sm-148 → Nd-144 → Ce-140 stable (all very long half lives)
	Eu-154	8.8	β & γ	Gd-154 stable
Iridium	Ir-192	0.20	β & γ	Pt-192 stable Os-192 stable
Plutonium	Pu-241	14.4	β	Am-241, neptunium series

Table 2.4 Long lived β/γ-nuclides (half life > 30 years) considered in the preliminary safety assessment

Element	Nuclide	Half-life in years	Decay radiation	Daughter
Carbon	C-14	5,730	β	N-14 stable
Calcium	Ca-41	140,000	EC & γ	K-41 stable
Nickel	Ni-63	96	β	Cu-63 stable
Technetium	Tc-99	211,000	β	Ru-99 stable
Silver	Ag-108m	418	EC & γ	Pd-108 stable
Samarium	Sm-151	90	β	Eu-151 stable

Danish Decommissioning (2010) also indicates the β/γ-nuclides: Cl-36, Ni-59, Nb-94, Mo-93 and Cd-113m, however these nuclides are only present in trace amounts and not considered in the preliminary safety assessment.

Table 2.5 α-nuclides considered in the preliminary safety assessment

Element	Nuclide	Half-life in years	Decay	Daughter
Radon*)	Rn-222	3.82 days	5 x α 4 x β	Po-218→Pb-206, radium series
Radium	Ra-226	1,600	5 x α 4 x β	Rn-222→Pb-206, radium series
Thorium	Th-230	75,380	6 x α 4 x β	Ra-226→Pb-206, r adium series
	Th-232	$14 \cdot 10^9$	6 x α 4 x β	Ra-228→Pb-208, thorium series
Uranium	U-234	246,000	7 x α 4 x β	Th-230, →Pb-206 radium series
	U-235	$704 \cdot 10^6$	7x α 4 x β	Th-231→Pb-207, actinium series
	U-238	$4{,}470 \cdot 10^6$	8 x α 6 x β	Th-234→Pb-206, radium series
Plutonium	Pu-238	87.7	8 x α 4 x β	U-234→Pb-206, radium series
	Pu-239	24,110	8 x α 4 x β	U-235→Pb-207, actinium series
	Pu-240	6,500	8 x α 4 x β	U-236→Pb-208, thorium series
Americium	Am-241	432	8 x α 4 x β	Np-237→Pb-209, neptunium series
Curium	Cm-244	18.1	9 x α 6 x β	Pu-240→Pb-208, thorium series

*) Rn-222 is a daughter in the radium series. The lists and figures in this report in general only indicate the parent nuclide, e.g. Ra-226 (in sources), Th-230, U-234, U-238, Pu-238 etc. Daughters, e.g. gaseous Rn-222 will be evaluated as part of the preliminary safety assessment.

** Danish Decommissioning (2001a & b) also indicates the α-nuclides: Ac-227, Pa-231 and Np-237, however these nuclides are only present in trace or very small amounts.

2.2.1 Distribution of nuclides on waste types

Figure 2.2 indicates the estimated activity (GBq) of all the radionuclides versus time (from 2008) in the entire amount of waste to be sent to the repository.

The figure shows that different nuclides will dominate as time develops due to differences in decay rate.

10-30 years	10-30 years from now, the overall activity will be in the order of 10^6 GBq. The dominating nuclides are the β/γ-nuclides H-3, Co-60, Ni-63, Sr-90, Cs-137 and Eu-154. Further the level of activity from the α-nuclides Pu-238 and Am-241 will be significant.
100 years	100 years ahead, the overall level of activity will be in the order of 10^5 GBq. Dominating nuclides will still be Ni-63, Cs-137 and Am-241. The short lived Co-60 will have decayed to an insignificant level of activity.
300 - 1,000 years	300 - 1,000 years ahead, the overall level of activity will be in the order of 10^4 GBq. The dominating nuclides will now be C-14, Ni-63, Pu-239, Pu-240 and Am-241. By now Sr-90 and Cs-137 will have decayed.
3,000 - 10,000 years	3,000 - 10,000 years ahead, the overall level of activity will be in the order of 10^4 -10^3 GBq. The dominating nuclides will be C-14, Tc-99, Pu-239 and Pu-240. By now Ni-63 and Am-241 will have decayed.
10,000 years and later	100.000 years ahead, nuclides of U (in particular U-238 and U-234) and their daughters accounts for a basic activity level in the order of 10^3 GBq. This level will be maintained for billions of years.

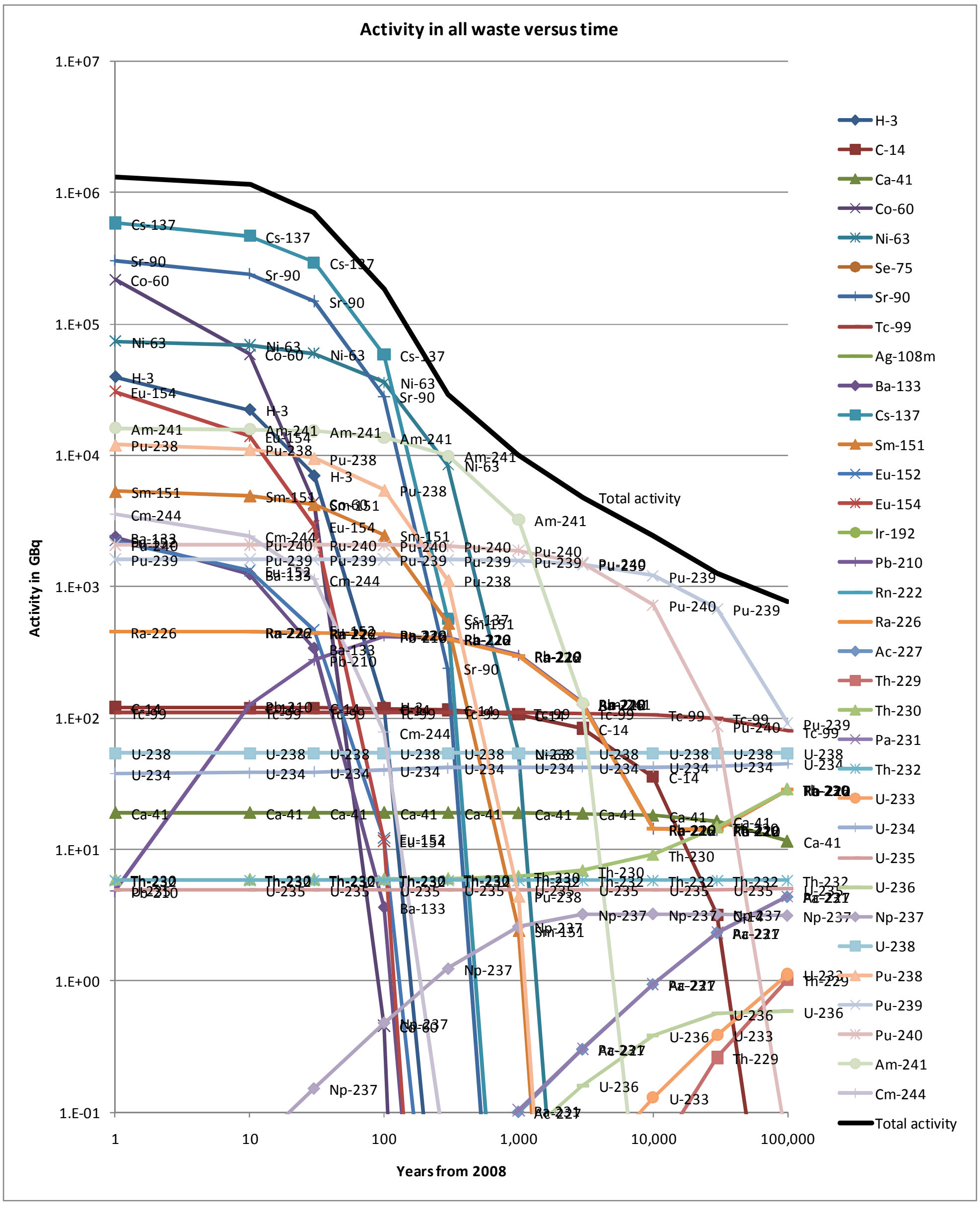

Figure 2.2 Estimated activity in GBq versus time (from 2008) for the entire amount of waste

3 Conditioning of waste

Before placing the waste in a final repository, the waste has to be packed in drums, containers, etc. Furthermore, different fill materials are necessary to ensure that radioactive nuclides are retained within the packages for a sufficient amount of time to render the material safe. This process has of course already taken place for the existing waste from decommissioning etc. in order to store the waste until final disposal.

In the prefeasibility study, initial assumptions about packing of the waste were required as a basis for the preliminary design of the repository and as a starting point for the preliminary safety assessment. As a result of the preliminary safety assessment, recommendations can be made as to how the waste is best packed and the drums and containers filled (the overall term for this process is conditioning). In this chapter the possible containers and the possible fill materials are described, together with the estimates made with respect to the necessary overall repository volume that is the basis for the preliminary design. The recommendations with respect to conditioning of the waste that are the result of the safety assessment are collected in Chapter 9.

Further, the chapter includes the characteristics of the different types of fill and backfill material that may be used at the conditioning, packing and disposal of the waste packages. Characteristics include both physical and chemical properties.

3.1 Waste packages

Certain, specified types of containers are used for the temporary storage of radioactive waste at the Risø area. These different types of containers are described in Appendix B.

In the pre-feasibility study, some assumptions and simplifications have been necessary, partly because the final waste amounts are not yet known, partly because final classification of the waste has not yet been carried out. This classification will form the basis of the final choices with respect to relevant conditioning. In accordance with the instructions of Danish Decommissioning in the Scope of Work for the prefeasibility study, the preliminary design and the safety assessment has been based on a relatively limited number of typified packing options.

Furthermore, mixing of different categories of waste within one package may be a feasible way to optimise packing both from a logistic point of view and to allow filling of containers within the stipulated maximum gross weight for the packages. In some special cases, high density material (lead or steel) could be combined with waste with high dose rate for shielding purposes. However, within this pre-feasibility study, different waste types are kept separately, in order not to complicate the safety assessment unnecessarily. These issues can be further evaluated at a later stage of the decision process.

To the extent possible and acceptable, containers with waste conditioned for interim disposal should be taken directly for final disposal without re-packing. However, for some of the already conditioned waste, this is not possible. Based on experience from other nuclear installations, reconditioning of such waste has to take place with great care and after special assessments.

There will be waste generated from nuclear facilities not yet decommissioned (e.g. DR3), and there is waste not yet conditioned for final disposal. Waste packages and volumes have been estimated based on the available information.

In principle, it is of advantage not to use too many different types of containers for the final disposal. Containers of the same shape and size will be easier to handle and to pile than various types and sizes. Also the cost to qualify a package type for the repository should be considered.

In general, standard containers or other packages already qualified for transport or disposal (elsewhere) could be cost-effective to use. ISO containers used elsewhere are normally classified as Industrial Package Type 1 (IP-1) or IP-2 according to the European agreement about transport on roads (European Agreement concerning the International carriage of dangerous Goods by Road) Those containers are classified for transport of radioactive material (waste) with a dose rate of up to 2 mSv/h on the outer surface of the container.

Although no exact inventory of the combination of nuclides in the waste is available, it is likely that for most of the waste categories the packages will not exceed the criteria for IP-2 packages.

The waste does not necessarily have to be packed for transport in accordance with ADR transport regulations. A few special waste types, such as irradiated uranium fuel and some of the sources, need to be transported in special transport packages and other waste may also be packed in special transport containers if found relevant. Recommendations on the choice of transport containers is assumed to be included in the transport pre-feasibility study carried out by Statens Institut for Strålebeskyttelse (SIS).

For the purpose of the conceptual design and the preliminary safety assessment, it is assumed that the four types of containers described below are used for the final disposal.

3.1.1 ISO containers

The ISO freight containers are available as Industrial Package Type 1, Type 2 and as Type A containers according to ADR. The different types are allowed to carry different amounts of radioactivity. However, the outside dimensions are the same.

For the current report and for the initial safety analyses, calculations are made for the ISO containers (10ft, half height, Industrial Package Type 2)

3.1.2 Steel drums

The existing already-packed drums are of slightly different design, with or without inner lining of concrete. Some drums have been placed in 280 litre rescue drums due to corrosion. However, in this study all drums have been considered to be 210 litre drums with concrete lining.

3.1.3 Special steel containers

Special, thick-walled steel containers have been designed and used by Danish Decommissioning (DD) for the decommissioned waste. In general, the ISO containers will be used where possible, and the special containers will be used when it is of advantage from a final repository point of view or due to high dose rates. The steel containers may be loaded in Type 2 or Type A ISO containers for the transport.

These containers may also be considered for packing of long-lived high radioactive waste with or without inner packages for scenarios, where no borehole will be used.

3.1.4 Canisters

So-called canisters are supposed to be used for packing of waste for final disposal in a deep borehole. A canister of the type, which is intended to be used for parts of the actual waste, is a cylinder-shaped container made of steel or cast iron with a wall thickness of at least 10 cm. The maximum outer diameter is 50 cm and the maximum height 100 cm. For further details on canisters, see Danish Decommissioning (2009). It is assumed that three canisters will be installed in the borehole at the same time. The three canisters are to be installed in a frame of steel profiles, which can be lowered into the borehole by means of a wire provided with a hook for remote release from the terrain level. With this arrangement, three canisters will occupy a length of 4 m in the borehole (for further details on this arrangement, see Danish Decommissioning (2008).

Canisters of this type or similar may also be used as inner package for repository alternatives without deep boreholes.

3.2 Number of containers and volume of waste for disposal

The estimated total number of different types of waste containers as described above is summarised in Table 3.1. Details on the proposed handling and packing of individual waste types are presented in Chapter 9.
In the future, waste generated by the health sector, research institutes and the industry are to be disposed of in the repository. This future waste generation is expected to correspond to 2 tons or 8 m^3 conditioned and packed waste per year (Ministry for Health and Prevention, 2008). It is assumed that the future waste will be packed in 210 l steel drums that may be taken directly for final disposal or alternatively further packed in steel containers, depending on the disposal solution.

For handling reasons, for the design of the proposed repository types, a horizontal distance of 300 mm is assumed between each container at the medium deep repository types. At above surface or near surface repositories, only 100 mm distance is assumed due to easier handling by means of for example a fork-lift. The drums are proposed to be placed in a horizontal close pack pattern. Three canisters together are assumed to occupy a length of 4 m in a borehole.

Table 3.1 Current and estimated waste, conditioned and packed for final disposal - No. of containers

Type	Waste	Steel containers	ISO containers (10ft)	210 l drums	Canisters
1	Graphite	10			
2	Aluminium		10		
3	Steel, stainless steel and lead	3	101		
4	Heavy concrete and concrete		230		
5	Stainless steel, steel and lead		2		
6	Concrete		6		
7	Various components		3		
8	Secondary waste	20			
9	Waste from wastewater treatment			1100	
10	Compacted waste and soil			4400	
11	Waste from DR3 (incl. TSP & TSR[3])	15	16		
12	Waste from Hot Cell	70			40
13	Radiation sources	3			
14	Approximately 20 larger sources				3
15	1.2 kg irradiated, dissolved uranium	6*			6*
16	12 kg irradiated fuel	7 *			14 *
17	222 kg irradiated fuel	9 *			9 *
18	Nuclear solution from DR1	6 *			6 *
19	Non-irradiated uranium	4			
21	Tailings		80		
22	Contaminated concrete		70		
	TOTAL	**125-153**	**518**	**5500**	**43 - 78**

* Note: Either steel containers or canisters. The type of container depends on the solution chosen for final disposal.

Based on the assumptions mentioned above and the number of containers estimated, the theoretically necessary repository volume has been calculated, as presented in Table 3.2. It should be noted that the small, future annual waste generation has not been taken into consideration. Thus, the repository must be designed with a slightly larger volume.

[3] Will probably be placed in special containers

Table 3.2 Estimated required volume of different repository types

Container type		Above Surface		Near Surface		Medium Deep		Bore hole	
Steel containers	No.	153		153		153		0	
	Net volume (m^3)	662		662		662		0	
	Outer void (m^3)	78	(1)	78	(1)	248	(2)	0	
	Volume incl. outer void (m^3)	741		741		910		0	
ISO-containers	No.	518		518		518		0	
	Net volume (m^3)	4,911		4,911		4,911		0	
	Outer void (m^3)	373	(1)	373	(1)	1,160	(2)	0	
	Volume incl. outer void (m^3)	5,284		5,284		6,071		0	
210 ltr. drums	No.	5,500		5,500		5,500		0	
	Net volume (m^3)	1,155		1,155		1,155		0	
	Outer void (m^3)	363	(3)	363	(3)	363	(3)	0	
	Volume incl. outer void (m^3)	1,518		1,518		1,518		0	
Canisters	No.	0		0		0		78	
	Net volume (m^3)	0		0		0		15.3	(5)
	Outer void (m^3)	0		0		0		69.5	(6)
	Volume incl. outer void (m^3)	0		0		0		84.7	
Total waste	Net volume (m^3)	6,728		6,728		6,728		15.3	
	Outer void (m^3)	814		814		1,771		69.5	
	Add. outer void (m^3)	0		543	(4)	611	(4)	0.0	
	Total outer void (m^3)	814		1,357		2,383		69.5	
	Total volume (m^3)	7,542		8,085		9,111		84.7	
	Total length (m)							104.0	

(1): 10 cm space between containers, horizontally
(2): 30 cm space between containers, horizontally
(3): Drums packed in close pack pattern, horizontally
(4): 0.4 m void above 4 containers
(5): 196 l per canister
(6): 0.686 m diameter x 4 m per every three canister

3.3 Fill and backfill material options

In the pre-feasibility study, distinction is made between *fill* to be used inside the containers and *backfill* to be used in between (outside) the containers. The same material may in some cases be used as fill and in other cases as backfill. For this reason, both types of material are described and discussed in this chapter.

3.3.1 Purpose of using fill and backfill materials

Fill and backfill materials may be used for several purposes. One purpose is to fill in cavities inside containers and in between containers to ensure the physical stability and avoid settlements or collapse within or around the repository. It must be easy to place or inject fill materials into the waste containers and backfill materials in between the containers. Pourable materials like concrete, granular material like bentonite, sand or gravel may fulfil the purpose.

The backfill material can also be part of the barrier system to reduce dispersion of radioactive or toxic compounds from the repository into the surroundings. This can be obtained by using materials with an ability to establish low-permeable barriers between the waste and the surroundings. It is important to avoid voids within the repository so that the flow of water can be minimized. Furthermore, the backfill material can retain the compounds for some time, which will let the radioactive nuclides decay before they are released to the surroundings.

Fill and backfill materials can be used as shielding of high gamma radiation, which can be important to keep the doses to the personnel as low as possible during storage and handling of the waste packages and during operation of the repository.

3.3.2 Fill and backfill materials and their characteristics

Different, desired properties of different types of fill and backfill material include the following:

- The materials must have a low friction angle in order to easily fill in the cavities between waste items when poured into the container.
- The materials must be non-aggressive with respect to possible corrosion of the containers.
- It is an advantage, if the material is manufactured from well known, easily accessible and relatively cheap base materials.
- It is an advantage, if the material is able to ensure a high pH in order to moderate any further corrosion of the waste items and to expand the lifetime of the waste containers.
- It is an advantage, if the material is able to retain radionuclides.
- It is an advantage, if the material to a certain degree is able to allow escape of gas from degradation of the waste, thus ensuring a gradual release of pressure and a lesser likelihood of premature breakage of the containers.
- It is an advantage, if the material allows for a possible increase in volume of the corroding units.

Within the pre-feasibility study, the following types of fill and backfill materials have been investigated:

- Cement-calcium granulate
- Concrete
- Bentonite
- Sand and gravel.

The main characteristics of the different materials are summarized in Table 3.3. In addition to the hydraulic conductivities, etc. the influence of the retention properties of the materials (usually expressed by K_D values) for the release of substances into the environment surrounding the repository has to be taken into account, when carrying out a safety assessment. The properties of the different possible fill and backfill materials are further detailed in Annex B.

Table 3.3 Summary of fill and backfill characteristics

Characteristics	Cement-calcium granulate	Concrete	Bentonite	Sand and gravel
Physical parameters: Density Porosity Permeability	Density 2.0 ton/m^3 (after mixing) The porosity has not been tested. Hydraulic conductivity not tested	Density 2.0-2.5 ton/m^3 depending on type of aggregates. Effective porosity very low, while the total porosity will depend on the formulation of the concrete. Very low hydraulic conductivity $K \leq 10^{-10}$ m/s	Density 1.1 ton/m^3 (pellets) up to 2.0 ton/m^3 (very compacted) Effective porosity very low after swelling, while the total porosity is ≥30%. Very low hydraulic conductivity $K \leq 10^{-10}$ m/s when moist Dry bentonite will shrink and become more permeable, but will expand again when water is reintroduced	Density 1.7-1.8 ton/m^3 (dry). Porosity approximately 30%. High hydraulic conductivity $K \geq 10^{-4}$ m/s
pH and corrosivity	High pH level Not corrosive to steel	High pH level Not corrosive to steel with proper selection of aggregate minerals.	Neutral pH level Not corrosive to steel	Neutral pH level Not corrosive to steel if composed of silicate minerals
Installation	Easy to install inside containers, but not suited for installation outside in gaps between containers	Easy to install inside containers and suited for installation outside in gaps between containers.	Bentonite pellets can be installed both in containers and outside in gas between containers.	Easy to install both in containers and outside in gaps between containers.
Stability after installation	After absorption of water, the material achieves a structure like semi cured concrete. The long term stability is not yet known.	Very stable after installation both in containers and outside in gaps between containers.	Very stable after installation both in containers and outside in gaps between containers.	Loose, and will fill voids created by corrosion of the waste containers. The total height of containers might subside after corrosion.
Retention of radionuclides	Laboratory tests suggest that the material has a high capacity for retention of selected radionuclides	Nuclides will be retained probably mainly due to co-precipitation.	Nuclides will be retained due to sorption to the clay minerals..	Nuclides will be retained due to sorption and precipitation but to a lesser degree.
Other comments	The materials properties are primarily known from tests. Long term experience is not available	-	The swelling pressure must be taken into account when bentonite is packed in and around the waste containers.	-

3.3.3 Combinations of fill and backfill material

The possible combinations for the use of the above materials as fill inside containers and backfill in between containers are listed in Table 3.4.

Table 3.4 Possible combinations of fill and backfill material.

Combination No.	Fill inside containers	Backfill between containers	Comments
1.1	Cement-calcium granulate	Concrete	This combination can only be used for irreversible solutions
1.2	Cement-calcium granulate	Bentonite	Can be used for both reversible and irreversible solutions
1.3	Cement-calcium granulate	Sand	In general, sand is considered less feasible than e.g. bentonite due to its high permeability
2.1	Concrete	Concrete	This combination can only be used for irreversible solutions
2.2	Concrete	Bentonite	Can be used for irreversible solutions[4]
2.3	Concrete	Sand	In general, sand is considered less feasible than e.g. bentonite due to its high permeability
3.1	Bentonite	Concrete	This combination can only be used for irreversible solutions
3.2	Bentonite	Bentonite	Can be used for both reversible and irreversible solutions
3.3	Bentonite	Sand	In general, sand is considered less feasible than e.g. bentonite due to its high permeability
4.1	Sand	Concrete	Aluminium depassivates in alkaline media and there is a risk of gas formation (Jones, 1996, Pourbaix, 1974 and Revie, 2000). Hence, concrete, cement-calcium granulate, or any other high pH backfill material, must not be used in containers containing aluminium. Although sand is very permeable, it can be considered as a fill material option for aluminium waste, as sand provides an environment with a low corrosion tendency for aluminium. The same holds for other pH neutral backfill materials.
4.2	Sand	Bentonite	
4.3	Sand	Sand	

[4] If the waste is decided moved for disposal in another setting, this option may be considered reversible

For the reasons mentioned in Table 3.4, the combinations including sand (i.e. combination 1.3, 2.3, 3.3 and 4.3) are in general not considered feasible due to the high hydraulic conductivity of sand. However, the exemption from this rule is the choice of fill material at the conditioning or packing of aluminium waste, where sand may be used in order to reduce the corrosion tendency of aluminium (combinations 4.1 and 4.2).

Based on the assessments mentioned in Table 3.4, the combinations of backfill material mentioned in Table 3.5 are considered the most feasible and have been used in the preliminary safety assessment.

Table 3.5 Combinations of fill and backfill material considered in the preliminary safety assessment.

Combination No.	Fill inside containers	Backfill in between containers	Comments
1.1	Cement-calcium granulate	Concrete	This combination can only be used for irreversible solutions
1.2	Cement-calcium granulate	Bentonite	Can be used for both reversible and irreversible solutions
2.1	Concrete	Concrete	This combination can only be used for irreversible solutions
2.2	Concrete	Bentonite	Can be used for irreversible solutions[5]
3.1	Bentonite	Concrete	This combination can only be used for irreversible solutions
3.2	Bentonite	Bentonite	Can be used for both reversible and irreversible solutions
4.1	Sand	Concrete	Only relevant at conditioning / packing of aluminium waste.
4.2	Sand	Bentonite	

Conclusions based on the preliminary safety assessments with regard to the use of fill and backfill materials at the packing and disposal of the individual waste types are presented in Chapter 9.

[5] If the waste is decided moved for disposal in another setting, this option may be considered reversible

4 Possible repository types

4.1 Introduction

The purpose of this chapter is to summarise the conceptual designs of different repository types suggested by the consultants. In this connection background information is given concerning the required volumes of the different repository types, geochemical impacts and durability, groundwater lowering and construction methods, general operation and monitoring aspects, and the general arrangement of the repository. The different repository types are introduced providing more details on the type specific layouts, construction methods, operation, closure, extension possibilities, monitoring and opening/emptying.

The included conceptual designs are based on a large number of assumptions, which have been established partly by Danish Decommissioning (as described in the terms of reference for the pre-feasibility study) and partly by the consultant (as described in different working reports, prepared during the initial phases of the study).

4.2 Design considerations

4.2.1 Geochemical impacts and durability design

This section provides the general background and recommendations for the durability design of the repositories with focus on concrete as main engineered barrier.

General background

The structural integrity and durability of the engineered barriers of the repository have a major influence on the expected retention of radionuclides, etc. Any long-term scenario for a repository must include present and future exposure and geotechnical conditions (e.g. future climate changes, thermal variations and differential soil settlements) as well as detailed considerations on the long-term durability of the construction materials, in particular for the barrier structures.

For all repository solutions, except for the above surface and the borehole repository, the main engineered barrier structures are consisting of concrete. Conventionally reinforced concrete, i.e. use of reinforcing bars, is only an option in certain environments, due to the risk of reinforcement corrosion.

Direct concrete deterioration is an important parameter in terms of durability of concrete structures, but reinforcement corrosion is the most frequently observed deterioration mechanism and just as important. In connection with reinforcement corrosion, concrete cracking plays a decisive role.

Concerning cracking, it is important to note that cracks are not only a decisive parameter in terms of concrete durability, but also play the most important role in connection with permeability. However, it should be emphasised that concrete cracks (also with a width of less than a millimetre) can increase the permeability of the concrete by several orders of magnitude. Although durability and structural design of the repository should aim at avoiding cracking of the concrete barriers at all costs, it is difficult to assess, if, when, and to what extent cracking of the concrete barriers should be taken into account.

Recommendations

The following (minimum) recommendations are given for the durability design of concrete barrier structures:

- If conventional (bar) reinforced concrete is used to construct the external repository structures (in particular, the barriers), it has to be assured that the environment in its vicinity is "chloride free" (i.e. the chloride concentration shall be negligible in terms of steel corrosion).

- If the requirements of the above bullet point are fulfilled, carbonation induced corrosion is considered to be the decisive modelling parameter in terms of reinforcement corrosion.

- If reinforced concrete is used to construct the external repository structures (in particular, the barriers), it has to be assured that there is no significant source for direct electrical currents in its vicinity.

- Concrete design and execution shall address all direct concrete deterioration mechanisms. It is expected that most deterioration mechanisms, i.e. sulphate attack, delayed ettringite formation, alkali-silicate and alkali-carbonate reactions and early age cracking, can be controlled this way, but with regard to the long service life of the repository leaching cannot be entirely prevented.

- For repositories more than 10 m below ground level, circular cross sections are preferable, as these structures are mainly in compression (hoop stresses).

- Even if structurally designed to remain uncracked in the service state, all concrete barrier structures shall be equipped with a suitable membrane on the outside.

4.2.2 Groundwater lowering methods

This section summarises the general background to the methods applicable for groundwater lowering.

General background

Groundwater lowering is normally required for excavation and construction of underground works located below the ground water table.

Only for relative shallow underground works, where the excavation can be performed from the surface, is it possible to do the excavation in wet conditions and construct the base slab with tremie concrete. Anchors required for uplift will then typically have to be constructed from a barge placed in the pit.

For the repositories at intermediate depth it is considered necessary to keep the excavation as dry as possible during construction. Since using the natural barriers against ground water flow is an integrated part of the strategy for the repository, the permeability at the relevant locations will be relatively small.

Limestone and rock may have fissures or flow zones, which could be problematic. Such zones will have to be grouted up, probably with an ultrafine cementitious grout.

Clay formations have themselves very low permeability and can as such not be improved by grouting, neither by the ultrafine cementitious grouts or the even finer chemical grouts available.

In sand/gravel formations, improvements can be made by grouting with still finer grouts, however it is considered more feasible to construct relative water tight cut-off walls around the pit, going into the deeper lying more impermeable geologies. But even with these precautions, some water will enter the excavation pit if nothing else is done.

Water actually entering the pit will easily be mixed up with fine grains due to the activities inside the shaft and water with large amounts of sediments is hard to clean. Settlement basins and cyclones may very well be required. Hence it is normally tried to catch as much water as possible in pumping wells below or next to the excavation.

In order to minimise the water being handled, it has been customary to pump water from wells situated inside the shaft. The water is pumped to a surface treatment plant capable of cleaning the water to a degree that it can either be discharged or re-infiltrated into the aquifer.

The cleaning requirements are dictated by environmental considerations and will differ for the various discharge possibilities at the location in question. Discharged to public sewer systems is costly and if the deficit in groundwater leads to unacceptable groundwater lowering in the surroundings, supplementary infiltration with city water may be required - this is equally expensive.

An alternative scheme may have more traditional retaining walls around the excavation and then at a somewhat larger periphery to have water barriers, e.g. in the form of diaphragm walls to minimise any flow toward the pit. Groundwater lowering can then be done in a closed system, where water is pumped up in the area between the retaining walls and the barrier walls and re-infiltrated directly back into the aquifer outside the barrier wall.

As the presence of water is also a nuisance to the construction works, it is in everybody's interest to keep the water ingress as small as possible by choosing appropriate construction methods. Section 4.2.3 describes the construction methods proposed for the repositories at intermediate depth and briefly discusses their functionality in relation to ground water control.

For the final repositories it is assumed that (at least in the filling period) the repositories are kept dry at the inside, whereas any groundwater lowering in the surroundings preferably has ceased. The cost of running a permanent groundwater lowering system would be very high. The system would have to be constantly overlooked, there would be regular and extraordinary maintenance cost, boreholes need to be cleaned or replaced from time to time. There would probably have to be regular reporting to the authorities. Any treatment plant would also have to be located inside its own house with heating facilities for the winters (cheaper than establishing tents or the like every year).

The expected cease of ground water lowering means that there will be a significant uplift force acting on the structure.

The most effective way of counteracting this uplift force is by ensuring adequate dead load of the structure plus adequate friction capacity between the structure and the surround ground. The construction methods and the layout for the 30 - 100 m deep shafts aim at this.

The circular shape ensures a 'closed' interface between the structure and the ground, as the inward movements of the structures acting in hoop is minimal. The surface of the cut-off structures suggested is for all methods relatively rough and thereby ensures a certain friction capacity.

Tension piles may be considered as a supplementary measure, however the piles have to be relatively deep and large in numbers, if a significant ground volume below the shaft shall be mobilised for uplift resistance. For temporary situations pre-stressed ground anchors may be applied, but due to the corrosion risk, they are normally not relied upon for long term behaviour.

4.2.3 Cut-off and excavation methods

This section summarises the general background to cut-off and excavation methods applicable for near -surface and medium deep repositories, i.e. repositories down to a depth of 100 m below ground level.

General background

The cut-off structures have to ensure stability of the excavation/shaft/cavern during construction, in the operational period as well as in the closed permanent stage. Thus, cut-off structures serve as temporary and permanent support.

By applying circular shaped shafts, the structures are mainly in compression (hoop stresses).

For the purpose of this investigation it is also assumed that groundwater might be present to a degree that adequate sealing off is required at each stage, i.e. the construction, the operation and the closed repository.

For the construction phase adequate water tightness can be obtained by sheet pile, secant pile or diaphragm walls in soft soils. In limestone and rock a combination of ground treatment, temporary drainage and sprayed concrete linings is sufficient.

The above mentioned measures do not ensure complete water tightness. Hence, to avoid water running into the repository during operation it is envisaged that a final in-situ cast base slab and inner lining will be constructed inside the temporary cut-off structures mentioned above.

The permanent inner lining will also be in hoop stress since water pressure will built up between the temporary cut-off structure and the permanent structure.

The base slab will have to withstand significant water pressures and preferably the inner cellular wall structure should support the slab.

In the following sections the most important construction methods are briefly described.

Sheet pile walls

Sheet pile walls are retaining walls usually used in soft soils and tight spaces. Sheet pile walls are made out of steel, vinyl or wood planks which are driven into the ground. Only steel pile walls, see Figure 4.1, are considered as a reasonable option for cut-off structures in connection with the repository.

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Figure 4.1 Construction of sheet pile wall. From Associated Pacific Constructors(2010).

Taller sheet pile walls will need a tie-back anchor ("dead-man" or ground anchors) placed in the soil at a distance behind the face of the wall, that is tied to the wall, usually by a cable or a rod. Anchors are placed behind the potential failure plane.

The maximum construction depth for sheet piles without anchorage is around 5 m. For sheet piles with anchorage a maximal construction depth of 12 - 15 m is considered reasonable.

Secant pile walls

Secant pile walls are retaining walls constructed by overlapping circular piles, typically with a diameter up to 1.2 m. Different layouts exist, i.e. every second pile reinforced, every pile reinforced, partly reinforced soft hard, and a fully reinforced hard wall, see Figure 4.2.

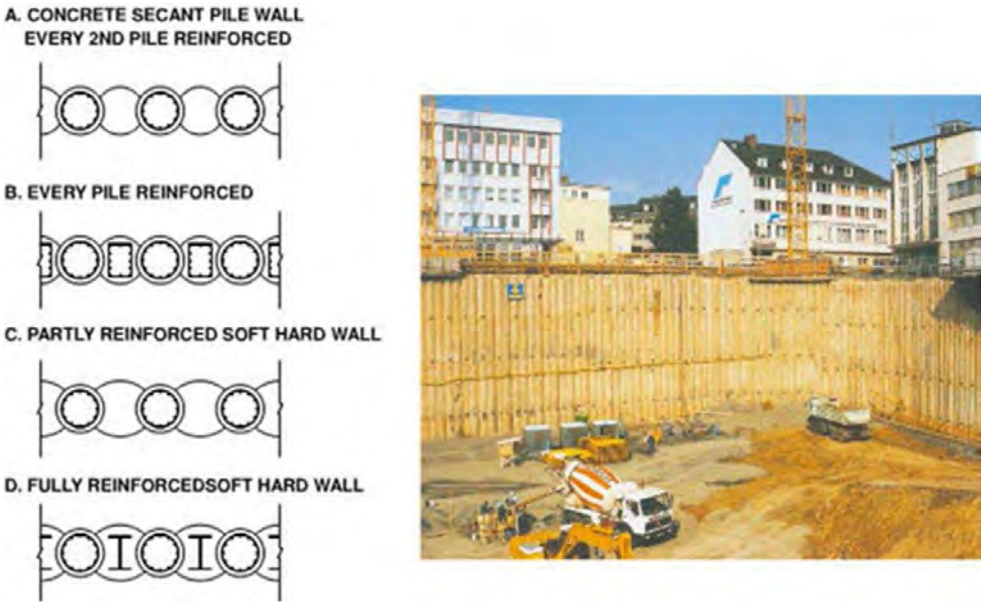

Figure 4.2 Construction of secant pile wall. From Bauer Group (2010).

During excavation a steel casing is lowered simultaneously into the bore hole in order to stabilize the sides. Furthermore, a positive water head is maintained inside the casing to avoid stability problems relating to water currents.

The construction is done by first excavating every second pile (the female piles) and subsequently remaining piles (the male piles).

The verticality of the secant piles can be kept at 1:200. Maximum construction depth is in the order of 25 m. If done in areas with limestone, secant piles typically extent 2-3 m into the limestone to create a sufficiently safe cut-off of groundwater.

Diaphragm walls

Diaphragm walls (aka slurry walls) are retaining walls constructed in panels of typically 3 - 7 m width and a thickness between 0.6-1.5 m. The construction method is illustrated in Figure 4.3.

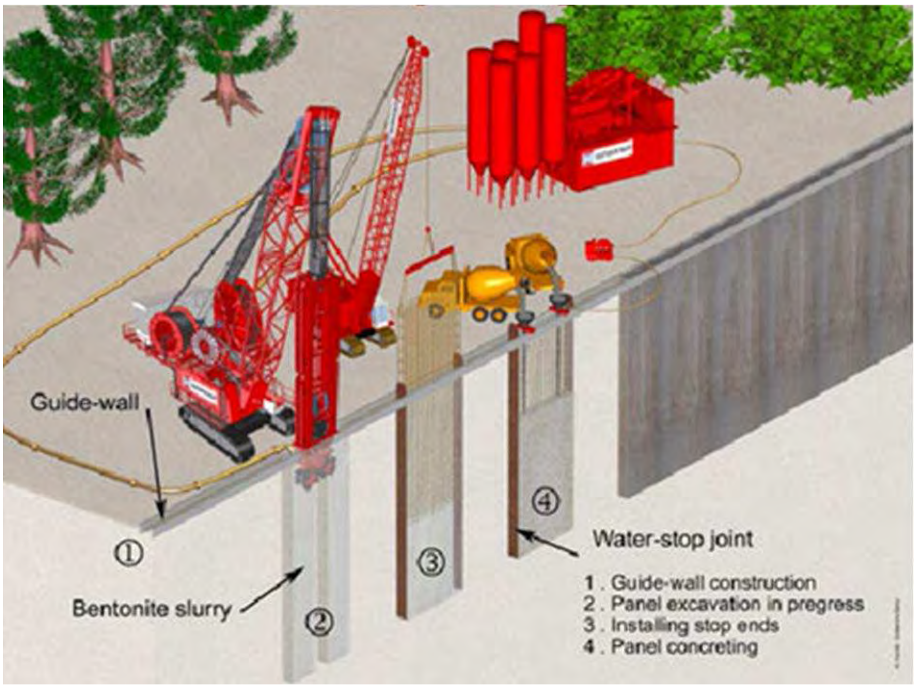

Figure 4.3 Construction of diaphragm wall. From Soletanche Bachy (2010).

The construction is done by first excavating a trench, which is kept filled with slurry in order to stabilize the sides. The excavation may be performed by a traditional auger in soft soils or with a hydrofraise in harder ground conditions. When the excavation is finalised, one or more reinforcement cages are typically inserted in the sleeve and the slurry is replaced by a tremie concrete cast bottom-up.

Adjacent panels are overlapped to create a continuous retaining structure. Even though methods exist to place waterstops in the joints between panels, it is not considered possible to cast a diaphragm wall cut-off structure that is completely watertight.

The verticality of diaphragm walls can be controlled during excavation and a resulting out-of plane verticality of 1:300-1:400 is obtainable. This means that this method may be used for shafts more than 100 m deep.

Sprayed concrete lining

In connection with the repository, Sprayed Concrete Linings (SCL) are only considered for limestone and rock excavations. Under certain conditions, this construction method may also be applicable in clay; however, this will usually require additional measures (e.g. freezing), which have a significant impact on the construction cost.

The limestone and rock condition at locations found suitable for the repositories are deemed to be sound average or above conditions. This means that the excavations to a large degree will be stable with minimum support.

The SCL method can be applied both for shaft construction and for construction of underground caverns. For the very best rock conditions no support may be required at all, but typically the exposed surface will be covered by a sprayed concrete layer of 50-250 mm dependent on the rock quality. The lower end of this range is assumed relevant for rock shafts and caverns and the upper end is relevant for the limestone shafts and caverns.

Rock anchors are applied in order to mobilise the zone around the cavern/shaft, so that the adequate load bearing capacity is achieved by a joint action of the sprayed concrete lining and the surrounding rock.

The sprayed concrete lining is not designed to carry any water pressures. This means that the rock around the caverns/shafts needs to be drained. This is achieved by drilled-in drainage pipes/bolts and application of drainage provisions (mats or similar) on the outer face of the sprayed concrete lining. The water drained off by these provisions is collected inside the excavation and discharged or reinfiltrated into the aquifer during construction. Drainage of the SCL areas can only be ceased after construction of the (watertight) permanent inner lining.

For rock conditions it may be sufficient to grout any water bearing fissures and strengthening of local weak zones with shotcrete and rock bolts.

The excavation in limestone is done either by hydraulic tools or by a road header. The latter method is more precise, but has some particular health and safety issues related to protection of employees against alpha-quartz dust caused by the milling. Ventilation and use of equipment with pressurized cabins or use of robotic equipment are available mitigating measures.

The rock excavation is done by the drill and blast method, where explosives are placed in a pattern of holes, typically 4-5 m long/deep. The explosives are detonated with different delays in order to control the obtained shape of the cavern/shaft.

4.2.4 Operation and monitoring of the repository

This section describes the relevant aspects of operating and methods for monitoring the repository. The aspects and methods described in here are valid for all types of repositories. Special considerations on the individual repository types are dealt with in the corresponding sections 4.4 to 4.10.

Operation of repository

It is estimated that the initial filling period, where the bulk waste amount shall be placed in the repository, lasts for approximately one year[6]. The activities within this period include final packing of the waste at the Risø area, transport to the repository facility, and the placement of the individual waste items at the repository, including possible backfill in between the containers.

[6] This does not include the time necessary for conditioning of the waste before placement in the repository

Hereafter, it is assumed that the active operation continues for 30 years with a waste amount of 8 m^3 per year. After the active operation period, the repository will be closed. A monitoring period of at least 30 years will follow before the repository site is left unmonitored.

Special aspects of operation that apply to the individual repository types are dealt with in the corresponding sections 4.4 to 4.10.

Monitoring of repository

Monitoring shall be applied to quantify the impact of the repository on the environment. In principle, monitoring relates to two quantities: the flow of water into and out of the repository and the outflow of gas (if any).

Monitoring shall be performed during and after construction/operation of the repository and shall be based on a suitable monitoring and surveillance programme.

Water monitoring shall comprise measurements on surface soil, surface water, water inside the repository (if any), and groundwater (samples from boreholes) with identification of the primary radionuclides. The general approach towards borehole samples shall be that samples are taken from one monitoring well upstream (direction of groundwater flow) and from three monitoring wells downstream. The distances between the monitoring wells and the repository may vary from a few hundred metres to one kilometre depending on the hydrogeological conditions of the site.

Gas, which may be formed when the waste gets into contact with (ground)water and decomposes, shall be monitored with identification of the primary radionuclides on ventilation pipes, which are designed to meet the specific requirements of the associated repository type. For example, the ventilation pipes may consist of 300 Ø HDPE pipes provided with an above ground discharge, which is protected against intrusion.

4.3 General layout of final storage plant

No matter which type of repository facility eventually will be included in the final storage plant, the plant will include various general facilities, as illustrated in Figure 4.4.

The borehole may in principle be established at a different location than the repository for the bulk part of the waste. However, in Figure 4.4 it is assumed that all possible repository facilities are located at the same site.

The final storage plant is expected to include the following facilities, as shown in Figure 4.4:

1. Guard house (approx. 50 m^2)
2. Office and staff facilities (approx. 300 m^2)
3. Packing and preconditioning facility (approx. 200 m^2)
4. Interim storage building (approx. 500 m^2)
5. Garage / maintenance workshop (approx. 200 m^2)
6. Main repository (max. 4700 m^2)
7. Borehole repository (if decided upon)

The total area needed for the finals storage plant is expected to be in the order of 2-3 ha.

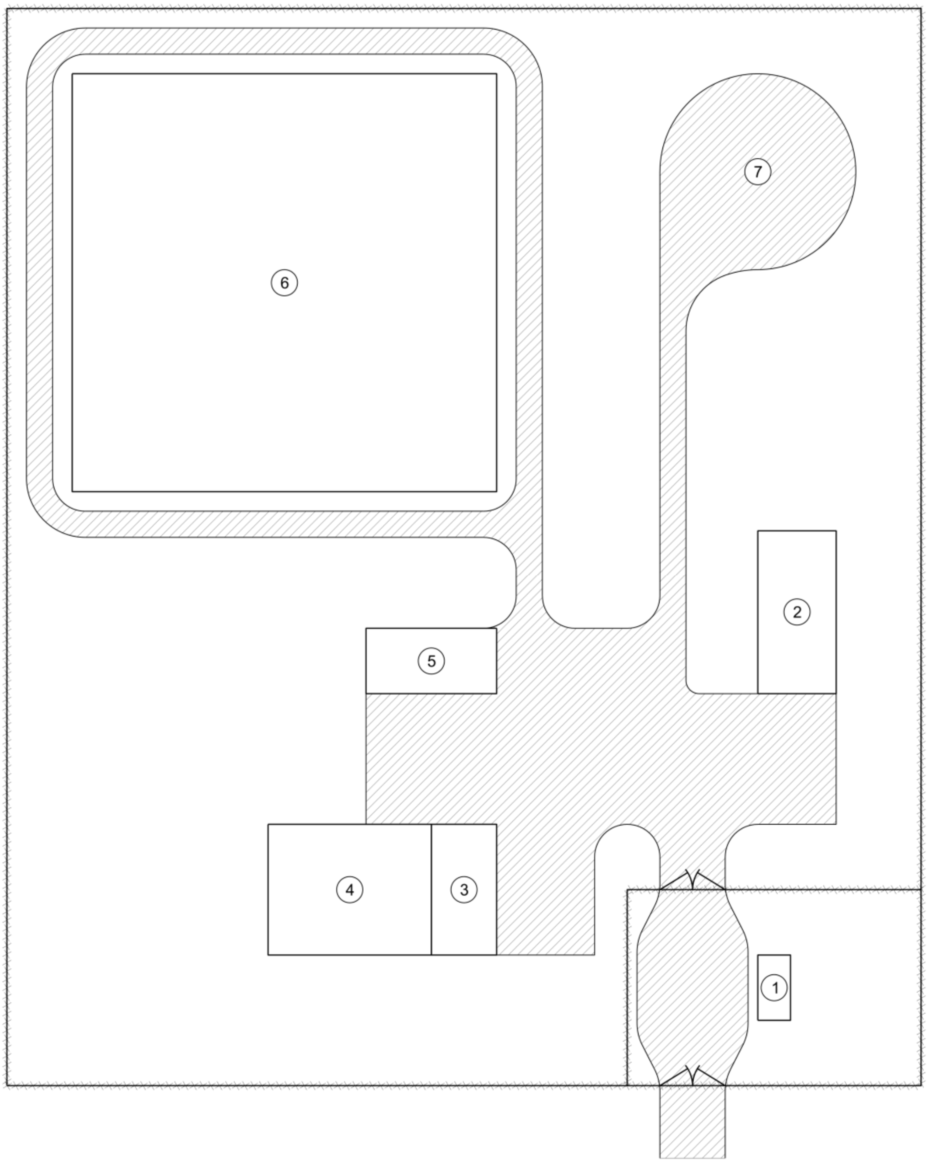

Figure 4.4 General layout of final storage plant. See also Drawing no. 1-01 in Appendix C.

In sections 4.4 to 4.10, the conceptual design of different types of repositories is described. This encompasses:

- Above surface repository
- Near surface repository
- Medium deep repository, shaft operated from ground level, irreversible
- Medium deep repository, shaft operated from ground level, reversible
- Medium deep repository, shaft operated inside repository
- Medium deep repository, cavern operated inside repository
- Borehole repository

4.4 Above surface repository

This repository type is in the safety assessment called ASR or Repository type 1.

4.4.1 Design and construction

The surface repository is designed as a multi-barrier system to be established above groundwater level, as shown in the cross section in Figure 4.5, Similar types of repositories exist at e.g. OKG in Sweden (Swedish Ministry of the Environment, 2007).

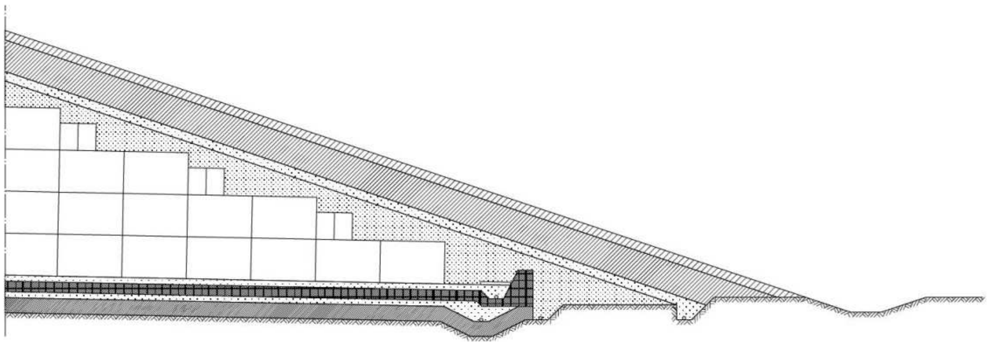

Figure 4.5 Cross section for above surface repository. See also Drawing no. 1-01 in Appendix C where the different materials are described.

The lower hydraulic barriers consist of:

- a 0.5 m thick clay liner;
- a 2.0 mm thick HDPE-liner; and
- a 0.35 m thick reinforced concrete slab.

The upper hydraulic barriers consist of:

- a 2.0 mm thick HDPE-liner; and
- a 1.0 m thick soil layer (e.g. clayey material).

The barriers are described in further details in the below.

Clay liner

The clay liner should be established as a mineral layer with a final thickness of minimum 0.5 m and a final permeability coefficient of maximum 10^{-10} m/s. The clay liner may consist of an existing clay layer (i.e. a natural geological barrier) provided the quality is sufficient, or of clay that has been taken from elsewhere and then built in at the actual site (i.e. an artificial geological barrier).

Before the clay liner is established, it must be ensured that the layers below posses a sufficient stability in order to avoid that possible settlement causes damages to the barrier or to the structures to be established above. Possible lower, unstable layers shall be excavated and replaced by suitable friction material.

It is recommended that the clay liner should fulfil the requirements for an artificial geological barrier, according to DS/INF 466 (1999). Clay for construction of artificial geological barriers for sanitary landfills must fulfil the following requirements:

- the clay content, L, shall be at min. 14% in the final clay liner (L is the amount of material with grain size less than 0.002 mm, according to weight and measures in %);
- plasticity index, I_p, higher than 5 %;
- limestone content less than 30%;
- water content during construction should be between W_{opt} and $W_{opt} + 3\%$. (W_{opt} is the optimal water content as determined by Standard Proctor tests); and
- no stone or gravel particles larger than 100 mm in diameter.

The clay liner is constructed in minimum two layers, each of them to be homogenised and compacted to a tightness of minimum 95 % Standard Proctor.

The clay liner shall be established with a uniform slope of 20 ‰ towards one side of the repository.

Lower HDPE liner

A 2.0 mm thick HDPE-liner (High Density Poly Ethylene) shall be established directly on top of the levelled and compacted surface of the clay liner. The HDPE-liner must as a minimum fulfil the requirements of DS/INF 466 (1999).

The HDPE liner shall be established with a uniform slope of 20 ‰ towards one side of the repository.

Reinforced concrete slab

An approximate 0.35 m thick reinforced concrete slab shall be established on top of a drainage layer, located above the clay liner. The concrete slab shall be designed to carry heavy weight, e.g. full containers in min. 4 layers, and a top cover consisting of drainage sand and soil, as described below.

The concrete slab shall be established with a uniform slope of 20 ‰ towards one side of the repository.

Upper HDPE liner

Once the waste containers and drums have been placed at the repository, a layer of suitable backfill material shall be installed in between and on top of the containers and drums and be compacted and shaped to form a hill, with slopes to all sides (max. slope of 1:3 to avoid erosion in layers above the liner). Different types of backfill material may be used, depending on the desired functionality of the repository (i.e. reversible or irreversible), see Chapter 3 and 9.

An upper HDPE liner, similar to the one described above, shall be installed on top of the backfill material.

It is recommended to install an electronic leakage detection system below the HDPE liner. Such system may be able to identify and locate possible leakages through the liner, in case the liner should be damaged.

Soil cover layer

A 0.3 m layer of drainage sand shall be established on top of the upper HDPE liner in order to remove infiltrating rain water and to protect the liner.

An approx. 1 m thick soil layer shall be installed on top of the repository as a barrier against intrusion by animals etc. Furthermore infiltration of surface water will be limited if clayey soil is used for the top cover.

Finally, a 0.3 m layer of top soil (humus) shall be put on top of the repository. Grass may be sown or bushes may be planted to protect against erosion and to limit the visual impact of the repository.

Gas ventilation

The gas generation from decomposition of the waste (i.e. plastic) is expected to be very limited. Anyway, one or more gas ventilation outlets should be established in the top cover, in order to ensure a controlled release of gasses. Furthermore, this will make it possible to monitor gas.

4.4.2 Operation of the repository

Filling the repository with waste

Filling the repository with waste is quite uncomplicated, as the containers and drums can be placed by means of either a fork lift or from outside the repository area by means of a crane. Surface run off during the filling period will be collected by the drainage layer and taken to the downstream drainage pipe, from where it may be lead to an open buffer pond for possible monitoring before outlet to external receiving waters.

This type of repository is not suited to be kept open for a long operational period (e.g. several years or more). Once the repository has been filled, it should preferably be closed and covered by the top sealing. Otherwise, it may be difficult to control and monitor the lower barriers, due to rain water infiltration at open areas.

Operation after closure

There will be no other operation needed than water monitoring and possible gas monitoring. Since the level of the concrete slab will be higher than the surrounding areas, there will be no surface water entering the repository area.

4.4.3 Possibilities for extensions

As mentioned above, the repository should preferably be closed after filling it with waste, delivered from the existing stock. Extension of the closed repository is possible, but may only be justified in case of a large additional supply of waste, which is unlikely to occur. Alternatively, a separate additional repository of the same type may be constructed if required after a number of years.

4.4.4 Monitoring of the repository

Monitoring of possible infiltrating rain water

If no outer damage happens to the repository and if the repository is frequently monitored, it is quite unlikely that infiltrating rain water will reach the waste. The main part of the annual precipitation will evaporate or run off due to a combination of the planting, the dense top soil layer and the hilly shape of the repository. The small net-infiltration through the top cover (most likely less than 50 mm per year) will be caught by the drainage layer and lead to the surrounding drainage system, where it can be monitored prior to release to the environment. In the event of cracks or holes in the upper HDPE-liner, a very limited water flow may pass the liner and leak into the backfill material surrounding the waste containers. Provided that a leakage detection system has been installed, it will then be possible to register when and where the leak happens. The actual area of the HDPE-liner can then easily be exposed and repaired.

If the leakage for any reason is not detected or if the liner is not repaired, infiltrating water may get into contact with the waste containers, depending on the type of backfill material used on top of and in between the containers. If e.g. concrete or bentonite is used as backfill material, it is very unlikely that water will reach the containers. However, if this still happens, the main part of the water will pass in between the containers and be caught by the drainage system on top of the concrete slab, from where it will be lead to a surrounding monitoring and control system. If there would be cracks in the concrete slab, there are further barriers below, i.e. the lower HDPE-liner and the clay liner.

Water monitoring can be introduced at the following 5 levels:
- Perimeter surface run-off ditch
- Perimeter drain below top soil
- Perimeter drain on top of concrete slab
- Perimeter drain on top of the lower composite liner (HDPE + clay).
- Possible downstream monitoring wells (depending on local hydrogeology).

Gas monitoring

Gas may be monitored at gas ventilation shafts established in the top cover.

4.4.5 Opening / emptying of reversible repository

The repository will remain reversible, since at any time it will be possible to open the repository and remove all or part of the waste.

4.5 Near surface repository

This section describes structural solutions for near surface repositories down to a depth of 10 m below ground level. Similar types of repositories exist in e.g. Japan, ref. IAEA (2005). This repository type is in the safety assessment called NSR or Repository type 2.

The Tender Documents categorize repositories at a depth of between 0 m and 30 m as ground-level repositories. However, the optimal layout for underground structures below a depth of 10 m differs from the optimal layout for near surface repositories.

4.5.1 Layout / design

A surface near repository can be constructed underground, but above the groundwater and with unrestricted drainage. This benefit can be promoted by placing the repository into an (artificial or natural) elevation above the surrounding surface level.

Down to a depth of around 10 m, box-shaped structures can be regarded as a reasonable solution. The optimal structural layout for repository structures at deeper depths is based on a cylindrical outer wall. This allows the soil and water pressure to be absorbed as hoop stresses, i.e. as circumferential compressive stresses. Considering that concrete is strong in compression, but weak in tension, this is of particular importance for concrete structures.

As illustrated in the Tender Documents, the near-surface repository may be operated from ground level or inside, see Figure 4.6 and Figure 4.7, respectively.

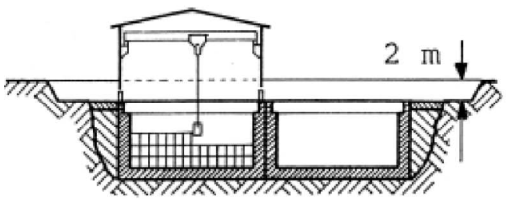

Figure 4.6 Cross section of possible structural solution for near surface repository, operated from ground level. From Danish Decommissioning (2009).

The near surface repository shall have a total volume of app. 8,200 m^3[7]. A layout with the internal measurements of 50 m × 28.5 m × 6 m is chosen, which fulfils the requirement with a total volume of 8,550 m^3.

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[7] See Table 3.2

Below a depth of 10 m, the soil pressure (and water pressure) increase to an extent, so that a cylindrical shell allowing the pressure to be absorbed as hoop stresses is a more suitable (and economical) structural solution. Appropriate solutions for depths below 10 m (and until a depth of 100 m) are considered as medium deep repositories. Different layouts of medium deep repositories are discussed in sections 4.6 to 4.9.

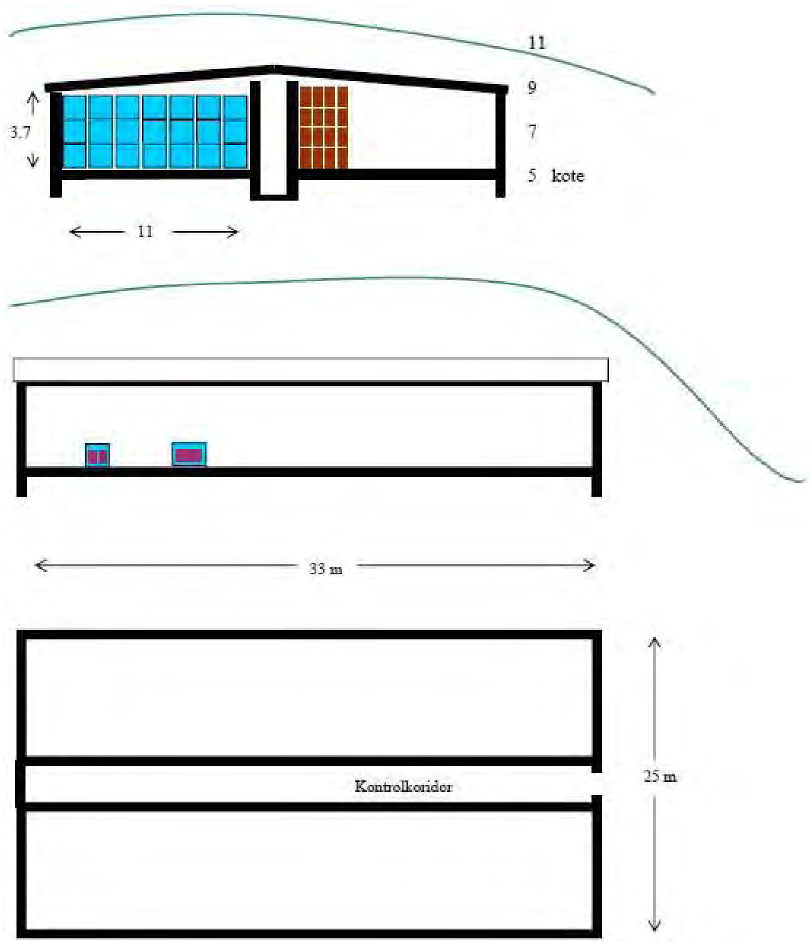


Figure 4.7 *Cross section and ground plan of possible structural solution for near surface repository, operated inside repository. From Danish Decommissioning (2009). The above dimensions are arbitrary; please see the text below for an example of actual (required) dimensions.*

4.5.2 Construction

The limited depth of the repository allows for use of all relevant cut-off structures (cf. section 4.2.3). Use of sheet or secant piles is considered preferable in clay till/clay or clay till/clay and limestone conditions, respectively.

The repository itself can be a relatively simple concrete structure. Even if it is located above the groundwater level, a membrane should be placed to prevent any leakage. The reinforced 0.5-1 m thick walls shall consist of dense good quality concrete and design shall make sure that cracking (also, due to differential settlements) is prevented. The bottom plate for both options operated from ground level and inside (Figure 4.6 and Figure 4.7, respectively) shall be cast in-situ, but other parts of the structure may consist of precast elements (possibly with in-situ cast top layer or as lost formwork). This approach generally leads to more economic solutions.

4.5.3 Operation of the repository

The maximal stacking load of the containers and drums must not be exceeded for the repository operated from the inside (cf. Figure 4.7) and it should not be exceeded for the repository operated from the ground level (Figure 4.6). In this case both solutions can be considered as reversible options. Based on available information on the existing containers used at Danish Decommissioning it is assumed that steel containers and ISO-containers can be stacked in a height of four and drums are expected to be stacked in a height of six.

It is possible to lower down all waste for final disposal and unhook it without the use of staff in the repository operated from the ground level (Figure 4.6). The other repository solution (Figure 4.7) can be operated from the inside by means of a crane system or forklifts.

4.5.4 Closure of the repository

After the intensive period of filling with the waste currently stored at Risø, waste will only be filled in the repository once or a few times per year. For safety reasons, it is suggested that the access facilities will be locked up to prevent all access after the intensive period.

When the repository operated from ground level (Figure 4.6) is filled for good with waste and backfill, the steel structure shall be removed and a self-supporting concrete structure shall be cast on top of the repository. In connection with the membrane on the outer perimeter, a tight membrane shall be established. On top of this membrane, a layer of protective concrete shall be cast, e.g. 300 mm steel fibre reinforced, high strength concrete. On top of this plate, friction material (e.g. gravel) shall be installed up to ground level.

The repository operated from the inside shall be locked and sealed by a concrete structure. If necessary, it may be backfilled through e.g. an opening from the top that shall also be sealed by a concrete slab.

4.5.5 Possibilities for extensions

Extension of the closed repository is possible, but it would in principle include construction of a new near surface repository next to the existing one. Thus, alternatively, a separate additional repository of the same type may be constructed elsewhere (on site or at another site), if required after a number of years. So, extension may only be justified in case of a large additional amount of waste, which is unlikely to occur.

4.5.6 Monitoring of the repository

See section 4.2.4.

4.5.7 Opening / emptying of the repository

As long as the space between the waste containers/drums is not filled, it is relatively simple to empty any given cell by hoisting up all waste containers/drums and any concrete slab above a specific cell.

If it is decided to reopen the repository after is has been sealed for good, the condition of the sheet pile or secant walls shall be assessed at first. Then, the membrane and the protective concrete slabs may be dismantled. Special effort is required for the concurrent removal of the backfill from inside the repository. Handling of containers and drums depends on their condition.

4.6 Medium deep repository, shaft operated from ground level, irreversible

This section describes structural solutions for an irreversible medium deep repository based on a shaft operated from ground level. Repositories at a depth between 10 m and 100 m from ground level are considered as medium deep repositories. Such types of repositories exist at several places, e.g. at SFR in Forsmark, Sweden, ref. SKB (2006). This repository type is in the safety assessment either called MDR, GI or Repository type 3, 4 or 5 dependent on the diameter and maximum depth of the repository, see section 4.6.2.

Irreversible repositories are designed without consideration to whether containers and drums can absorb the load from the upper layers, i.e. the containers and drums may deform in an uncontrolled manner.

4.6.1 Layout / design

As mentioned in section 4.5, the optimal structural layout for a repository at depths below 10 m is based on a cylindrical outer wall.

Medium deep repositories can be established in a shaft, as described in section 4.2.3. Three general designs have been considered for the medium deep repository based on a shaft. At the bottom layer of the repository an inside diameter of 33.8 m, 26.0 m and 18.0 m was chosen and the wall thickness has been set to 1.50 m. The optimum cut-off solution depends on the given depth and geology, as described in section 4.2.3. For this design approach diaphragm walls with a maximum thickness of 1.52 m are assumed. The circular pressure in the diaphragm walls limits the depths of the three repository designs.

For medium deep repositories operated from ground level, the open shaft has to be covered by a hall structure fitted with a travelling crane, see Figure 4.8.

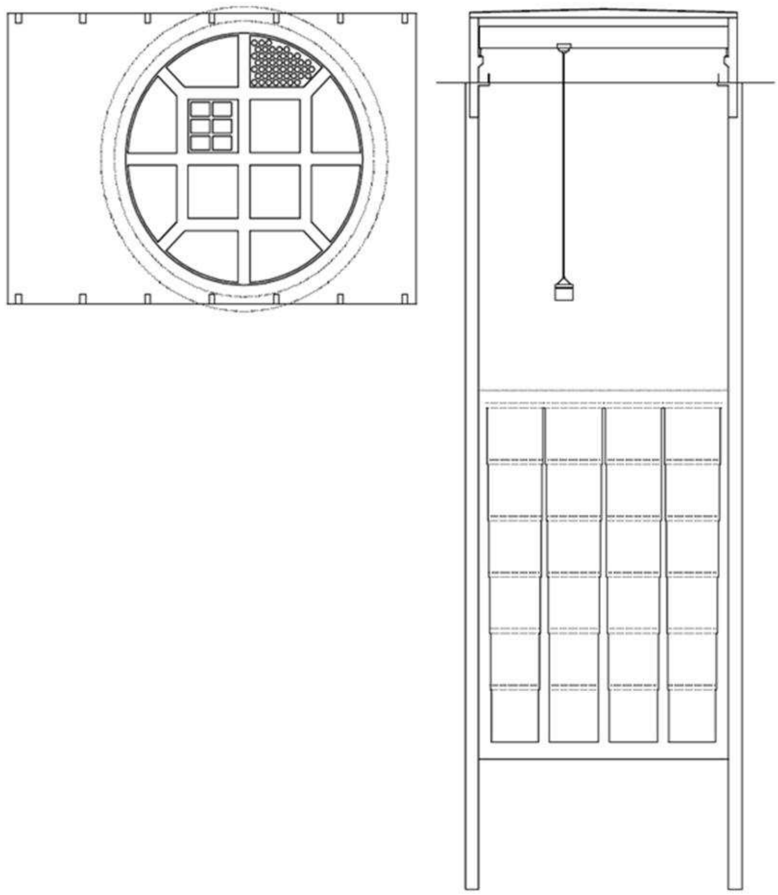


Figure 4.8 Ground plan and cross section of repository based on shaft operated from ground level, see also Drawing no. 1-10 in Appendix C.

All material to be deposited in the repository is transported into the hall and then lifted into the repository.

4.6.2 Construction

Figure 4.8 shows an operating hall constructed of ribbed roof sheets on precast Paroc panels. Nowadays, this is often the least expensive solution.

For all cut-off structures (cf. section 4.2.3), the shaft cannot be constructed with the same durability as a conventional concrete structure. Thus, all medium deep repositories based on a shaft shall be cast against the shaft walls with a drain and membrane in between, cf. drawings no. 1-06 to no. 1-11 in Appendix C.

The ground plans and cross sections of the proposed designs for the irreversible repository based on a shaft operated from ground level are shown in Figure 4.9 and Figure 4.10.

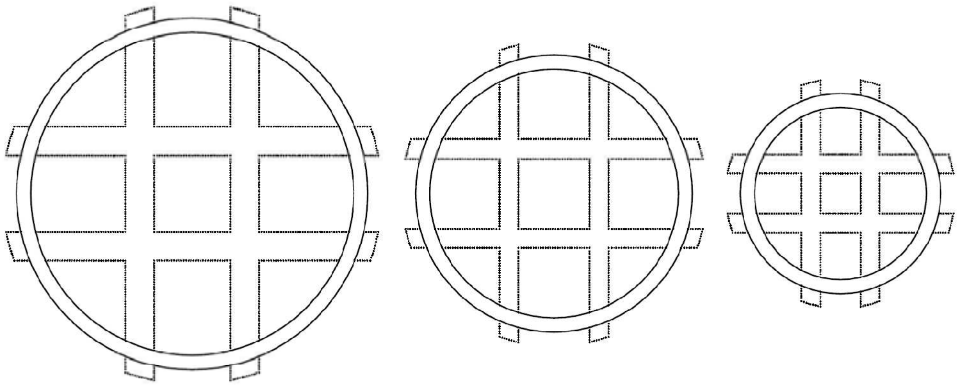


Figure 4.9 Ground plans of proposed designs for irreversible repository operated from ground level. Inside diameters are 33.8 m, 26.0 m and 18.0 m. See drawings no. 1-06 to no.1-11 in Appendix C for possible cut-off structures.

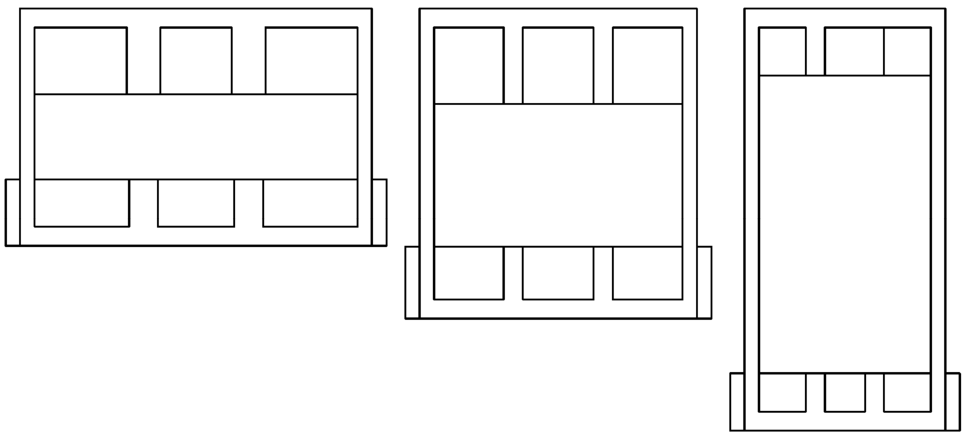


Figure 4.10 Cross sections of proposed designs for irreversible repository operated from ground level. Inside diameters are 33.8 m, 26.0 m and 18.0 m. See drawings no. 1-06 to no. 1-11 in Appendix C for possible cut-off structures.

For the designs shown in Figure 4.9 and Figure 4.10 it is necessary that the bottom and top slab are constructed especially sturdy.

The bottom slab structure is necessary to absorb the water pressure. The four intersecting beams are lead into the diaphragm walls (not included in Figure 4.9 and Figure 4.10), entailing that the water pressure is transferred to the walls.

Similarly, the top slab is necessary to transfer the weight of soil and groundwater and ground level surface load to the cylinder wall. Settlement of the waste and the fill should be taken into account, which means that the top slab must be designed to span freely across the entire repository to prevent structural failure and consequential leakage. Otherwise it is likely that the top slab would break within few years after construction. This is not considered acceptable.

The thickness of the bottom and top slab shown in Figure 4.10 corresponds to the deepest possible location.

The repository with a 33.8 m inside diameter is 8.9 m high. The lower side of the bottom slab cannot be placed deeper than 58 m below ground level. At that location, the bottom slab must be 7 m high and thus the floor is 51 m below ground level. The top slab must be 9 m high.

The repository with a 26.0 m inside diameter is 15 m high. The lower side of the bottom slab cannot be placed deeper than 79 m below ground level. At that location, the bottom slab must be 7.5 m high and thus the floor is 71.5 m below ground level. The top slab must be 10 m high.

The repository with an 18 m inside diameter is 31.2 m high. The lower side of the bottom slab cannot be placed deeper than 106 m below ground level. At that location, the bottom slab must be 6 m high and thus the floor is 100 m below ground level. The top slab must be 7 m high.

4.6.3 Operation of the repository

It is possible to lower down all waste for final disposal and unhook it without the use of staff in the underground repository. This can be done both if the repository is dry and if it is filled with water.

When a suitable quantity of waste material has been lowered down, backfilling material is added and the lowering is proceeded.

4.6.4 Closure of the repository

After the intensive period of filling with the waste currently stored at Risø, waste will only be filled into the repository once or a few times per year. For safety reasons, it is suggested that the shaft down to the repository shall be covered by a steel structure rendering it inaccessible after the intensive period of filling. The steel structure shall be fitted with heavy, locked covers, allowing subsequent filling to take place during the active period of the repository.

When the repository is filled for good with waste and backfill, the steel structure shall be removed and a self-supporting concrete structure shall be cast on top of the repository. In connection with the membrane on the outer perimeter, a tight membrane shall be established. On top of this membrane, a layer of protective concrete shall be cast, e.g. 300 mm steel fibre reinforced, high strength concrete. On top of this plate, friction material (e.g. gravel) shall be installed up to ground level.

4.6.5 Possibilities for extensions

Extension of this repository type is not possible and a separate additional repository of the same type may be constructed, if required after a number of years. This may only be justified in case of a large additional amount of waste, which is unlikely to occur.

4.6.6 Monitoring of the repository

See section 4.2.4.

4.6.7 Opening / emptying of the repository

Opening / emptying of the repository is not foreseen for the irreversible medium deep repository based on a shaft operated from ground level.

4.7 Medium deep repository, shaft operated from ground level, reversible

This section describes structural solutions for a reversible medium deep repository based on a shaft operated from ground level. Repositories at a depth between 10 m and 100 m from ground level are considered as medium deep repositories. This repository type is in the safety assessment either called MDR, GR or Repository type 6, 7 or 8, dependent on the depth of the repository, see section 4.7.2.

Reversible repositories are designed taking into account the maximal stacking load that can be absorbed by the containers and drums.

4.7.1 Layout / design

In accordance with the irreversible medium deep repository based on a shaft operated from ground level (cf. section 4.6), the proposed designs for the reversible medium deep repository based on a shaft operated from ground level are based on a cylindrical concrete structure with inside diameters of 33.8 m, 26.0 m and 18.0 m, and a wall thickness of 1.5 m at the lowest level. The structure is assumed to be cast against diaphragm shaft walls with a drain and membrane in between. Moreover, corresponding to the irreversible design solution, an operating hall with a travelling crane above the shaft is required.

As opposed to the irreversible design solution, the maximal stacking load that can be absorbed by the containers and drums has been taken into account for the proposed designs of reversible repositories. Based on the information included in the Tender Documents, it has been assessed that containers can be stacked in a height of four and drums are expected to be stacked in a height of five (entailing that all cells have the same height).

4.7.2 Construction

The repository is to be fitted with a cylindrical outer shell designed to absorb soil and water pressure as hoop stresses. Inside the cylinder, the room is divided into cells, both horizontally and vertically.

Horizontal division

The water pressure on the bottom is transferred by a 2 m thick bottom slab both directly to the cylinder wall and to the inside walls. These are used as high beams to transfer the water pressure on the bottom slab to the cylinder wall. From the cylinder wall, the forces are transferred to the shaft wall.

Similarly, a 2 m thick top slab transfers the weight of soil and groundwater and ground level surface loads both to the cylinder wall and to the inside walls. These transfer the forces directly to the bottom slab, which in this case functions as a standard foundation.

The horizontal division is illustrated in Figure 4.11.

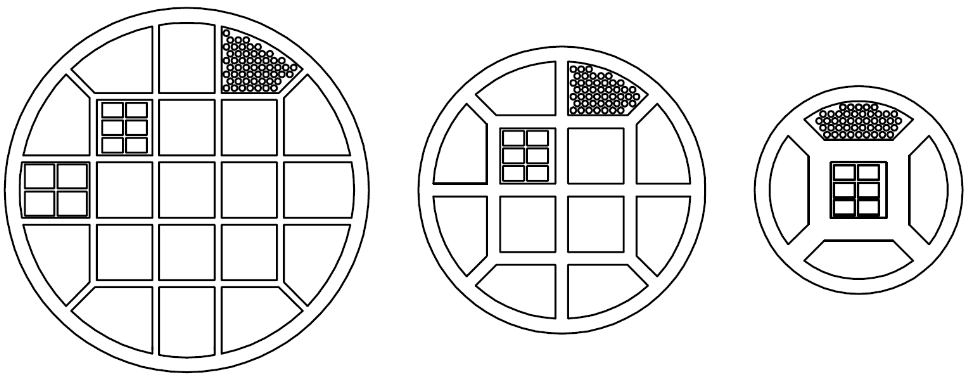


Figure 4.11 Ground plans of proposed designs for reversible repository operated from ground level. Inside diameters are 33.8 m, 26.0 m and 18.0 m. See drawings no. 1-06 to no.1-11 in Appendix C for possible cut-off structures.

The cell sizes shown in Figure 4.11 have been chosen so that they can be covered by a single concrete element slab, which does not exceed the maximal weight of a travelling crane with normal lifting capacity.

The cell divisions and the limited stacking height entail that the repository has obsolete space: The theoretical volume of the waste (considering a distance of 300 mm between the items) is around 4,940 m^3. The repository design with 21 cells in three layers (a total of 63 cells) contains a total of around 14,400 m^3; the one with 12 cells in six layers (a total of 72 cells) contains a total of 16,550 m^3 and the one with five cells in eleven layers (a total of 55 cells) contains a total of 14,990 m^3.

Vertical division

A height of 6 m has been chosen for each cell, allowing containers and drums to be stacked to the maximum permitted height. The walls have been designed to reduce the thickness by 100 mm on either side for every level.

On top of each cell a 500 - 600 mm thick concrete slab in the shape of a concrete element shall be lowered down from ground level. The slab shall be mounted on the 100 mm wide shelves with butt joints without further joint casting.

The vertical division is illustrated in Figure 4.12.

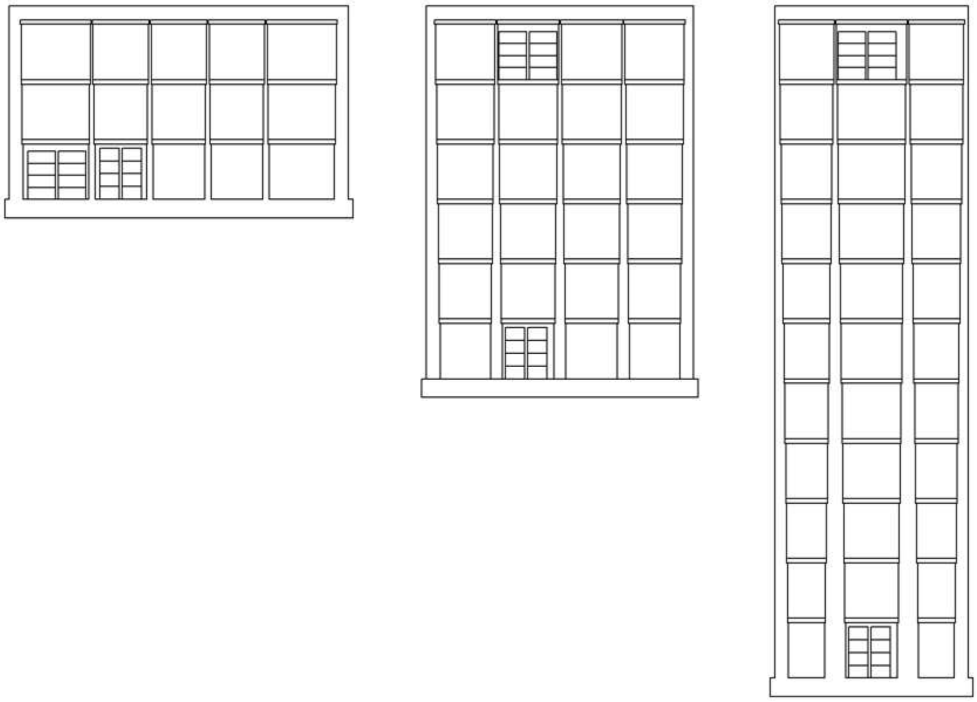


Figure 4.12 Cross sections of proposed designs for reversible repository operated from ground level. Inside diameters are 33.8 m, 26.0 m and 18.0 m. See drawings no. 1-06 to no.1-11 in Appendix C for possible cut-off structures.

The repository with a 33.8 m inside diameter is 23 m high, measured from the lower side of the bottom slab to the top side of the top slab. The lower side of the bottom slab cannot be placed deeper than 58 m below ground level and the floor is 56 m below ground level. Thus, the top slab is located no more than 35 m below ground level.

The repository with a 26 m inside diameter is 42.5 m high, measured from the lower side of the bottom slab to the top side of the top slab. The lower side of the bottom slab cannot be placed deeper than 79 m below ground level and the floor is 77 m below ground level. Thus, the top slab is located no more than 36.5 m below ground level.

The repository with an 18 m inside diameter is 75.5 m high, measured from the lower side of the bottom slab to the top side of the top slab. The lower side of the bottom slab cannot be placed deeper than 102 m below ground level and the floor is 100 m below ground level. Thus, the top slab is located no more than 26.5 m below ground level.

Under the bottom slab, on the outside perimeter and on the top side of the top slab a membrane shall be mounted and the top slab shall be fitted with protective concrete.

4.7.3 Operation of the repository

It is possible to lower down all waste for final disposal and unhook it without the use of staff in the underground repository. This can be done both if the repository is dry and if it is filled with water.

When a repository cell is filled, a concrete slab (cf. Figure 4.12) is lowered down from ground level. Afterwards, the next cell can be filled with waste.

Piping shall be embedded in the walls, which allows backfilling of the cells with e.g. bentonite.

4.7.4 Closure of the repository

The procedure for closure of the reversible medium deep repository based on a shaft operated from ground level corresponds to the one for the irreversible medium deep repository based on a shaft operated from ground level (cf. section 4.6.4).

4.7.5 Possibilities for extensions

Extension of this repository type is not possible and a separate additional repository of the same type may be constructed, if required after a number of years. This may only be justified in case of a large additional amount of waste, which is unlikely to occur.

4.7.6 Monitoring of the repository

See section 4.2.4.

4.7.7 Opening / emptying of the repository

As long as the space between the waste containers/drums is not filled, it is relatively simple to empty any given cell by hoisting up all waste containers/drums and any concrete slab above a specific cell.

If it is decided to reopen the repository after is has been sealed for good, the condition of the diaphragm walls shall be assessed at first. If their condition is sound, the friction material may be removed exposing the top slab of the repository. If the diaphragm walls are not considered to be strong enough come time for the opening of the repository, new diaphragm walls or a similar cut-off structures shall be established before removal of the friction material.

Then, the membrane and the top slab may be dismantled and the slabs on top of the cells may be opened and the cells emptied one after each other. Special effort is required for the concurrent removal of the backfill from each cell.

Handling of containers and drums depends on their condition. If they are intact, they may again be handled from ground level. If the lifting eyes of the containers are heavily corroded, personnel with safety gear may be required inside the repository to hook up the items.

4.8 Medium deep repository, shaft operated inside repository

This section describes structural solutions for medium deep repositories based on a shaft that is operated inside the repository. Repositories at a depth between 10 m and 100 m from ground level are considered as medium deep repositories. In the safety assessment this repository type is called MDR, IR or Repository type 9.

4.8.1 Layout / design

In accordance with the irreversible and reversible medium deep repositories based on a shaft operated from ground level (cf. sections 4.6 and 4.7), the proposed designs for the medium deep repository based on a shaft operated inside the repository are based on a cylindrical concrete structure with inside diameters of 33.8 m, 26.0 m and 18.0 m, and a wall thickness of 1.5 m at the lowest level.

In accordance with the reversible medium deep repositories operated from ground level, the maximal stacking load that can be absorbed by the containers and drums has been taken into account for the proposed designs of medium deep repositories operated inside the shaft (cf. section 4.7). Thus, this design is a reversible option.

As opposed to the medium deep repositories operated from ground level (cf. section 4.6 and 4.7), medium deep repositories operated inside require the presence of heavy transport equipment inside the repository.

4.8.2 Construction

Corresponding to medium deep repositories operated from ground level (cf. sections 4.6 and 4.7), medium deep repositories operated from the inside are established in a shaft as cylindrical concrete structure cast against a suitable cut-off structure for the given depth and geology, cf. section 4.2.3. A diaphragm wall (with a drain and membrane between) is also assumed in this design approach.

The waste for final disposal can only be handled expediently by means of forklifts or cranes.

Forklift solution

The weight of the containers entails that the capacity of electrically powered forklift is insufficient, for which reason diesel forklifts would have to be used. This would require major ventilation systems.

Figure 4.13 illustrates the dimensions of an examined forklifts make.

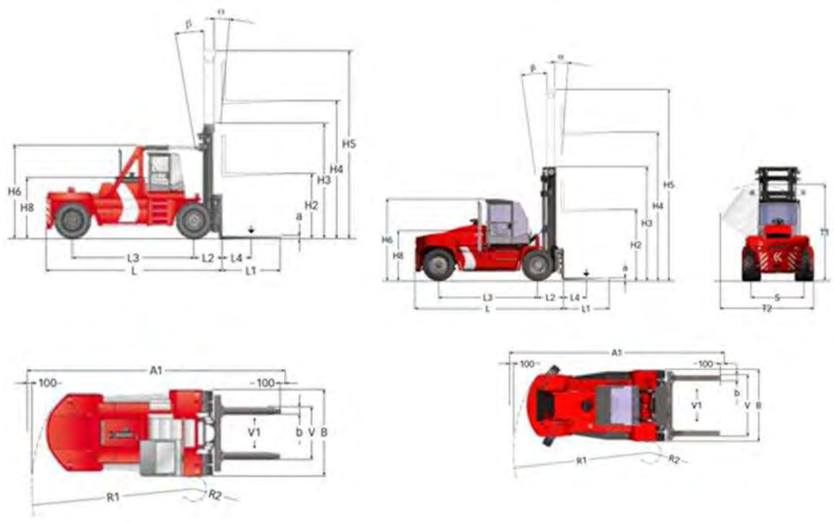


Figure 4.13 Dimensions of forklift models.

The forklift on the left side of Figure 4.13 is the smallest model (of the examined make) that can handle a 21 tonne ISO container. The total length and width (A1 and B on figure) are 9.2 m and 3.1 m. The dead weight is 31.2 tonne and the maximum axle load is 49.5 tonne. The length without forks (considering lift size or shaft for lowering, L in Figure 4.13) is 6.1 m. This forklift requires a clear width of 10 m for manoeuvring.

The forklift on the right side of Figure 4.13 is the smallest model of the same make that can handle a 13 t container. The total length and width are 8.3 m and 2.5 m. The dead weight is 21.4 t and the maximal axle load is 33.8 t. The length without forks is 5.4 m. This forklift requires a clear width of 7 m for manoeuvring.

Getting a forklift for transporting containers into the repository would require a very large freight lift. Considering the measurements and loads indicated above, a forklift solution is assessed to be unrealistic.

Crane solution

Considering the indicated container sizes and loads, only travelling cranes or two-rail cranes are realistic options. Both can be used as indicated in the cross sections and ground plans in Figure 4.14 and Figure 4.15.

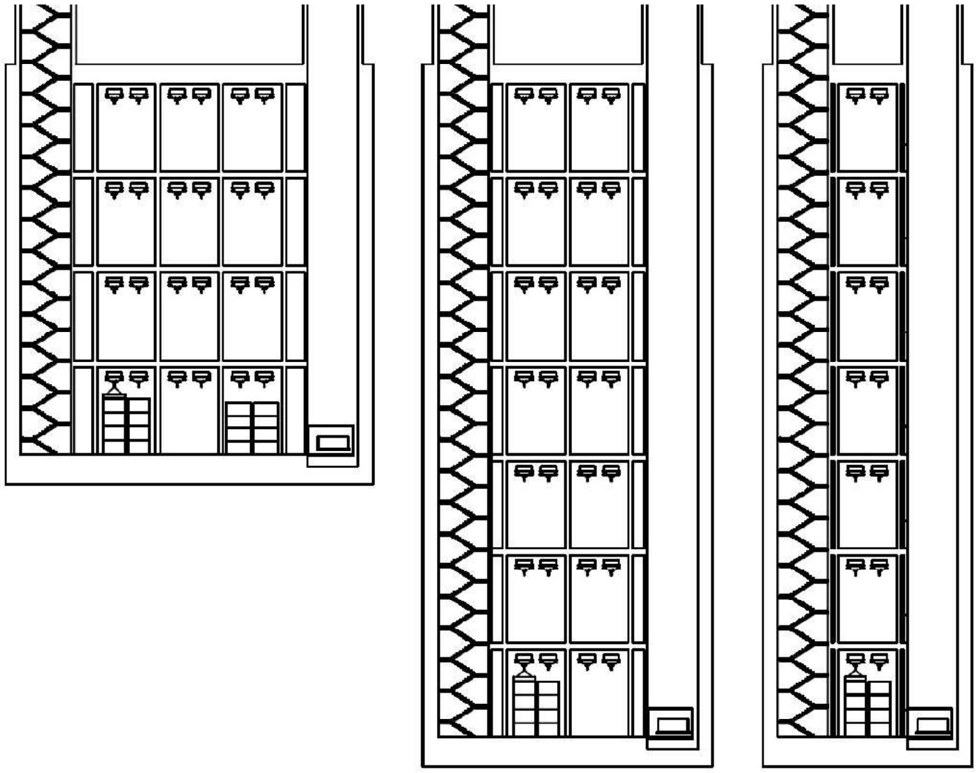


Figure 4.14 Cross sections of proposed designs for repository based on a shaft that is operated inside the repository. Inside diameters are 33.8 m, 26.0 m and 18.0 m. See drawings no. 1-06 to no. 1-11 in Appendix C for possible cut-off structures.

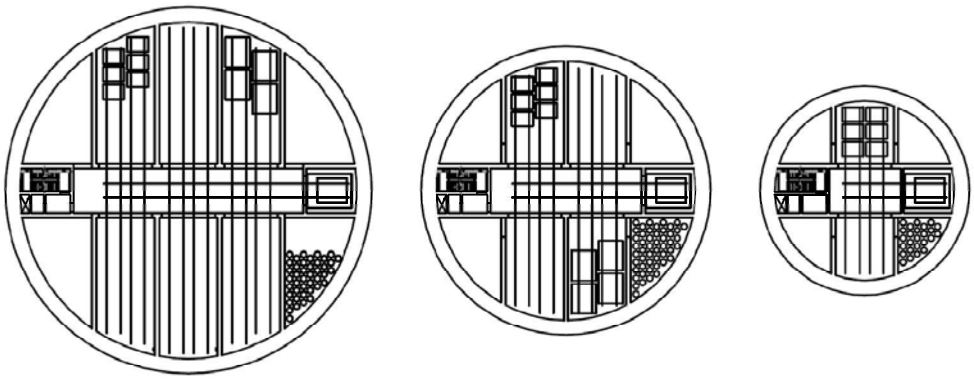


Figure 4.15 Ground plans of proposed designs for repository based on a shaft that is operated inside the repository. Inside diameters are 33.8 m, 26.0 m and 18.0 m. See drawings no. 1-06 to no. 1-11 in Appendix C for possible cut-off structures.

The repository with an inside diameter of 33.8 m is 42.1 m high measured from the lower side of the bottom plate to the top side of the top plate. The lower side of the bottom slab cannot be placed deeper than 58 m and the floor is 55 m below ground level. Thus, the top slab is placed maximum 15.9 m below ground level.

A medium deep repository operated from the inside may only be a relevant option, if a reversible repository is chosen. The individual rooms are so high that that only the repository with the maximum diameter of 33.8 m can be established in one shaft. The other two solutions required two or three shafts. Moreover, all repository solutions operated from the inside require stair shaft(s) and lift shaft(s) protruding above ground level. Besides that, it would be necessary to establish an escape staircase, various installations (ventilation, sanitation, power and emergency lighting, etc.) and staff facilities.

For the above reasons, it is recommended to exclude this repository type from further considerations.

4.8.3 Operation of the repository

When a room has been filled with waste, steel plates shall be mounted by means of bolts, and the room shall be filled with backfilling material.

4.8.4 Closure of the repository

The procedure for closure of the reversible medium deep repository based on a shaft operated from ground level corresponds to the ones for the medium deep repository based on a shaft operated from ground level (cf. section 4.6.4). It shall be noted that for safety reasons the entire shaft (including stair shafts and lift shafts) shall be filled with backfill.

4.8.5 Possibilities for extensions

Extension of this repository type is not possible and a separate additional repository of the same type may be constructed, if required after a number of years. This may only be justified in case of a large additional amount of waste, which is unlikely to occur.

4.8.6 Monitoring of the repository

See section 4.2.4.

4.8.7 Opening / emptying of the repository

As long as the space between the waste containers/drums is not filled, it is relatively simple to empty any given room. The advantage of the medium deep repository operated inside the shaft compared to the ones operated from ground level is that any room can be accessed an emptied without removing the waste on top of it.

If it is decided to reopen the repository after is has been sealed for good, the condition of the diaphragm walls shall be assessed at first. If their condition is sound, the friction material may be removed exposing the top slab of the repository.

If the diaphragm walls are not considered to be strong enough come time for the opening of the repository, new diaphragm walls or a similar cut-off structures shall be established before removal of the friction material.

Then, the membrane and the top slab may be dismantled and the slabs on top of the cells may be opened and the cells emptied one after each other. Special effort is required for the concurrent removal of the backfill from each cell.

Handling of containers and drums depends on their condition. If they are intact, they may again be handled from ground level. If the lifting eyes of the containers are heavily corroded, personnel with safety gear may be required inside the repository to hook up the items.

4.9 Medium deep repository, cavern operated inside repository

This section describes structural solutions for medium deep repositories based on a cavern that is operated inside the repository. Repositories at a depth between 10 m and 100 m from ground level are considered as medium deep repositories. In the safety assessments this repository type is called MDR, CA or Repository type 10.

4.9.1 Layout / design

The proposed design for the medium deep repository based on a cavern operated inside the repository is based on a cylindrical shaft concrete structure with inside diameters of around 12 m down to the level of the horizontal caverns.

The horizontal caverns can be configured in multiple ways. For instance, different cavern fingers may be used for deposit of different waste fractions.

The capacity of the caverns is controlled by the cross section area and the length. In this proposal the cross section has been kept to a minimum, with maximum internal diameters of around 8 m in order to create a structure as robust as possible. The length of each finger can be varied at will.

Generally, the cavern solution is only considered feasible in rock and limestone. In limestone, the cover above the cavern may have to be increased depending on the quality of the upper limestone layers at the given location.

Caverns may also be built in clay. However, it must be expected that a substantial soil improvement has to take place in connection with such works. Certain plastic clay formations do not necessarily require soil improvement during cavern excavation and use of industrial tunnel techniques may be an option, ref. Bastiaens & Bernier (2005). However, it will always be necessary to install a temporary and/or permanent lining solution during excavation to construct caverns in the considered diameters, e.g. around 8 m.

4.9.2 Construction

The shaft down to the level of the repository caverns shall have a size just adequate for transports (waste, equipment and personnel) and therefore the shaft for the cavern type deposit is much smaller than for the shaft deposits described in sections 4.6, 4.7 and 4.8. The repository is built as short caverns extending from the transportation shaft.

The caverns are shaped so that the concrete linings stand in hoop stresses. This will allow a construction using e.g. steel fibre reinforced concrete, which is less vulnerable to corrosion, since corrosion of the fibres does not lead to degradation of the concrete. In the final stage the load bearing capacity of the inner lining will not depend on the steel fibres, since the structure will stand in compression.

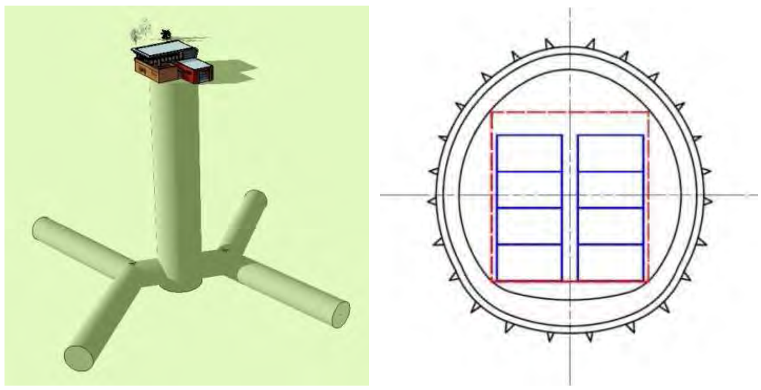


Figure 4.16 Example layout of intermediate depth cavern repository. See also drawing no. 1-12 in Appendix C.

In principle, the caverns could be done by tunnel boring machines. However, due to the limited volume of the repositories, such a construction method would not be economically attractive.

Thus, the drill and blast method described in section 4.2.3 is the recommended method for excavation of rock and lime stone.

Plastic clay formations that do not require soil improvement could be excavated by traditional mining methods and, for instance, be permanently lined with steel fibre reinforced concrete segments. The safest soil improvement method for excavation in other clay formations (also requiring the least drilled penetrations) would probably be the freezing method (presuming that the clay is saturated and thus freezable). Freezing may take place through freezing pipes drilled horizontally from the shaft. By use of directional drillings it is possible to create a freezing zone of more than 100 m length at a time.

4.9.3 Operation of the repository

Stacking of containers and drums inside the cavern correspond to the stacking in each cell in the reversible shaft solutions (cf. sections 4.7 and 4.8).

Like the medium deep shaft repositories operated from inside the shaft, the medium deep cavern repository requires the presence of heavy transport equipment inside the repository. However, since the layout of the caverns can be made to better suit the manoeuvrability of forklifts, it is for the present assumed that moving of the drums and containers at the repository level takes place by use of forklifts lifting the waste to its final position.

Space for turning of the forklifts can be arranged in the Y-shaped fork sections, ref. Figure 4.16 above.

The ventilation required during fill up of the repository can be arranged at the top of the caverns (above the red square in Figure 4.16). In the shaft there will be room for vertical ventilation ducts.

4.9.4 Closure of the repository

One cavern finger at a time may be sealed off, when it is filled up. The sealing off is suggested to be done by a thick internal concrete plug at the shaft end of the cavern finger with a shape ensuring that the structural stability of the final deposit does not rely on traditional reinforcement as all parts of the lining stand in compression.

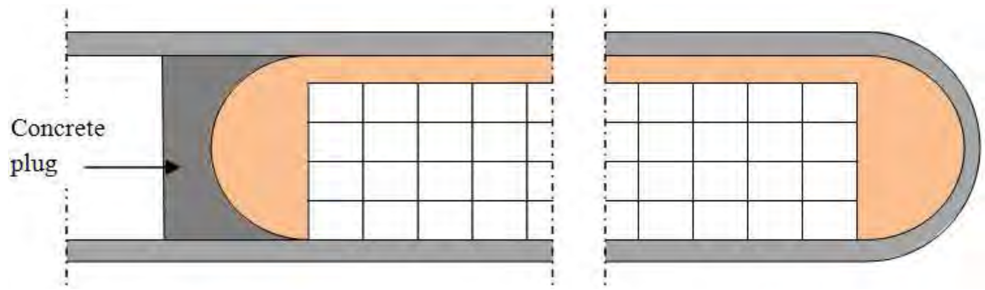


Figure 4.17 Example for final closing of cavern repository. See also drawing no. 1-12 in Appendix C.

The cavern interior may be filled up with suitable backfill around the drums and containers, e.g. pumped in through pipes penetrating the concrete plug.

4.9.5 Possibilities for extensions

Extension of this repository type is principally possible, if the cavern system was prepared for future additional cavern fingers. Also the length of each finger that has not been filled up and sealed can be extended. Extension of fingers that have already been sealed is not recommended. So, it is considered impossible to extent a completely filled and closed cavern repository and preferable to construct a separate additional repository of the same type, if required after a number of years. This may only be justified in case of a large additional amount of waste, which is unlikely to occur.

4.9.6 Monitoring of the repository

See section 4.2.4.

4.9.7 Opening / emptying of the repository

Before the proposed final closure of the caverns, it is possible to retrieve the deposited containers and drums, though access to the innermost containers and drums require removal of all later placed waste in the cavern finger in question. Hence, the cavern repository is considered to constitute a reversible solution.

It is possible to leave the shaft open and, hence, inspectable and maintainable even after the caverns are sealed off. Ultimately, a new inner lining may be cast inside the shaft, thereby increasing the shaft structures service life time considerably.

In such a scenario the internal concrete plug may be broken down at a later stage providing access to the cavern repositories.

These possibilities may be exploited for 2-300 years.

At some stage the shaft should probably be backfilled and opening/emptying of the cavern repositories will be similar to the procedures described for the reversible shaft repositories, ref. sections 4.7 and 4.8.

4.10 Borehole repository

Borehole repositories or borehole disposals have not been used in Denmark so far for any type of disposal. For the purpose of this report a borehole repository is defined as a geological disposal. The term “geological disposal” refers to the disposal of waste in a geological stable formation below ground and “disposal” means that it would normally not be the intention to retrieve the waste once disposed, ref. IAEA (2009).

For the purpose of this pre-feasibility study, borehole disposals at intermediate depth shall be considered. Intermediate depth means deeper than 30 m below ground surface, i.e. at a depth where human intrusion is limited, e.g. to drilling, tunnelling, quarrying, etc. ref. IAEA (2009). In the safety assessments this repository type is called BORE or Repository type 11.

4.10.1 Layout / design

A borehole repository can either be located in a separate location or at the same location as other types of repositories. The main difference being that a separate location will be more costly (site acquisition, roads, welfare facilities, conditioning building, etc.). This type of repository exists in e.g. South Africa, ref. NECSA (2004). A simplified site layout is shown in Figure 4.18.

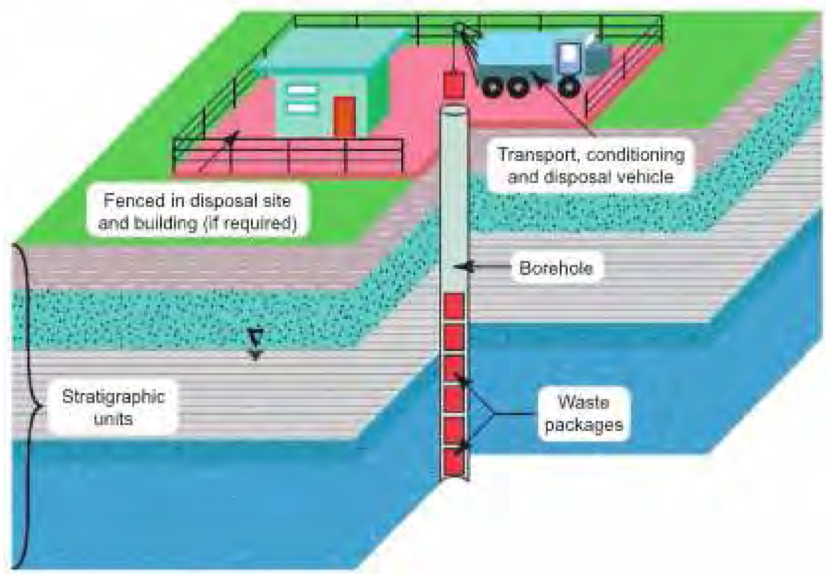


Figure 4.18 Schematic layout of a borehole disposal facility. From IAEA (2009).

A feasible design of the borehole itself is given in IAEA (2009). In the design it is assumed that the waste is packed in canisters each with a height of 1 m and an external diameter of 0.5 m. Each canister can hold approx. 55 l of waste. It is suggested that the canisters are placed three at time packed in a cage for placing purposes as shown in the below Figure 4.19.

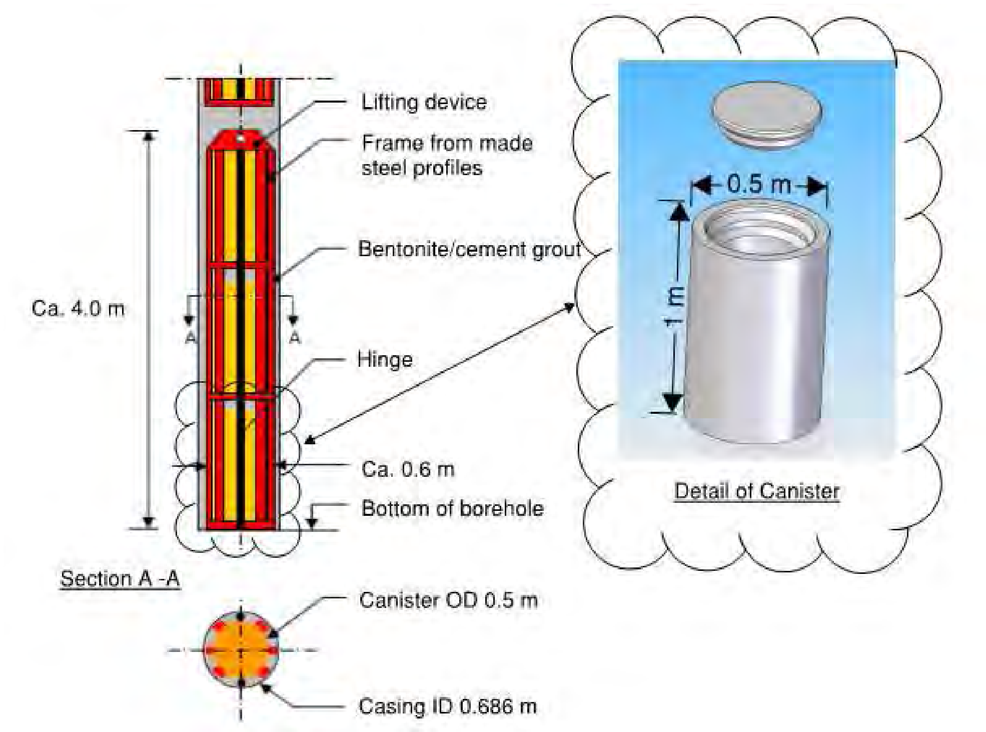


Figure 4.19 Canisters in steel cage. From IAEA (2009) and Danish Decommissioning (2008).

The total volume of a borehole will depend on both the depth of the borehole and the height to which the canisters are stacked. In IAEA (2009) 50 canisters per borehole are suggested, which is equivalent to 2750 litre per borehole. Allowing for 4.5 m between each cage this will give a height in each bore hole of 75 m. Accordingly, the minimum boring depth is (75+30) 105 m[8].

With a total number of 43 to 72 canisters of waste and a stacking height of 75 m (corresponding to 50 canisters) 2 boreholes are needed. However, it should be possible to stack all canisters in one borehole if deemed necessary.

The final layout is shown in the below Figure 4.20.

[8] The bottom of the borehole can be up to 300 m below surface

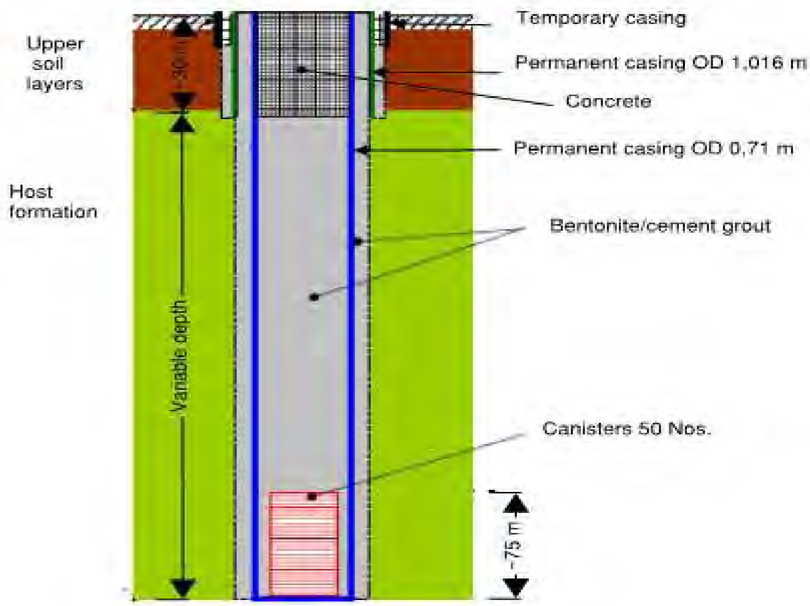


Figure 4.20 Borehole repository layout

4.10.2 Construction

The borehole is constructed using different techniques depending on soil/rock types and depth.

The most common method for deep drilling is the air-lift or reverse circulation systems. Reverse circulation borings are mostly done without casing except for a few meters at the top. The borehole is supported by means of the drilling mud. The drilling mud is pumped down through the drill string and carries up the cuttings from the drill bit back to the top in the annulus between the borehole wall and the drill string. The air-lift works in a similar way. Air is forced down the drill string similar to circulating drilling mud. The air mixes with water, and forms foam. The foam picks cuttings off the bottom of the hole and returns through the annulus to the surface

However, especially in the upper soil layers, it is often not possible to crush e.g. boulders and stones to finer cuttings as they tend to role around in the matrix of clay sand and drilling mud. Thus, normal dry drilling methods may also have to be used. Dry drilling is normally done with temporary casing which advances simultaneously with the excavation toll. Dry drilling may also be performed without casing if the formation is stable. Although it is called dry drilling, the borehole is almost always filled with water or drilling mud in order to avoid failure of the formation at the bottom of the borehole. However, dry drilling is much slower than reverse circulation drilling, as the drill string needs to be extracted for every run.

Pictures of a large dry drilling unit are shown below. The unit can be converted to a reverse circulation unit, ref. Schwank & Mielenz (2006).

Figure 4.21 Left: Bauer dry drilling unit; right: Bauer dry drilling unit fitted with casing oscillator.

When the borehole is at target depth the drill string is retracted and a permanent steel casing is inserted into the borehole. The permanent casing is grouted with a cement mixture in the annulus between the borehole wall from the bottom to the surface. The inside of the permanent casing is flushed with water until no drilling mud is present and the borehole is ready to receive the cages with canisters.

4.10.3 Operation of repository

Filling the repository with waste

It is possible to lower down the cage and unhook it from the surface using a simple wire rope winch mounted with a special hook. During lowering the borehole is always kept filled with water and the cage is centralised using centralizers mounted on the cage. When a cage is placed the annulus between the cage and the permanent casing is backfill/grouted using a bentonite/cement mixture. Once the mixture has cured (within 24 hours) the process is repeated.

When the desired number of canisters has been placed the borehole is backfilled to say 10 m from surface using bentonite (either as pellet or slurry) and at the rest with concrete.

Operation after closure

There will be no operation of the repository once closed except for monitoring, see section 4.10.6 below.

4.10.4 Closure of repository

When the backfilling is completed the repository is for all practical purposes closed and the site can be brought back to its original appearance.

4.10.5 Possibilities for extensions

Once the repository has been filled to its full capacity, there will be no possibilities for extension, and a new repository has to be constructed if required.

4.10.6 Monitoring of repository

The monitoring of a stand alone borehole repository can be assumed to include surface sampling and ground water samples, ref. IAEA (2009).

The monitoring shall be performed both prior, during and after construction/operation and comprises measurements of activity levels in air, surface soil surface water and ground water (samples from boreholes) with identification of the primary radionuclides.

A suggestion for a monitoring and surveillance programme is given in IAEA (2009), Appendix V.

4.10.7 Opening / emptying of repository

As mentioned, it would normally not be the intention to retrieve the waste once disposed of in a borehole repository.

5 Visual impact of the repository

As part of the assessment, the visual impact assessment of the facility is analysed. This analysis deals with aesthetic aspects related to different placement options in the Danish landscape and furthermore with the functional design, combined with aspects related to the future and present recognition of the facility as something special, in order to ensure that everybody perceives the facility as what it is and nothing else. Some aspects are considered to be relatively straight forward and therefore easy to assess and conclude upon, while others are more complex and will eventually be related to a specific location and require more attention at a later stage

The visual design is based on the technical design described in Chapter 4.

The repository is to be an integral part of a Danish landscape. An urban placement has from the start been ruled out as a very unlikely placement and is therefore not part of this analysis. Other unlikely placements are on the coast or directly in the littoral zone, because of the technical aspects and planning restrictions.

Prerequisites for the design have been:

- The repository has to be placed on a site equal to or less than 25,000 m².
- The repository has to be an integrated part of the landscape, and therefore limited in height, estimated to less than 10 m. This fact limits the visual impact in relation to common surrounding features.
- The repository has to be recognizable as a special plant, with a specific function.
- The visual appearance is to be inviting and visually pleasing and at the same time maintain strict security measures.
- The facility has to ensure good working conditions and easy maintenance.
- It also has to be as environmentally sustainable as possible.

5.1 Principles for the design of the repository

Recognition over time

It is of great importance that the repository can be recognized as something out of the ordinary over a vast expanse of time. This aspect of the visual appearance is therefore the backbone of the analysis. This aspect is, of course, difficult as can be seen during history where archeologists have tried to translate written language and symbols, with a varied amount of success.

Experience shows that if a text or subject matter is described in more than one language, it is less difficult to understand the meaning as time passes.

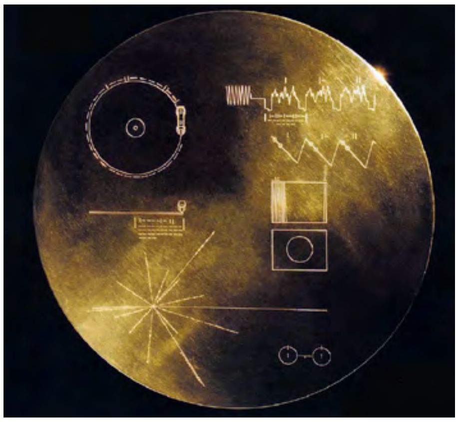
A feature of the repository could be a brief description, incorporated into the structure by letters, symbols, numbers, etc.

Other very recognizable structures representing imprints from times past are geometrical shapes in the landscape. Euclidian shapes are very different from the shapes seen in the naturally formed landscape, especially as seen from the air[9]. Local examples of this are the Viking fortresses, such as Trelleborg, Fyrkat and others. Nobody doubts that these structures have been used for a specific purpose, and that they are man-made. Therefore they are recognized as something to be aware of and even preserve for the prosperity.

Figure 5.1 Geometrical shapes of fortresses and other facilities of the past stand out, seen from the air.

If the repository is designed with an element of aesthetic value to the environment, there is an increased chance that the future authorities and the public will be more inclined to preserve and even appreciate the facility as something worth conserving and maintaining. The uniqueness of the facility can be further enhanced by designing the repository as something with an aesthetic value in itself.

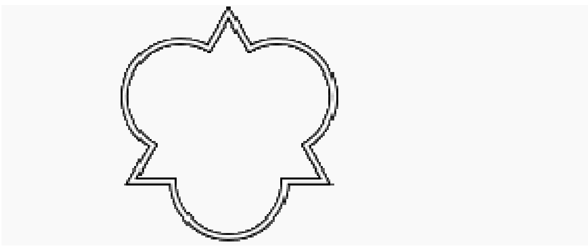
[9] Aerial archaeology is now an integral part of the continuing search for traces of the past.



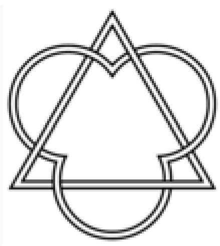
Recognition over time can be compared to communicating with alien civilizations. A good example of an attempt to communicate with other civilizations is the plaques attached to the Voyager satellites. If similar obvious symbols can be incorporated into the design of the repository, there is an increased chance that future generations will know the purpose of the repository. This could be as imprints into the concrete walls or as inlayed materials forming parts of the relevant formulae for some of the relevant processes.

Symbolic value

Trefoil is a feature used in Gothic architecture in the form of ornamental foliation introduced in the heads of window-lights, tracery, panelling, etc., in which the centre takes the form of a three-lobed leaf (formed from three partially-overlapping circles). A trefoil combined with an equilateral triangle was also a moderately common symbol of the Christian Trinity during the late Middle Ages in some parts of Europe. Two forms of this are shown below:

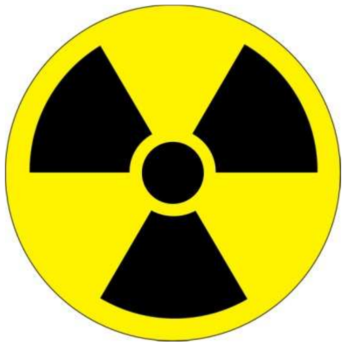


Outlined



Interlaced

The heraldic trefoil is a stylized clover and the universally recognized symbol for radioactivity is a trefoil. The international radiation symbol first appeared in 1946, at the University of California, Berkeley Radiation Laboratory. At the time, it was rendered as magenta, and was set on a blue background. The modern version is black against a yellow background, and it is drawn with a central circle of radius R, an internal radius of 1.5 R and an external radius of 5 R for the blades, which are separated from each other by 60°.



The internationally recognized symbol for radioactivity is unique and uniquely accepted all over as, not only a warning sign, but also as a symbol for radioactive materials as such.

To have a unique symbol for a specific subject matter is a great asset. Nobody doubts what this symbol means, and it is very unlikely that the meaning of the symbol will change or loose meaning in the foreseeable future. This is, of course, not a certainty, but it is a fact that most other meanings of symbols are interpreted differently in various cultures. This is not the case with the symbol for radioactivity. In the unlikely event that the symbol is replaced another symbol will take over, but there will be a long transition period where the existing symbol is recognized because of the universal acceptance and the wide distribution.

This is the reason that the symbol is chosen as the basis for the design of the repository.

5.2 Possible characteristic locations

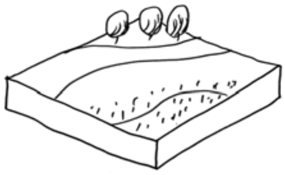
The Landscape Character is the particular expression that is created in the interaction between basic natural resources, land use and spatial visual conditions. It is the landscape character that makes a location stand out. The purpose of the Landscape Character Method (Caspersen & Nellemann, 2005) is to show the landscapes that especially must be taken care of, when the countryside is changed. This method is recommended to use in the process of choice of a specific location, see further in Part III.

Denmark has very few dramatic landscapes. The topography is always or nearly always only slightly varied with undulated hills or straight out flat, open spaces with more or less plantation.

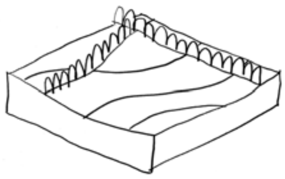
If placed in a wooded area the facility will be hidden completely from the surroundings being only visible from the immediate surroundings.. If the facility is visible from higher ground or structures, the visual aspects are to be considered as important similar to placing the facility in an open landscape.

Types of landscapes to be considered are e.g.:

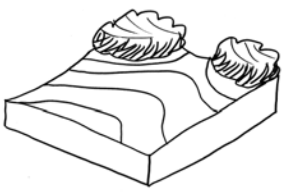
1 Flat, open farmland, meadow-like areas or moors with no or only low vegetation.



2 Slightly undulated hilly area with some, semi-transparent vegetation.

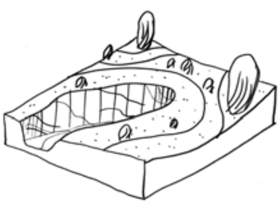


3 Hilly area with the facility placed upon a sloping area.



4 Complex area with more of the above mentioned characteristic features.

5 The island of Bornholm which is a geologically very special place in Denmark, because of the protruding bedrock.



5.3 Layout and design of the repository

A survey of intermediate and final repositories in other countries show that they are to a large extend erected as technical facilities, without much regard for the surroundings and no deliberate position regarding the layout or architecture, apart from the technical aspects. All of these facilities have many of the features in common with other industry based structures. This means that future generations will not necessarily be able to distinctly point out these facilities as something to regard, preserve or be aware of.

Most of the facilities stand out in great contrast to the surrounding environment. This makes the facilities stand out as sores in the landscape and subject to critic from the public.

The suggested design tries to take the landscape into account, while at the same time showing the repository as a unique feature.

Figure 5.2 A visualization of the repository in an archetypical Danish landscape. The repository is shown without the visitors' centre and treatment plant.

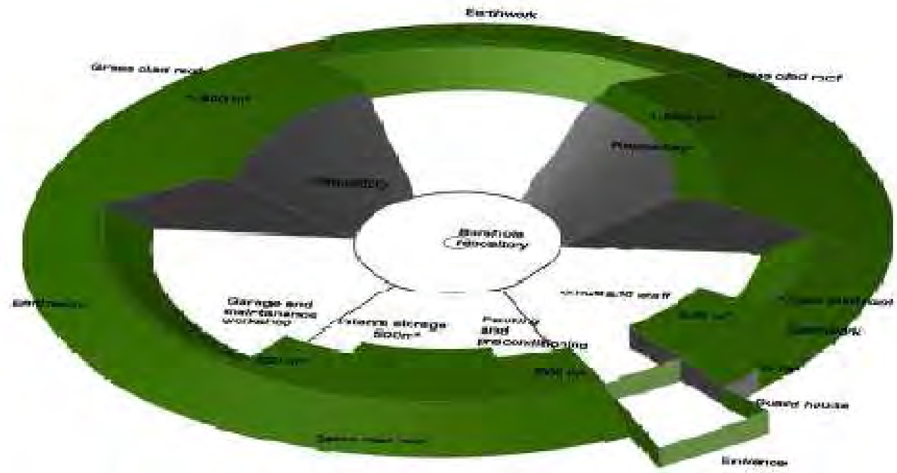


Figure 5.3 The basic model for the near surface repository.

Figure 5.3 shows a basic model for a near surface repository. All building volumes are integrated, as part of the earthworks and the roofs are clad in the same material as the earthworks e.g. grass. The repository itself is divided into two parts in order to fit into the trefoil theme of the symbol. As a variation, the repository itself can be united into one building volume. In order to fit into the trefoil theme of the symbol it fills one of the voids between the "petals". This also means that the area of the inner courtyard is larger and more versatile. Placement of the borehole is not critical to the design.

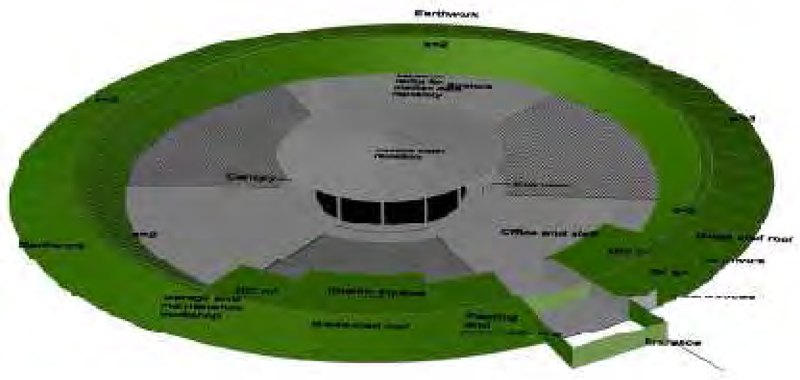


Figure 5.4 A basic model for a medium deep repository

A medium depth repository requires a canopy or building volume over the pit, here placed formally as centre of the symbolic plan, see Figure 5.4. All the smaller building volumes are integrated, as part of the earthworks. All proposed, possible sizes can be integrated into this draft.

As a supplement to the repository it may be desirable to build a visitor centre with audio-visual capabilities for e.g. visiting school classes. The building is to be of limited size, but big enough to facilitate the needs of up to around 25 persons.

Furthermore a treatment plant is needed. The spatial requirement is 2-300 m². The building encompasses facilities for solidifying, offices etc. Height is to be no less than 5 m for the workshop. Offices could be lower. The treatment plant could be part of the comprehensive layout, placed outside the perimeter within its own security zone, perhaps in connection with the visitor centre, or as an integral part of the plant, inside the main security zone.

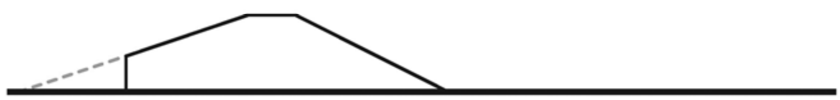
These facilities, and the repository itself needs parking space for the employees, delivery trucks and visitors.

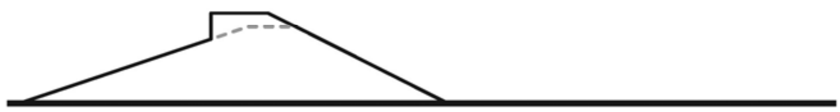
It is proposed that the visitor centre, the treatment plant with offices and the parking area form a supplementary part of the entire layout, outside the perimeter of the repository itself. It has to be integrated into the landscape which will dictate the final design of the repository and the supplementary buildings.

Preferably, the supplementary buildings form a marked entrance, to some degree a portal or landmark, again dictated by the surrounding landscape and the preferred appearance to the outside environment. Further details and possible design variations are shown in Appendix L.

The general layout is based upon the idea that the outer perimeter earthwork fences the repository, shielding it from the surroundings in a non-intrusive way, and at the same time creates a flexible inner area, where the work can take place unobstructed.

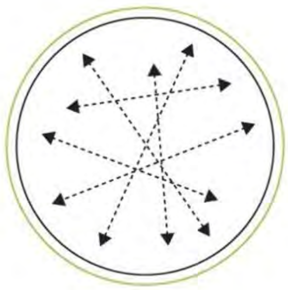
An outer security perimeter has to be established. This could be an integral part of the earthworks, combined with a wall.





The paving is meant to provide a solid basis for the heavy machinery that is to operate the facility and at the same time indicate the internal radius and the blades of the radioactive symbol. It is very important that the pavement is made from the most sturdy and long lasting materials available. The suggestion is to use non corrosive metal, granite and concrete to form the radius and blades and use a more degradable paving on the "voids" between the blades and the inner radius.

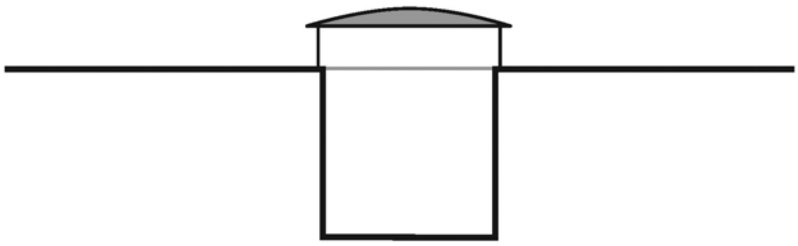
The circular layout of the facility provides for flexibility, both in terms of initial planning of the layout and during the active period of the repository. It also takes up minimum space relative to the surroundings, and it means that internal connections between the different functions are the shortest possible. These aspects of the circular design are of course less important than the need to separate the repository from virtually all other facilities. The only explanation for the "rational/square/perpendicular" design of other industrial plant is a very limited geometrical design process, without many considerations regarding aesthetic or even practical aspects. In this instance, it is a pronounced advantage that all other plants are laid out as variations of perpendicular grids, so that the repository can be seen as something out of the ordinary.



In a near surface repository, all buildings and the repository itself are an integral part of the earthworks that constitutes the perimeter. The buildings are lower than the earthworks, an integral part of the earthworks, or protrude from the slopes towards the centre of the facility. The roofing of the buildings are clad in the same material as the earthworks, grass clad, if the earthworks are grass clad.

The fencing of the facility could be placed outside the earthworks, at a distance or as a supplement to the earthworks. Perhaps the fencing could be integrated into the profile of the earthworks or perhaps electronic solutions could be considered, in order to dispense of the construction of a traditional wire fence.

If a medium depth repository is preferred the centre of the facility is laid out with the open pit covered with a canopy or a more closed, comparatively light structure, which will be removed, when the open period is over. The structure is relatively discreet, seen from the surroundings.



Technical structures are to be integrated in the repository to a large degree. The size of the technical structures, such as cranes should be lower than the earthworks, if possible, or retractable in order to minimise the visual impact when not active.

5.4 Adaptation to the surrounding landscape

As mentioned above the design of the repository can be changed in order to either be integrated fully into the environment or it can be designed to stand out as a separate element, geometrically, in contrast to the landscape.

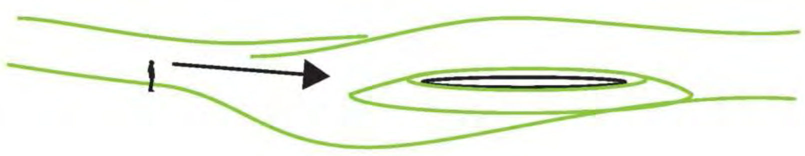
There are advantages to both solutions:

- If you integrate the facility as much as possible, nobody will take offence to the facility because of its "invisibility". On the other hand, the repository may be harder to detect in the future.
- If the repository is exposed fully to the surroundings and only to a small degree incorporates the features of the landscape, it will be easier for the future to be aware of the facility and maintain the necessary awareness.

The suggested solution is to design the repository with an outer perimeter that connects the facility to the surroundings and, at the same time, maintain the inner geometry. This design encompasses both a local adaptation and a possibility to recognize the structure of the repository for a very long time, seen from above.

If the repository is placed on a flat surface, the facility will be perceived as a small hill. The rounded shape ensures a familiarity with the natural shapes of the surrounding landscape. In this kind of environment it is recommended to uphold the outer perimeter as a purely geometrical shape. Adjoining buildings such as visitors centre and treatment plant can be placed near the entrance in order to form a "portal".

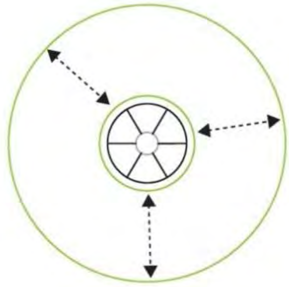
If the facility is placed depressed in relation to the surrounding landscape, the reference to the round Viking-fortresses are more obvious. The surrounding shapes of the landscapes may be integrated into the outer shape of the earthworks, if the inner shape is maintained. If the repository is supplemented with outside buildings, these structures could be placed overlooking the facility, at some distance from the site.



If the repository is placed on top of a hill or above a slope, the earthworks will make the facility seem like an integrated part of the landscape, seen from below. The outer perimeter ought to be shaped in accordance with the features of the surroundings. Complementary buildings can be situated at some distance, forming the main entrance to the site.

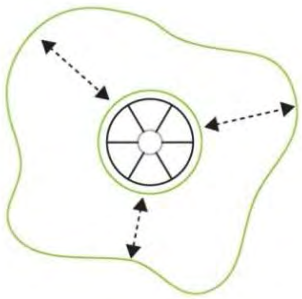
The size of the repository is related to the technical and functional demands, but the proposed design can be transformed without problems to encompass and adapt to the features of surroundings.

- If the outer perimeter is changed, the inner perimeter can be maintained.



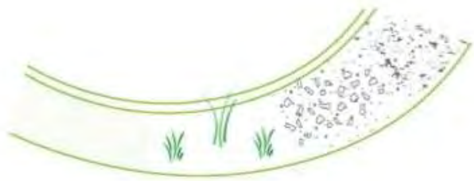
- If the surroundings demand a reduced height, the design can be altered, accordingly.

- If the integration in the surrounding landscape demands that the outer form changes into something more irregular than the proposed circle, this can be done, as long as the inner circle is maintained.



- The incline of the outer slopes can be altered, in order to either enhance the repository's presence or further integrate the facility.

- The slopes can be clad with grass, other kinds of plantation or even gravel, stone or other materials in order to adapt the features of the environment.

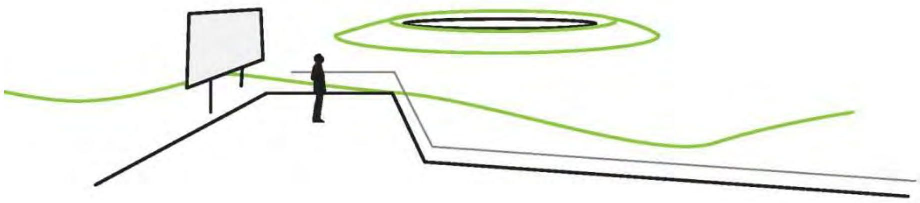


5.5 Public access and local visual appearance

The repository has the attention of the public, no matter what is done to the facility in order to make it blend into the surroundings. The subject matter of the facility is unique and has an evident educational value. This is why a visitor centre is suggested in order to facilitate the visitors' need for an interactive exhibition space, combined with more menial service functions. Although the security level of the facility is high, it must not indicate secrecy in a clandestine way. The connection between the visitor centre and the repository needs to be close and even intimate, even though it is placed outside the security perimeter.

The access to the entire facility should be easy and not hidden. The above mentioned adaptation to the surrounding environment should enhance the positive perception in the local area seen from the ground.

If the facility can be seen from a certain vantage point in the surroundings, a sculptural point of reference could be erected, along with an explanatory signing or a visitor centre could be built here.



If it is desired, a visitor centre could form a small landmark at the entrance, outside the perimeter. The structure should be designed along the principles of the repository, exposing the principles to the outside surroundings.

5.6 Planting of the repository

If the repository is placed in a landscape with little or no plantation, the earthworks ought to encompass the grass, soil or stones of the landscape. It is not advisable to introduce plants foreign to the landscape as part of the establishment of the repository.

If the surrounding landscape has a comparatively tall, linear plantation, the detailed site of the repository could be based to some degree on an attempt to link these features to the repository. The lines of “blades” of the inner pavement of the repository could be extended out into the surroundings in order to link to the linear features.

If the repository is placed in a dense plantation, the “outcrop”, has to be considered. The relation between the strict geometry of the earthworks of the repository and the outcrop has to be considered carefully. When placing an object (the repository) in a defined space (the outcrop) a symbiosis or contrast is the result. Local considerations or the final design of the repository will influence the decision whether the symbiotic or the contrasting solution is the right solution. The repository should be perceived as “a piece of furniture” in a fixed space, where the plantation constitutes the walls.

If lower plantation, such as shrubbery, tall grasses or reeds, is a major feature of the surroundings, patterns of these low plantations could protrude the perimeter of the repository in order to integrate the facility in the surroundings.

5.7 Sustainability

In order to ensure the largest possible degree of sustainability, as much as possible of the earthworks, the outer perimeter and the roofs of the buildings ought to be clad in the same, penetrable material as the surroundings.

Local handling of rainwater is to be part of the project, if at all possible. This means that the relation between hard and soft, penetrable surfaces should where possible allow rainwater falling on and around the repository can be dispersed of locally. This should of course take into account the necessity to minimise the amount of rain water intruding in to the repository itself.

Soil balance should be maintained as much as possible. This would imply reuse of the materials excavated for the repository to be reused on site, e.g. as part of the outer earthworks. Material could also be gathered in the near surroundings. Some material for the earthworks could be the result of the establishment of ditches or rainwater basin(s) leading surface water away from the repository.

All materials used ought to be long-lasting, non-degradable.

The entire facility has to be accessible for maintenance. Shrubbery may be planted on the earthworks. It is important that the species of plants chosen are suited for low maintenance, the ability to grow in the 1 m top soil of the earthworks, and at the same time prevent erosion.

5.8 Management of the appearance of the repository during the entire lifespan

During the open period within the first 30 years, the repository maintains the appearance as when established, perhaps with additions of minor building structures, internally and externally. The main impression of the facility remains to a large extent unaltered. In order to maintain the appearance, it is recommended that a maintenance plan is implemented. Parts of the facility are perhaps sealed off in stages during this period, but this ought not to have influence of the overall appearance.

It is recommended that the repository to a large degree keeps its appearance the next 30-100 years with the same perimeter as established from the beginning. The inner circle ought to be maintained and kept free of intruding plant growth, in order to be able to see the radioactive-symbol and perceive the impression of a special place with a special function.

In the years after this period it may be possible to uphold especially the interior circular courtyard and perhaps re-evaluate the security perimeter. The facility could be permitted to blend more into the landscape, integrating the outer slopes more into the surrounding features of the landscape.

It is important that the facility and the maintenance plan is mentioned in all records and databases possible in order to recognize the plant as something to beware of and preserve as pristine as possible.

6 Preliminary safety assessment, concept

The aim of the prefeasibility safety assessment is to serve as part of the basis for the choice between possible repository types. This encompasses comparison of repository types and of the geological settings that are relevant for each repository type. The repository types are thus the primary categories in the comparison, with the relevant geological settings making up the secondary categories.

The primary categories are compared through the comparison of the total number of hazards potentially arising from each repository type and of the overall risk of radiation impact related to these hazards and encountered by the defined representative person. Since especially the risk depends on the geological setting[10], separate evaluations of risk are carried out for a number of relevant geological settings (and depths) of each repository type.

The overall framework for the comparison of risks in the safety assessment is a hazard identification and an assessment of the hazards indentified.

One of the major risks connected with a repository for radioactive waste is the long-term leaching of nuclides due to the degradation of the repository construction with time. In probability terms, the probability of this occurring is 1 (or 100%); the question is when the leaching will commence, and when it will have reached a sufficient degree to cause impact in relation to the time it takes for the radioactive nuclides to decay. For this reason, the modelling of leaching of nuclides and heavy metals to surface and/or groundwater, and to the atmosphere in the case of gaseous nuclides, is a very important part of the overall risk assessment.

For this type of impact, an analysis has been carried out with respect to the influence on the impact of the possible variation in the parameters influencing the exposure of the reference person. Due to the generic nature of the assessment, the possible variation of the relevant parameters is often quite large. This will obviously not be the case for a specific setting of the repository, which is discussed as part of the evaluation. This is also the basis for the later recommendations with relation to more specific data collection of the parameters relevant for the safety assessment of a specific site.

[10] Due to difference in retention and relevant exposure routes

Other sources of risk are related to accidents causing release of radionuclides from the repository (intentional or unintentional). These releases can typically occur either to the atmosphere or to surface water. One of the features of the risk assessment of these types is the incorporation of uncertainty by means of Monte Carlo simulation.

Figure 6.1 gives an overview of the assessment process.

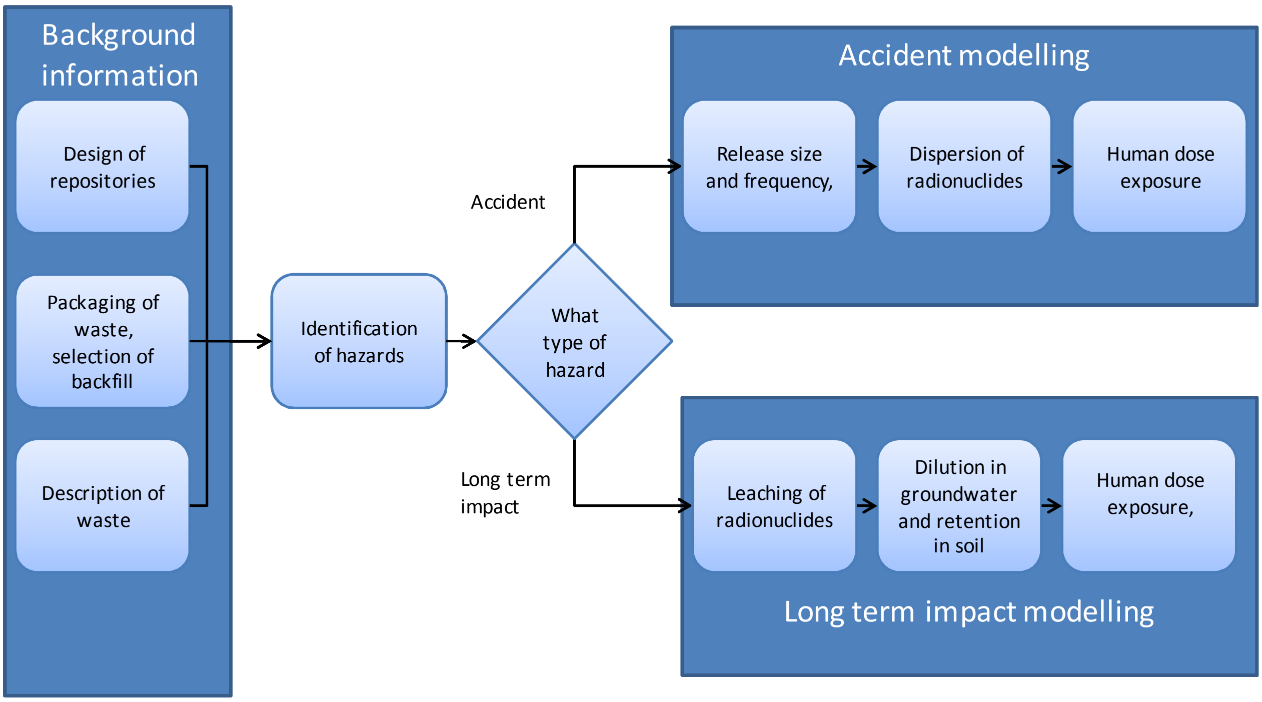


Figure 6.1 Overview of assessment process

6.1 The hazard identification and assessment process

A *hazard* can be defined as the potential to cause harm. A *risk,* on the other hand, is the likelihood of harm under defined circumstances and qualified by a statement of the severity of the harm.

During the lifetime of the repository, a particular hazard may arise due to an accidental event with unwanted consequences, but most likely it will not.

The analysis of hazards is in principle performed in two steps serving two different purposes:

- Selection of acceptable design[11] and location for the repository.
- Selection of the optimal design and location for the repository.

The purpose of the first step is to support the selection of acceptable repository types and settings. Some types and settings may cause an unacceptably high likelihood for appearance of unwanted consequences, i.e. the risk[12]. They are unacceptable and should be excluded from further analysis. The purpose of the second step is to support the selection of the optimal repository type and setting. In addition to the planned costs of construction and operation of the repository, also accidental events may involve costs. In this step accidental events having the risk of unwanted consequences expressed by means of costs are analysed.

In the prefeasibility study, the first step is the main issue due the generic nature of the sites and repository types that can be evaluated at this stage, and thus the large possible variation in impact. The second step is more appropriate as part of the following stages, where choice has to be made between specific sites. At that stage, the costs related to the repository can be evaluated in further detail and with a smaller span in uncertainty, and the actual number of potentially impacted persons can be taken into account. In the prefeasibility study, cost indications for relevant types of accidental incidents are given to indicate potential magnitudes of economic impact, see Chapter 10.

A hazard may be illustrated as a scenario. An initiating event starts a development towards some unwanted consequences for the exposed. In order to prevent this outcome, a number of barriers are established, as illustrated in Figure 6.2 below. The frequency of occurrence of the initiating event and the probability of failure of the barriers determine the frequency of unwanted consequences to the exposed. The hazard's risk is the combination of the frequency of occurrence and the severity of the unwanted consequences.

[11] This includes choice of waste types, packaging and fill

[12] Either through high likelihood of occurrence of accidents or though high overall dose due to the long term leaching from the repository (the planned exposure in ICRP (International Commission on Radiological Protection) terms).

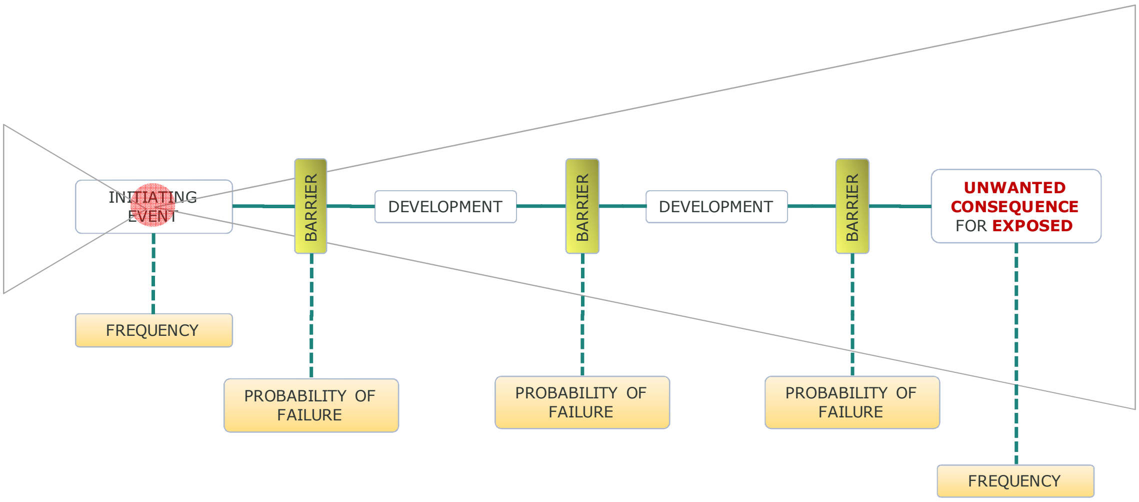


Figure 6.2 The process of development of unwanted consequences

There may be several causes for the initiating event, and the hazard may develop towards various levels of severity. To estimate the risk level of a hazard, in principle the risks of all sub-scenarios must be added. Normally, this may be simplified by assessing an overall value for the initiating event and selecting one worst case development as representative of all possible developments together with the likelihood of these severest consequences, as illustrated in Figure 6.2. The rest is then considered insignificant. The likelihood of the worst-case development is set conservatively, i.e. at a higher value than the fraction of developments having the exact severest consequences. This is to accommodate a reasonably large fraction of the developments with less severe consequences. In most conservative cases, this likelihood is assumed to be 1.

The nature of the hazards in the present project includes transportation processes. Harmful compounds or radiation may be withheld or transported trough various media before a fraction reaches the exposed. The fraction withheld by the various barriers may be a rather uncertain parameter requiring description by a distribution rather than by a single parameter to give meaningful results.

6.1.1 Hazard identification

A risk analysis is very dependent on a complete identification of all significant hazards. All hazards must be identified and addressed, and those constituting the highest risk must be analysed and evaluated.

Planning the hazard identification process requires a strict definition of which hazards to pursue, e.g.

- Who are the exposed?
- Which consequences are unwanted?
- And which hazards may cause such harm?

When these questions have been answered, the detailed hazard identification process can be planned as shown in Figure 6.3.

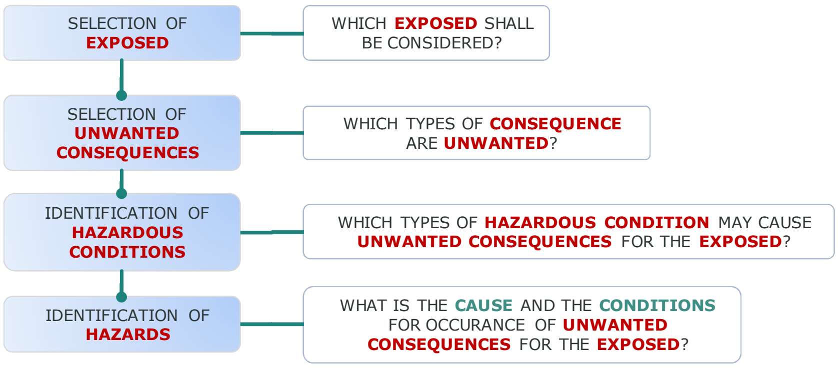


Figure 6.3 Planning the hazard identification process

In the preliminary safety assessment, the answers to the two first questions are defined as described in Chapter 6.3 and Chapter 6.4.

A hazard identification process has been carried out including a workshop with the participation of a group of experts and Danish Decommissioning. At this workshop, each participant contributed with his/her specific knowledge according to a previously prepared basic list of potential hazards. This has served as inspiration for the participants coming up with new hazards. The prompt list has ensured that all relevant types of hazard are considered.

6.1.2 Hazard screening

At the hazard identification workshop, all hazards are in principle considered to be important and to be evaluated in detail. The following analysis may show that not all hazards need to be addressed in detail. The probable occurrence may be considered insignificant, or the consequences immediately found to be negligible. When required, a set of rules is set up describing a procedure for fast assessment of the risk level. The rules define a risk level considered so low that no further hazard evaluation is required.

6.1.3 Detailed hazard evaluation

Each hazard constituting a significant risk is analysed in detail. The hazards, i.e. the accident scenario[13] have been described including:

- The initiating event and an assessment of the frequency of occurrence.
- Consequences considered and those exposed.
- Stages of development towards the consequences considered.
- All barriers preventing development of the hazardous event from one stage to the next and an assessment of the barriers' probability of failure considering the consequences in question
- Assessment of the risk of the consequences considered, based on the frequency of occurrence of the initial event, and the probability distribution of performance of the individual barriers.

[13] or the long term impact

The consequences of a hazard may have to be modelled in further detail, e.g. if the hazard causes spreading of nuclides to the geosphere and/or atmosphere with a subsequent dose to the reference person.

Based on the evaluation, an updated hazard list is drawn up encompassing all relevant hazards and associated consequences[14] and the probability of their occurrence.

6.2 Hazards relevant for this assessment

The repository has the potential to cause the following types of exposure, which may lead to unwanted consequences for the reference persons:

- radiation
- toxicity[15].

The nature of the two types of hazard, radiation and toxicity is somewhat different. The radiation level will decay over time, while the toxic elements will last.

The preliminary safety assessment primarily addresses exposure related to radiation, since this is considered the most important issue. Exposure to toxic substances is also evaluated with respect to resulting concentrations in the environment. These are then compared to either existing Danish drinking water criteria[16] or, where such do not exist, to typical natural background and/or other drinking water criteria. This is done, since values for acceptable intake of the heavy metals do not exist for all metals, and since the drinking water criteria are set to assure that acceptable intake is not exceeded, where such values exist[17].

Harm due to radiation at the repository may occur due to exposure to radiation originating in items at the repository. This may be of concern, if the barriers have become ineffective.

Harm due to radiation in compounds spreading from the repository may occur due to:

- inhalation of dust and gases containing radioactive isotopes (tritiated water vapour can pose a special problem).
- consumption of items or liquids containing radioactive isotopes.

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[14] including variability distributions, where relevant

[15] The term "toxic" is meant to cover all permanent (not decaying) harmful effects of elements and compounds in the wastes.

[16] The comparison to drinking water criteria is also carried out for tritium, since a Danish quality criterion for tritium exists.

[17] E.g. for cadmium and lead

- exposure to radiation originating in items near the exposed. This exposure may be enhanced if the radioactive items are accumulated in areas where the exposed are often located.
- water may be contaminated by both radioactive isotopes and toxic compounds.

The discussion above indicates that the hazards of concern are all of the same nature:

- The waste packaging and other man made barriers are damaged/not performing as intended, and the contents are released to the surroundings. By various ways of transportation, radiation/compounds reach the reference persons causing harm.

As described in the beginning of this chapter, the following is necessary to describe a hazard: an initiating event, various stages of development and the barriers preventing development.

6.2.1 Initiating events

The initiating event may be considered as the transition between 'normal' planned operation and the 'accidental event'[18]. In this study, the 'initiating event' includes damage to the conditioning of the waste in a way that either reduces the shielding effect towards radiation or allows the harmful elements and compounds to move away from the repository.

The frequency of the initiating event may be considered as the frequency of a particular action times the probability of failure of all barriers preventing development into an 'accidental event'.

The identification of the initiating events is performed systematically prompted by a basic list of guide words prepared for this particular project considering the nature of the hazards and those exposed. The list is structured considering:

- Operational mode
  - Storage of the waste
  - Operation and monitoring of the repository
  - Removal of waste from the repository
  - The repository when closed and no longer monitored.

The purpose of the analysis is to identify type-specific conditions to distinguish between repository types.

When the waste is stored in the repository, the containers will be exposed to the climate and conditions in the repository. The conditions of the waste containers will be monitored to some extent, while the repository is still open.

[18] which does not have to be an accident as such but can be the result of long-term processes in the repository

In case damage or potential causes for damage are recorded, additional safety measures may be taken. When the waste has been stored in the repository for a number of years, it may be decided to remove it. During storage time, the conditioning of the waste units may have decayed. However, the radiation hazard may also have been reduced. When the repository is closed, access is no longer possible, and the conditions inside the repository will no longer be managed and monitored.

6.2.2 Developments

The scenarios include stages of development from initiating event to exposure. When setting up scenarios focus has been on the location of the harmful elements.

Typical 'developments', i.e. locations on the elements' pathway from the initiating event to exposure of the 'representative person', would be:

- Conditioned in closed container (basic)
- In individual packing in open container
- Free as dust inside the repository
- Free as gas inside the repository
- Free in ground around the repository
- In water solution inside the repository
- In water solution in the ground near repository
- In groundwater
- In shallow groundwater abstraction wells
- In deep groundwater abstraction wells
- In a surface water recipient such as a lake, a wetland or a stream
- In a fiord or the sea
- In animals outside the repository
- In plants outside the repository
- In food
- As a gas in the open
- Dust free in the open
- On environmental surfaces
- On reference person
- "Inside reference person" (final).

Examples of scenarios are given below:

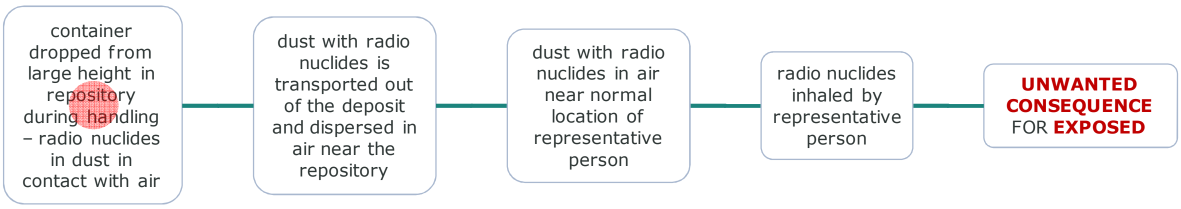

Figure 6.4 Storing and removing waste

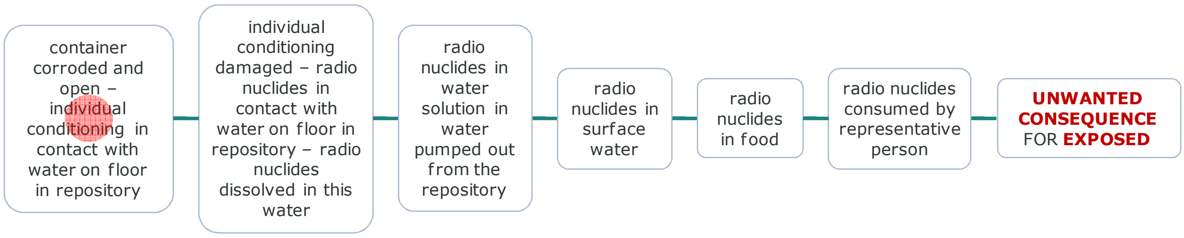

Figure 6.5 Repository in operation

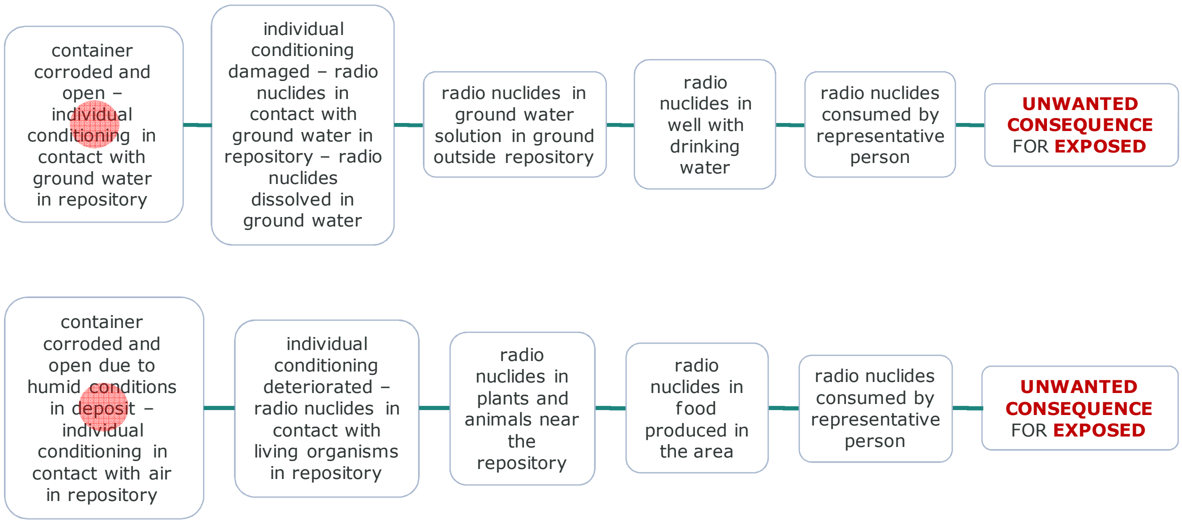

Figure 6.6 Closed and passive repository

It is assumed that the developments and thus the scenarios are relatively similar for the various repository types. What may differ is the frequency of the initiating event and in particular the distribution of performance of the barriers.

6.2.3 Barriers

A barrier is characterised by being unable to cause unwanted consequences by itself just by failing. The performance of the barriers working after the initiating event, i.e. during the 'accidental event' should be understood and quantified into probability distributions.

The barriers in between developments are to describe modes of transport from one development (location) to the next and their ability to stop, retain and disperse the harmful elements.

Personnel are located inside the repository, and thus they may be immediately exposed to the harmful effects of the waste even if contained. Neighbours (the reference person) are located at some distance, and mechanisms transporting the waste from the repository to the exposed are of great importance.

Barriers hindering exposure to radiation would be:

- Conditioning
- The repository's structures, e.g. the concrete walls
- Filters in ventilation and water discharge systems
- The ground around the repository
- Distance
- Building
- Time (decay of nuclides).

Barriers retaining the spreading of elements and thus emphasizing the effect of time with respect to considering the radioactive isotopes would be:

- Retention in the repository
- Retention in the soil matrix
- Retention in other organisms.

In particular, the soil around the repository may become a barrier of major importance. The quality of this barrier is very dependent on geological and hydrological parameters and thus the location of the site.

However, time may also degrade the effect of barriers. In time, barriers may fail unnoticed due to various activities. Both at present and in the future, when the exact location and use of the repository may have been forgotten, the following issues are relevant:

- Deterioration of structures due to frost and thaw.
- Compression of earth above the repository.
- Damage due to future construction/excavation activities.
- Damage due to nearby activities for extraction of raw materials.
- Damage due to earth movement, earthquakes etc.
- Changes in groundwater level.
- Flooding e.g. due to rising sea level.

Decay of barriers with time is important when considering transport of the compounds in the (far) future.

Barriers dispersing the elements would be:

- Wind
- Ground water
- Surface water.

These barriers will in general reduce the level of exposure for the reference persons, but are also be means of transporting the compounds to this person.

The safety assessment for the different repository types[19] includes the probability of the release scenarios, the nuclide mass in the release, the exposure of the representative individual from all exposure pathways and the uncertainty of the model parameters.

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[19] including the related geological setting, waste amount and fill properties

6.3 Selection of the exposed

Radiation exposure is to be evaluated for a representative person, which is the term that has replaced the term critical group in the ICRP recommendations. In line with these recommendations, the representative person may be hypothetical, but the habits used to characterise the person should be typical for a small number of individuals representative of those most highly exposed i.e. not the extreme habits of a single member of the population. The habits encompass issues such as consumption of food items, breathing rate, location, usage of local resources and other behavioural aspects that can possibly influence the yearly radiation dose (ICRP, 2006 & 2007).

Presently, there are no dose limits with respect to protection of the environment, although it has been on the worktable of the ICRP for some time.[20] Similar to humans, a concept of Reference Animals and Plants is presented in ICRP Publication 108 (ICRP, 2008).

It has been decided[21] not to include a specific evaluation of impact to the environment at this stage due to the generic quality of the study. A short description of possible methodologies will be included in the overall description of the biosphere modelling.

Likewise, a specific evaluation of the possible impact on the personnel involved is not included in the assessment, since such issues typically fall under the overall systems for occupational safety, similar to the present systems at the nuclear facilities at the Risøarea. Specifically with respect to the possible accidents related to the initial storage phase and possible removal phase, potential hazards will be listed and possible differences between the repository types will be pointed out.

6.4 Acceptance criteria

Radiation

When the establishment of a repository for radioactive waste in Denmark was decided, the principles for assessment of the radiation exposure of the general public were set up (Ministry of the Interior and Health, 2008):

Pre-closure[22] (operational phases):

- In general, all doses should be kept as low as reasonably achievable (optimisation of protection)

[20] A generic screening value of 10 microGy per hour is suggested in: Andersson et al (2009): Protection of the environment from ionising radiation in a regulatory context (protect): proposed numerical values, Journal of Environmental Radioactivity, 100, pp 1100 - 1108.

[21] In a meeting with Danish Decommissioning

[22] And in case of later removal of the waste from the repository

- No person shall be subject to doses exceeding the dose limits set in the government order nr. 823, October 31st 1997, on dose limits for ionising radiation.

Since dose limits are valid for all potential radiation sources, a specified fraction of the dose limit for radiation exposure from the repository has to be set (a dose constraint for planned exposure and a risk constraint for potential exposure). The dose constraint for members of the public is suggested to be 0.1 mSv per year, which is in line with the requirements set for the nuclear facilities at Danish Decommissioning.

Post-closure[23]:

- The expected and potential exposure to radiation shall be minimised taking societal and economic considerations into account.

- The dose constraint for members of the public for releases of radionuclides from the repository should be set at 0.01 mSv per year.

- The dose constraint for potential events as basis for planning and approval of the repository should be set to 1 mSv per year.

In both cases, optimisation of protection should still be an issue below the set constraints, in line with the ICRP recommendations (ICRP, 2007).

Heavy metals

Danish drinking water quality criteria exist for lead and cadmium (Ministry of the Environment, 2006) to which the modelled concentrations in the well water will be compared for the different scenarios:

- Cadmium: 2 µg/litre
- Lead: 5 µg/litre.

No Danish criteria exist for beryllium and uranium. The US EPA has set a criterion for beryllium, which is suggested used (US Department of Health and Human Services, 2002). For uranium natural background in Swedish quaternary deposits are suggested used as a comparison (IAEA, 2005a). This leads to the following suggested criteria for beryllium and uranium:

- Beryllium: 4 µg/litre
- Uranium: 0.001 µg/litre.

6.5 Time frame for the assessment

The time frame for the assessment is, as for the pre-feasibility study in general, as follows:

[23] With or without monitoring

- Initial storing of the waste, year 1,
- Operation of the repository, the first 30 years,
- Removing waste, some time within the first 30 years,
- Monitored period, at least 30 years
- Post closure, 30 to 10,000 years for accidental events and up to 1,000,000 years for long term impact.

The post closure phase is assumed to encompass a monitoring phase including the first 100 years. Knowledge of the repository is assumed to be lost after 300 years. During the post closure period, evaluation with regard to potential accidents is also made for year 100, 300, 1,000, 3,000 and 10,000. The very long range for the long term impact is required to examine the risk due to long lived α-nuclides. The comparison of the resulting concentrations of the heavy metals is

7 Assessment of long term impact:

In the preliminary safety assessment, long term impact from the different possible repository types placed in the different possible geologies is evaluated. As mentioned previously, both the repository types and the geologies are generic at this time, which results in a large possible variety in the parameters that influence the calculated impact on the chosen reference person. At a later stage in the decision process, the safety assessments can be carried out based on a more narrow span of parameter values relevant for the specific combinations of repository type and geology.

As part of the hazard identification, a number of incidents related to long term impact have been identified. These incidents are listed Table 7.1.

Some of the hazards may only be relevant to some of the repository types. Relevant combinations of hazards and repository types are indicated by an "x", no relevance by an "-". A number of hazards do not pose an immediate risk, but may cause barriers to become ineffective e.g. by speeding up corrosion. This is indicated by a "b".

The incident numbered D-4 in the hazard identification list is the general long term impact caused by leaking of nuclides to the groundwater zone and thus the situation modelled in the geosphere model, while a number of other incidents:

D-3, D-5, D-6, D-12, and D-14

are variations of this primary model caused by changes in the time span before leakage from the different repository types occurs. . For above and near surface repositories some incidents may develop so that nuclides leaks to surface water, these are considered to be accidental events (see Chapter 8). D-11 considers the gradual release to air due to diffusion of gaseous nuclides.

The safety assessment of the listed incidents related to long term impact is described in this chapter. As mentioned previously, the variability in the parameters governing the dose received due to long term impact is quite large[24]. The variations in time, where the leakage occurs described in the 5 incidents listed above can therefore at this stage be assumed to be included in this vary large variation, and separate calculations have therefore not been carried out.

[24] The actual orders of magnitude are commented on in the following sections.

Calculation of long term impact due to leakage to ground water is described in Sections 7.1 to 7.6. Dose due to release of gaseous compounds is described separately in Chapter 7.7. Overall results are presented in section 7.8.

Table 7.1 Incidents evaluation long-term impact and variations herein relevant for the different repository types. ASR = Above Surface Repository, NSR = Near Surface Repository, MDR = Medium Deep Repository, BORE = Borehole.

#	Description of hazard	ASR	NSR	MDR	BORE
D-3	The groundwater level may become so high that the level fluctuates in the zone where the waste is located. Compounds are dissolved in the water. Neighbours are exposed via water and biota	-	x	-	-
D-4	The repository becomes water filled by groundwater. The nuclides etc. are dissolved in the groundwater and transported out of the repository	x	x	x	x
D-5	The pressure in the repository increases due to the formation of gases. This damages the repository containment and the process of deterioration is enhanced	b	b	b	b
D-6	The top membrane in an above/near surface repository is damaged and surface water flows through the repository dissolving nuclides etc. Neighbours are exposed	x	x	-	-
D-11	Gaseous compounds diffuse through the repository containment and into the ground above. Neighbours are exposed	x	x	-	-
D-12	The repository containment is damaged due to settlements, ground movement, earthquakes or like. Cracks allow more water to flow through the repository structure	b	b	b	b
D-14	Sea level rises and removes repository. Compounds are dispersed by and dissolved in the water	x	x	-	-

In Chapter 6 the term reference person was introduced. At this stage, the safety assessment is carried out as an evaluation of the possible long term impact on such a reference person.

Since one of the main aims of the prefeasibility study is to make recommendations with respect to choice between repository types and generic geological settings, the use of a reference person for the comparison is deemed the most appropriate approach at this stage.

At a later stage, more specific knowledge of the actual possible exposure routes relevant at a specific location can be taken into account as well as the knowledge of how many people actually can be impacted.

7.1 Potential exposure from long term impact

In the long term (after at least 300 years), both the packaging containing the waste and the repository construction will have begun to degrade, even if careful measure is taken to postpone this as long as possible. This can result in exposure of the reference person. Exposure can be internal, i.e., through ingestion or inhalation of radionuclides or heavy metals, or external, i.e., dose due to staying in a contaminated environment. Exposure can occur through impact from a number of exposure routes:

- Use of groundwater for drinking water for humans[25]
- Use of groundwater for drinking water for animals
- Use of groundwater for watering of crops
- Eating of freshwater fish from streams and lakes
- Eating of fish, crustaceans and shellfish from the sea
- Inhalation of vapour and (resuspended) particles
- External exposure while staying outside.

7.1.1 Internal exposure

Internal exposure is always preceded by intake of radionuclides into the human body. This can occur by inhalation of contaminated air, ingestion of contaminated water and food, and by skin penetration, which as a simplification is not considered here, since the relative impact of this exposure route is considered small compared to the other exposure routes. [26]

Inhalation of contaminated air can occur both indoors and outdoors, although indoor exposure will be lower due to the filtering effects of the building.

Radionuclide contamination of air can come both from resuspension of soil particles and the release of nuclides bound to particles or aerosols and from elements that exist in gaseous form (e.g. radon and ^{14}C as CO_2 if present).

Exposure from oral intake of radionuclides through water and food will depend on the fraction of food and water consumed that is contaminated, and the level of activity in this part of the foodstuffs and water. Diet composition can also have an impact on the internal exposure, as different food types can have different contamination levels.

[25] As mentioned previously, this is the exposure route used to evaluate the heavy metals since criteria exist (or are suggested) for the comparison of exposure via this route. These criteria are originally set to assure the overall intake of the compounds is not above allowable intake.

[26] This may not be true, e.g. for tritium, which should be evaluated in greater detail as part of the safety assessment of the specific locations.

7.1.2 External exposure

External exposure is a consequence of radiation emitted by the radionuclides in the surrounding environmental media, air, water and soil. Exposure to radionuclides in the air and in the ground will be lower indoors due to the shielding effects of the building[27]. It should be noted that dose assessments considering normal operation or release from a deep geological repository often conclude that the external dose is insignificant in comparison with the internal exposure.

In the comparison in the prefeasibility study, doses will be calculated for adults only. In further assessments, assessment should be made for all the age groups defined by the ICRP.

7.1.3 Exposure assumptions

When performing a dose assessment during a prefeasibility study, where neither location nor repository concept are chosen, and where the initial assessment of the waste is also fairly basic, a number of assumptions need to be made about living habits and exploitation of the landscape by representative individuals inhabiting the site. To secure a conservative/cautious approach, it is assumed that the reference person exploits the contaminated landscape maximally, thus consuming all potentially edible food produced (see for instance SKB, 2006).

Other situations can easily be assessed at a later stage by introducing corrections to account for the fraction of consumed water and food that is not contaminated. In this analysis, human consumption rates are based on Fødevaredirektoratet (2002), Nielsen & Andersson (2008) and DANVA (2009), where available, and otherwise on estimates based on Swedish data, see Table 7.2.

[27] A number of safety assessments carried out in Sweden and elsewhere have shown that the only external exposure contributing significantly to the total long term dose is from contaminated ground (Bergström et al, 1999).

Table 7.2 Human consumption rates used in the dose calculations

Parameter	Unit	Value	Reference
Water	l/year	550	Based on the recommended 1.5 l/day
Milk	l/year	96	Nielsen & Andersson (2008)
Meat	kg/year	34	Nielsen & Andersson (2008)
Vegetables & fruit	kg/year	98	Nielsen & Andersson (2008)
Root crops/potatoes	kg/year	60	Nielsen & Andersson (2008)
Cereals	kg/year	75	Nielsen & Andersson (2008)
Soil	kg/year	0.01	Bergström et al (1999)
Fish	kg/year	6	Fødevaredirektoratet (2002)
Crustaceans	kg/year	1	No data, assumed half of the Swedish consumption
Inhalation rate	m^3/h	1[28]	ICRP 101

The ingested food and milk is all assumed coming from cattle, and in the calculation of exposure, the data concerning cattle's intake of different foodstuffs are shown in Table 7.3.

Table 7.3 Consumption rates for cattle used in the dose calculations

Parameter	Unit	Value	Reference
Water	l/day	70	Bergström et al (1999)
Pasturage	kg dw/day	12.5	Nielsen & Andersson (2008)
Cereals	kg/day	13	Nielsen & Andersson (2008)
Soil	kg/day	0.1	Bergström et al (1999)

It is further assumed that the representative person is exposed to outdoor conditions 20% of the time. In safety assessments conducted for the specific locations, an evaluation of the importance of this assumption should be made.

A shielding factor of 0.35 is included in evaluation of external exposure indoors.

Possible effects of food processing are not included.

[28] Rounded up from the figure 0.92 m^3/h

7.2 Overall model for the groundwater route

In order to determine the long-term risk of leaching from the repository, a modelling framework is applied to assess:

- the (risk of) leakage/release from the repository (the Repository Model)
- the (risk of) transport to a representative person (the Groundwater Model)
- the (risk of) exposure to the representative person (i.e. radiation dose assessment) (the Dose Model).

Parts of this modelling structure have also been used to model consequences of some of the types of accidents that can occur. This is further described in Chapter 8, where a description of other modelling tools used in the assessment of consequence for other types of accidents is also given.

As mentioned above, the probabilities of the occurring events are more or less all 1 in the case of the long-term leaching from the repository. Here, the modelling mainly centres around the evaluation of a probable time scale for the deterioration of the packaging and the repository structures, and the possible variability in the parameters describing the possible flow through the deteriorated units, the uncertainty describing the transport through the geosphere , and the uncertainty describing the exposure of the representative person. Some of the accidental incidents may - with a certain probability - alter the probable time where the repository units deteriorate and thus, as a consequence, alter the overall exposure.

In the modelling framework assessing the long-term leaching from the repositories, the following modelling steps are carried out for each type and placement of the repository:

Possible leakage from the repository is modelled based on evaluation of the deterioration of the repository elements (e.g. drums, containers, concrete walls) and seepage through backfill materials. This repository model calculates the best estimate and possible variability distribution of the source term [29]in the groundwater model.

On this basis, the groundwater model determines the resulting nuclide concentration in the various recipients involved in the risk assessment. The groundwater model will determine the retardation and dilution of various species taking their geochemical characteristics into account. The approach related to groundwater modelling may be called semi probabilistic, since a deterministic groundwater model is used to run a number of likely scenarios with varying geological and hydro-geological settings and parameters within likely intervals. Parameter variability will be assessed through sensitivity analyses, which can give an estimate of the uncertainty of the resulting concentrations.

[29] As nuclide concentrations

The dose model uses the concentrations from the groundwater model as inputs to the calculation of exposure through water-related exposure routes (e.g. drinking water and impact on fish used for human consumption). The overall concentrations are then for the radioactive nuclides converted to radioactivity based on the activity related the each nuclide. To these inputs is then added exposure due to direct radiation from soil and due to gaseous nuclides (e.g. radon). Finally, these concentrations are converted into dose to allow for comparison with the relevant dose limits and constraints for the radioactive nuclides. The heavy metal concentrations calculated for the relevant wells are compared to Danish drinking water quality, where these exist, and otherwise to natural background or drinking water quality for other countries.

In order to make the different parts of the modelling framework clear and manageable, decay of the radionuclides is handled in the final coupling of the models. This also makes it easier to adapt model results to changes in time scale of release etc. due to occurring accidents as mentioned above.

7.3 Repository model

The aim of the repository model is to:

- Provide a true and fair source term for assessment of the absolute dose received by a neighbour
- Provide a true and fair representation of important phenomena like location and geometry of repository, durability of repository structures, durability of packaging materials, water solubility of radionuclides, retention of radionuclides in fill and backfill materials
- Provide a model able to represent differences in the following parameters:
  - Repository location with respect to geology
  - Repository location with respect to depth
  - Repository geometry
  - Repository structures and membranes
  - Repository backfill
  - Packaging materials for waste
  - Fill in waste units.

Because the various radionuclides have very different retention times in the repository, the model keeps track of individual nuclides. The subsequent groundwater model considers the additional travel time for each nuclide and its daughters to the various recipients. Consequently decay can only be handled, when the total retention and travel time to specific recipients is known. Thus decay is handled in the subsequent dose model only.

The majority of the radionuclides will be retained in the repository and the surrounding soil layers for hundreds if not for thousands of years before they reach a recipient in the form of a well, a stream or the sea. A few radionuclides e.g. H-3 and C-14 will not be retained due to their low sorption capacity (see Appendix B, Section B.1.1).

The leakage and release of radionuclides from the repository encompasses complex physical and chemical processes, which depend on both the type of waste and the construction of the repository and its barriers. The actual waste to be considered in this study varies considerably in type and packaging. The suggested model below for the release of nuclides from the repository is a generic, simplified, and conservative model.

Table 7.4 Simplified model for the release of nuclides

Scenario	**Period**	**State of the engineered barrier**	**Assumed flow through the repository**
Normal evolution	Active institutional control (x years)	Intact	No flow
	Passive institutional control (y years)	Gradual degradation	Increased flow from 0 to A m^3/year
	Post institutional control (z years)	Complete degradation of barrier	Maximum flow of A m^3/year

x,y = number of years of the periods. x is suggested to be 30 years, and y is suggested to be 300 years (after the initial filling of the repository)
z = year after closure of the repository. The assessment period has been set to 10,000 years
A = water flow through the repository

The repository model combines information on waste, packaging materials, repository and soil in order to generate a source term describing the contents of radionuclides in the groundwater flowing through the repository versus time.

The repository model represents the repository as a simple geometrical shape (box or cylinder) located in a uniform soil. When the groundwater level inside the repository is no longer controlled, the repository becomes water filled[30]. Clean water flows into the repository through its entire cross section with a flow rate determined by the soil and the repository structure and its contents. Inside the repository the water initially causes the waste packaging to corrode and deteriorate. After some time, the packaging will open, the water will get into contact with the waste and the nuclides will begin to dissolve. If the repository is backfilled, or the waste units are filled, the dissolved nuclides will be adsorbed to this material and retained in the repository until the adsorption capacity is exceeded (breakthrough). After breakthrough, the water will begin to transport the radionuclides away from the repository. In the model, each nuclide of interest and each type of waste unit is considered separately both considering contents, time to deteriorate, and fill.

[30] In case of the below ground repositories, while the above ground repository will have water leaching through it with time, at the membranes deteriorate.

The model combines sub models for deterioration of structures and packaging materials, water flow and flux through the different repositories, the solubility of the specific radionuclides[31], the amount and fraction of radionuclides versus all nuclides of the element, and the retention of individual elements in fill and backfill materials. An overview of type of data and sub models in the repository groundwater model is presented in Figure 7.1.

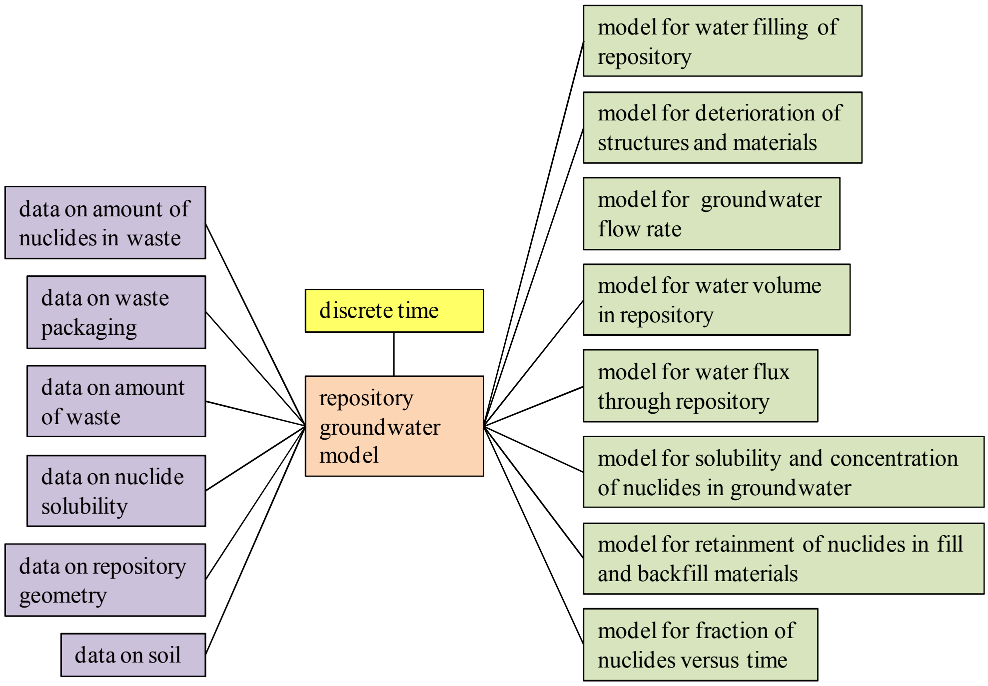

Figure 7.1 Overview of data and sub models in the repository groundwater model

The processes thus considered for the leakage of radionuclides are water flow[32], solubility[33], retardation due to sorption on different fill and backfill materials and decay of the nuclides.

7.3.1 Scenarios

When stored in the repository the distribution between waste units sent to the borehole (B) and waste units stored in a near surface or medium deep repository (A) may vary. 4 combinations of scenarios have been considered (see Chapter 3 for description of the different waste types):

[31] Dependent of the geochemical environment created by e.g. the fill and backfillmaterials
[32] as a function of fill material if any and the hydro geological properties of the surrounding soil types.
[33] as a function of waste mix and combinations with fill materials if present.

Scenario 1A: Waste types 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12a, 13, 15, 16, 17, 18, 19, 21
Scenario 1B: Waste types 12b, 14

Scenario 2A: Waste types 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12a, 13, 19, 21
Scenario 2B: Waste types 12b, 14, 15, 16, 17, 18

Scenario 3A: Waste types 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17, 18, 19, 21
Scenario 3B: Waste types NONE

Scenario 4A: Waste types 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12a, 19, 21
Scenario 4B: Waste types 12b, 13, 14, 15, 16, 17, 18.

Figure 7.2 presents the activity of the total amount of waste versus time. For each of the 4 options the two graphs A and B encompass the entire amount of waste.

It can be seen from Figure 7.2 that in particular waste types containing U-238 and U-234 contribute to the long term activity of the waste. Thus also sending waste of type 19 (non irradiated uranium) to the borehole would further reduce the long term activity of waste in the near surface/medium deep repository. However, the tailings (type 21) and some of the waste from the water treatment (type) that also have a considerable amount of these nuclides cannot be sent to a borehole.

The results of the model for each calculated scenario (encompassing geology, repository type, waste types and amounts, fill and backfill types[34], and other installed barriers) is a set of parameters indicating the amount of each nuclide released per year, when the release starts and when the total amount has been washed out if relevant.

The repository model is run both deterministically based on most likely values and probabilistically with uncertainty distributions for the relevant parameter to form the basis for Monte Carlo simulations resulting in aggregated distributions of the resulting parameters constituting the source term.

More details about the model is given in appendix D.

[34] The distinction being that fill is the material added in the drums and containers, and back-fill being the material that is added around the containers.

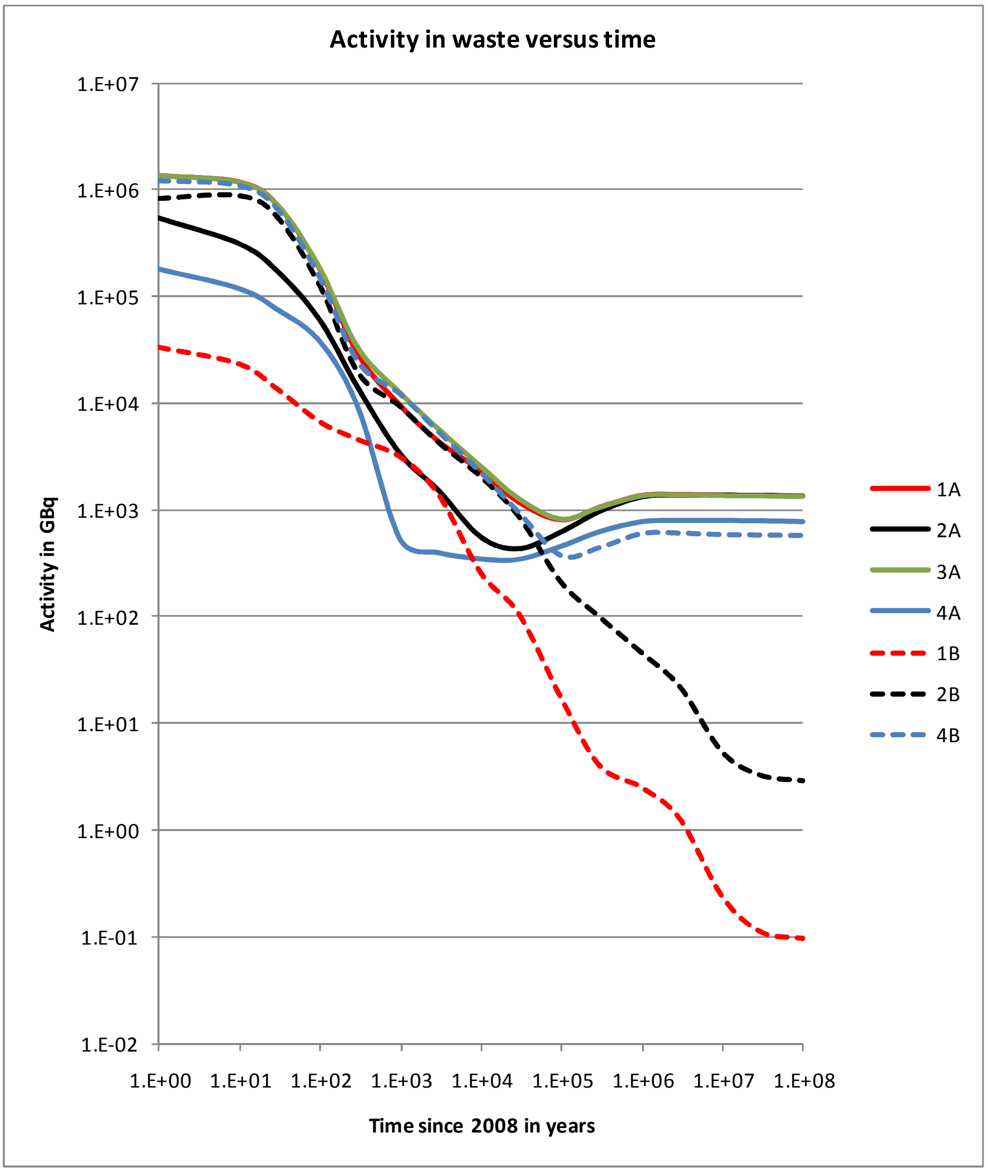

Figure 7.2 Activity of the total amount of waste versus time

The repository types evaluated in the repository model[35] are shown in Table 7.5.

[35] And the safety assessment as such

Table 7.5 Repository types included in the safety assessment. For further detail see Chapter 4

Design #	Repository
1	Above surface repository
2	Near surface repository
3	Medium deep repository, shaft operated from ground level, irreversible, Ø 33.8 m
4	Medium deep repository, shaft operated from ground level, irreversible, Ø 26 m
5	Medium deep repository, shaft operated from ground level, irreversible, Ø 18 m
6	Medium deep repository, shaft operated from ground level, reversible, Ø 33.8 m
7	Medium deep repository, shaft operated from ground level, reversible, Ø 26 m
8	Medium deep repository, shaft operated from ground level, reversible, Ø 18 m
9	Medium depth repository, shaft operated inside repository, Ø 33.8 m
10	Medium depth repository, cavern operated inside repository, 4 fingers Ø 8 m
11	Borehole repository

7.3.2 Model results

The model calculates release of concentrations of nuclides, which are then used as input to the groundwater model.

As intermediate results of the repository model, total release of activity per year can be shown as a means of comparing releases from different repository types and with different types of fill and backfill. These results of the repository groundwater model are of an intermediate character with no physical interpretation (GBq/year), (because decay is not considered yet and it is not the concentration of the nuclides). However, the results[36] provide preliminary information on the scale of the benefits of backfill and fill materials. Figure 7.3 is an example, showing the usefulness of bentonite as backfill and fill material in waste units. The bentonite has several effects all working to reduce the amount released. When the repository is located in types of soil with a higher hydraulic conductivity than bentonite (clay and limestone) the material serves to reduce the flow rate. Also the bentonite takes up space and reduces the overall porosity of the repository. Finally the bentonite retains nuclides in varying degree due to sorption. More examples are included in Appendix D.

[36] It is emphasized that these sums of activity only serve the purpose of examining and illustrating the qualities of fill and backfill materials. They are not part of any dose estimates

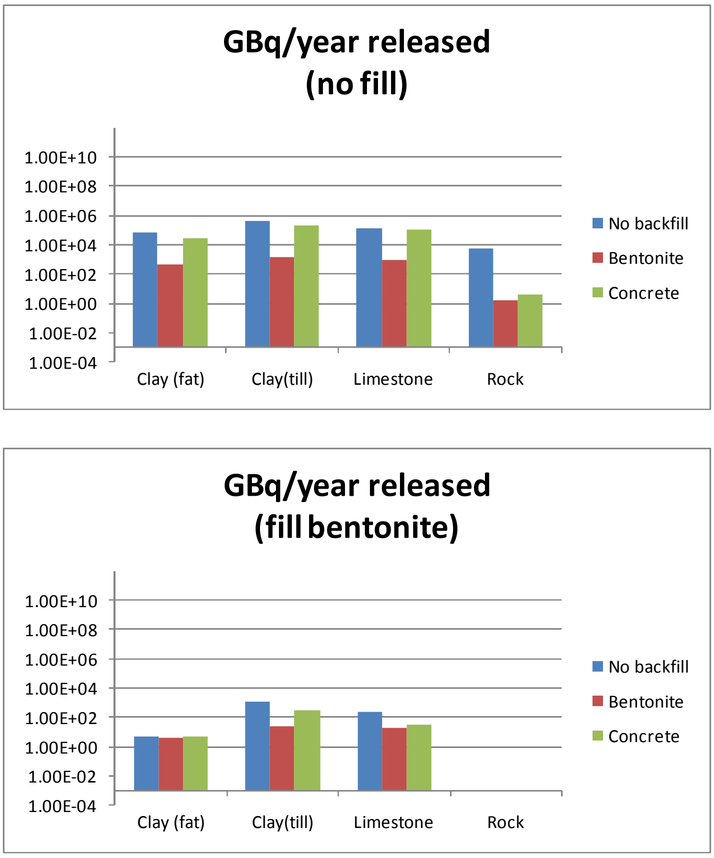

Figure 7.3 Example comparing the sum of the source terms for a repository with no fill in waste units and a repository where bentonite is generally used as fill (intermediate result not taking decay into account)

Results with respect to the differences in radionuclide release between the repository types can also be extracted, see Figure 7.4. The figure presents the sum of the source terms for the near surface (Repository type 2), the shaft operated medium deep repositories (Repository type 3, 4 and 5, irreversible at different depths) and the borehole (Repository type 11). The values are estimated for repositories with no backfill and no additional fill in waste units.

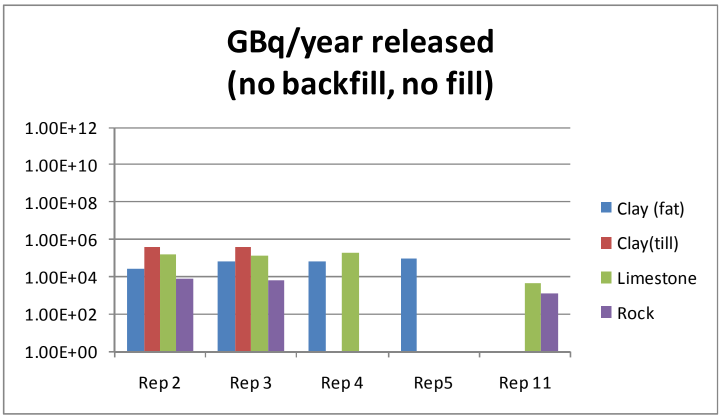

Figure 7.4 *Sum of source terms for near surface and medium deep irreversible repositories and the borehole, Scenario 1A and 1B, see section 7.3.1. Intermediate result, not taking decay into account*

7.3.3 Uncertainty and variability

The value of the source term is subject to a wide range of uncertain parameters and variables. Among those, the hydraulic conductivity of the soil layers around the repository is of particular importance as the value of the source term is proportional to the groundwater flux through the repository and variations in the hydraulic conductivity of the soil around the repository have a direct impact on the amount of nuclides washed out.

Uncertainty simulations were performed using the EXCEL add-in @RISK. Simulations were performed for a medium deep repository located in clay (till). This type of soil was selected because it permits the highest flow of water through the waste units and thus allows the greatest amount of nuclides to be washed out. The repository type was selected because it is representative of most options.

The importance of variations in hydraulic conductivity and the variations due to implementation of backfill have been examined. The variables considered are the water flux, the concentration of representative elements, the time to wash-out elements present in small amounts and the retention time in backfill.

The hydraulic conductivity of the soil around the repository determines the maximum possible groundwater flow rate. The uncertainty of this parameter is very large when nothing is known on the specific conditions at an actual location. The value of this parameter may for the same type of soil be 100 times smaller than the indicated mean or 100 times larger. When the location of the repository is known the hydraulic conductivity may be determined with a smaller uncertainty. To assess the importance of the parameter in such a situation, simulations were performed using a probability function which varies in the interval 10 times less to 10 times larger than the mean.

Flux through the repository	When the hydraulic conductivity is allowed to vary, the ratio between maximum and minimum flux out of the repository will be about 1,000, when fixed the ratio is about 2.
Retention time	The variability in retention time has been simulated. For Ba-133 as an example, the mean retention time is about 1800 years for bentonite backfill while only about 300 years, when concrete is used. In both cases the ratio between maximum and minimum is a factor of 2.
H-3	The ratio between the maximum and minimum values for the source term concentration of H-3 is about 60 when the value of the hydraulic conductivity is allowed to vary and about 30 when fixed.
Ba-133	The ratio between maximum and minimum values estimated for the source term concentration for Ba-133, as an example of the nuclides with medium half life, when the hydraulic conductivity varies, is in the order of 3,000. The ration for a fixed conductivity is in the order of 600.
U-238	If the repository is not backfilled, the variation in the estimated concentration of U-238 will be large. The variation between maximum and minimum estimates is a factor of about 1,000.

The simulations indicate that when the hydraulic conductivity is allowed to vary +/- a factor of 10 and other parameters vary within reasonable limits then the probability of washing out 10 times more of a nuclide per time unit than the estimated mean value varies between 0.01 and 0.001 for the specific nuclides. For example the probability that the value in the source term is more than 10 times the indicated mean value for H-3 is 0.001. For the nuclides Ba-133 the probability is 0.01 and for U-238 the probability is 0.01. Simulated values 100 times the estimated mean value were not observed.

7.4 Groundwater model

The groundwater transport modelling approach and model selection is further described in Appendix E. It is based on a semi-deterministic approach, where several combinations of geology, hydrology, repository location and nuclide retention properties are examined.

The groundwater modelling includes four different geologies found in the Danish underground i.e. limestone, rock, clay till and plastic, fat clay. In principle, the repository can be located close to ground level or above the ground surface, as a semi-deep geological repository at 30 to 100 metres depth or as a borehole repository at depths from 100 to 300 metres.

As the groundwater flow model code, the well-proven MODFLOW model code was chosen (see Appendix F for discussion of its validity). As the reactive solute transport code the MT3D code was selected.

7.4.1 Conceptual models

In Appendix E, the conceptual geological models are described in detail together with the conceptual understanding of the catchment in which the repository is located and the understanding and modelling of the pathways to recipients, including dilution in streams and coastal waters. Furthermore, the parameterisation of the conceptual models is described, including the approach to modelling of retention in combined geologies.

Near-surface repository

A near-surface repository will typically be located on the ground surface or up to 30 m below. The repository may be located in an unsaturated environment, with the possibility that radioactive waste can reach the biosphere and/or atmosphere directly, in the form of gases or dust. It may also be located below groundwater level, but in that case the modes of transportation are the same as the ones for the semi deep repository and will be addressed in the next section. Geochemical conditions may be different though, because of e.g. oxygen content. Other potential transport paths that have been dealt with are shown in Figure 7.5 and include:

- Transport through the unsaturated zone to the groundwater zone.
- Transport through the unsaturated zone via root uptake (evapotranspiration) to plants, trees, etc.
- Transport through the unsaturated zone to the atmosphere via evaporation
- Transport through the groundwater zone to:
  - a shallow groundwater abstraction well
  - a deep groundwater abstraction well
  - drainage pipes (and thereby to a surface water recipient such as a wetland or a stream)
  - a surface water recipient such as a wetland or a stream
  - the sea.

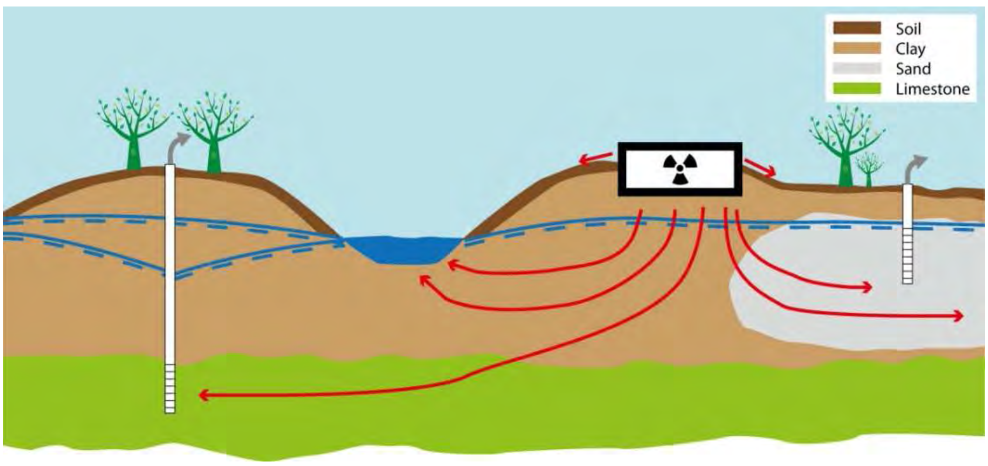

Figure 7.5 Conceptual model of the location of the repository close to ground level

Semi deep repository

The semi deep repository is assumed to be located at depths from 30 - 100 metres below ground surface. A semi deep repository will obviously be located in a saturated environment. Potential transport paths that have been dealt with are shown in Figure 7.6 and include:

Transport through the groundwater zone to:

- a shallow groundwater abstraction well
- a deep groundwater abstraction well
- a surface water recipient such as a wetland or a stream
- the sea.

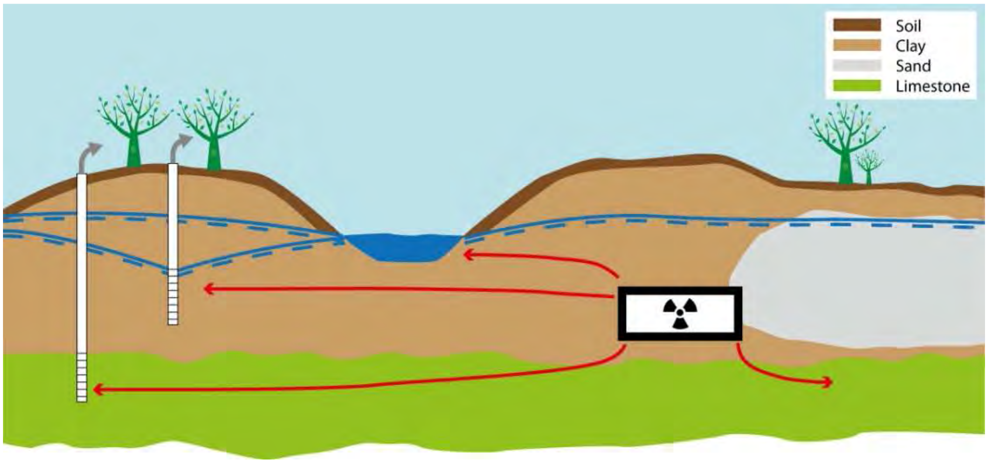

Figure 7.6 Conceptual model of the location of the semi deep repository

Borehole

The borehole disposal facility is assumed to be located at depths from 100 to 300 metres below ground surface and will as such obviously be located in a saturated environment. Potential transport paths are similar to the ones described for the semi deep repository. The borehole option is in the groundwater modelling assumed to be 50 m high, even though the technical layout suggests that the storage height is 75 m. This is due to the model layers being set to a thickness of 50 m. However, changes in the storage height will not affect the overall conclusions from the groundwater modelling of the borehole option.

Above surface repository

If the repository is located above surface additional issues will have to be accounted for. Firstly, percolation to the groundwater zone - eventually through an unsaturated zone - may take place if several consecutive incidents occur and secondly, contaminated leachate may flow to the nearby stream through a drainage system.

Leakage to the ground surface may occur if 1) infiltration occurs through the topsoil and membrane and 2) all drainage systems inside the repository malfunction. In the worst case situation the infiltration will be equal to the net precipitation (400 mm/year) times the area of the repository (4,700 m^2) which is equal to 1,880 m^3/year. This amount will eventually infiltrate to the subsurface or discharge to the stream via the drainage system - or a combination of the two situations.

Leakage to the subsurface may occur if 1) infiltration occurs through the topsoil and membrane and 2) all drainage systems inside the repository malfunction and 3) there is a hole in the membrane below the repository. In this case a water table will built up inside the repository and infiltration to the subsurface system will depend on the water table and the leakage through the clay membrane below the repository. The clay membrane is constructed to have a hydraulic conductivity of 1 x 10^{-9} m/s but may - in the worst case - be as high as 1 x 10^{-7} m/s.

In the case that the repository is located on clay till or limestone all leachate will infiltrate to the groundwater zone because the infiltration capacity is high.

In the case that the repository is located on fat clay deposits a part of the leachate will infiltrate while another part will flow to the river via drainage and/or surface run off. It is estimated that the infiltration is 150 mm/year and flow to the river is 250 mm/year corresponding to 750 m^3/year and 1,200 m^3/year, respectively.

The leachate infiltrating to the groundwater system will eventually flow through an unsaturated zone where different adsorption and/or degradation processes will take place. However, it is conservative to disregard this and use the discharge as the source term for transport modelling in the geosphere model.

If the repository is located on top of rock all leachate will flow to the surface water system, i.e. the stream.

A typically used stream discharge for dilution in such cases is the median minimum discharge. There is a large variation in median minimum discharges in streams in Denmark. The specific discharge varies from 0.6 l/s/km^2 as a mean for Zealand to 6 l/s/km^2 for Jutland and within these regions the variation is considerable (Ovesen et al., 2000). As a conservative assumption a value of 0.6 l/s/km^2 has been applied and with an assumed catchment area of 25 km^2 the discharge is in the stream close to the repository assumed to be 15 l/s.

Horizontal setup

The horizontal geometrical setup of the groundwater transport model is suggested to be as shown in Figure 7.7. It is a relatively simple setup, primarily having flow from upstream of the repository towards the sea that includes the repository (at various depths), a wetland, two groundwater extraction wells, a river, and the sea as the final recipient. It should be emphasized that the result of the safety assessment is to investigate the overall safety level of the disposal concepts and a ranking of the various scenarios (geology, hydrology, repository depth).

The deliberations behind the horizontal setup have included:

- The repository will be located at some distance from groundwater wells and not directly upstream such wells. It is described in Danish Decommissioning (2009) that the repository will not be located in OSD (areas with special drinking water interests), so the wells are located such that abstracted water is expected to be affected only to some degree.

- Attention is also expected to be given to potential impacts on surface waters like rivers/streams and wetlands, when deciding the location of the repository. However, in Denmark it is difficult to find a location, where a river/stream is further away than one kilometre. This is the reason for having the river located one kilometre from the repository. Similarly, it has been evaluated that a wetland could also be present in the downstream area of the repository.

- A location relatively close to the coast is also likely to occur in Denmark, and the proposed location allows inclusion of coastal issues in the safety assessment.

- The groundwater gradient, the water level in the river, and the level of the wetland are determined based on typical Danish conditions. The mean groundwater gradient is assumed to be 5 ‰, which is in the high range, but may appear close to the coast, and is a conservative estimate.

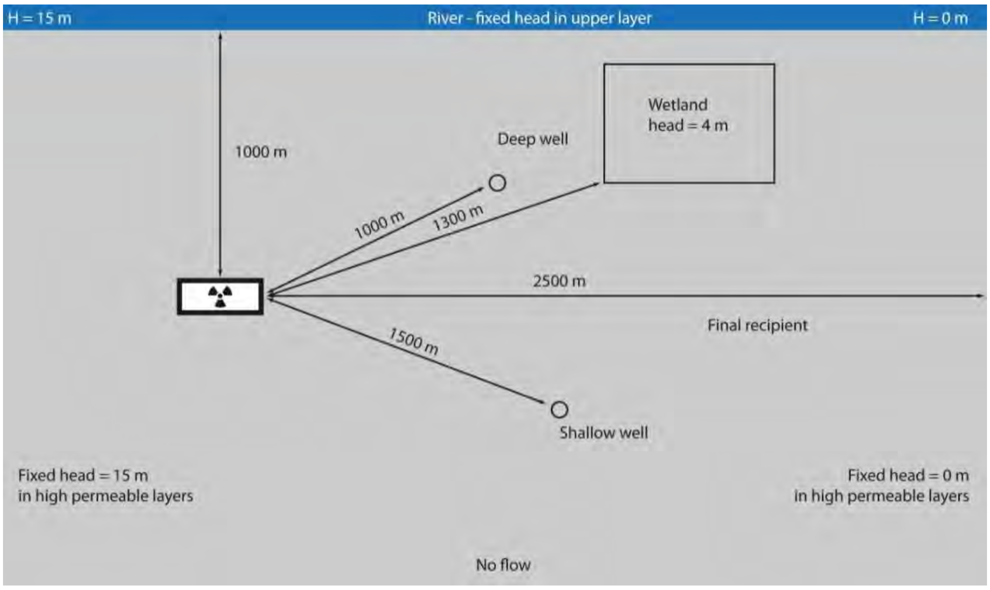

Figure 7.7 Conceptual model - the horizontal geometry of the situation is defined by the location of the repository, wells, a river, a wetland and boundary conditions

The result of the model calculations is a set of results for each combination of repository type, geology, retention class[37] and physical setting (primarily depth) showing resulting concentrations at each of the following outlet points from the groundwater zone:

- a deep well
- a shallow well
- in a stream
- in the sea water in the coastal zone (approx. 50 m from the coast)
- on the beach (due to seepage from a shallow aquifer).

[37] that is, a specified set of K_D values representative for the geology and each nuclide.

The calculated concentrations are based on a unified input concentration to the groundwater model and can thus be seen as concentration conversion or dilution factors. In some of the geological situations either the shallow well and/or the deep well is not present simply because the particular layer, from which the well is pumping, is not present in that situation. The same argument applies to outlets to the sea and on the beach.

7.4.2 Semi-deterministic groundwater modelling approach

Since a hypothetical location of the repository is evaluated, it is not given how the geological formations are built into a deterministic model, i.e. how thick is the formation in which the repository is located, and which layers are below and above this formation. Based on common Danish geological environments, a number of geological scenarios have been defined that include the above described conceptual situations. Figure 7.8 shows the resulting deterministic understanding of the geological environments surrounding the repository.

Clay till situation

In the clay till situation, the top layer consists of 50 metre thick quaternary sediments including 20 m clay till on top, 10 m sand (shallow aquifer) and 20 m clay till again. This series is underlain by 20 m Danien limestone above a thick package of Maastrichtien limestone. The repository may be located in the upper or lower till formations to represent the "repository on or close to ground level" situations and the "semi deep repository" situation, respectively.

Clay, plastic, fat situation

The fat clay may be located very close to the ground surface, but will typically be overlain by quaternary sediments. It is assumed that the fat clay layer is 50 m thick, and the nuclear waste repository may be located at various depths down to 100 m.

Limestone situation

On top of the limestone layer are typically clay till deposits that cover from a few metres up to 60 metres or even more. The limestone layer extends to large depths.

Rock situation

A rock formation typically extends to very large depths i.e. eventually more than 30 km. In some areas the rock formation is overlain by shallow sedimentary rock sediments. Various types of fractures may be present in rock formation, especially in the upper weathered parts as indicated in Figure 7.8. but in the modelling this is disregarded, since it is related to specific local conditions that are not known at this stage.

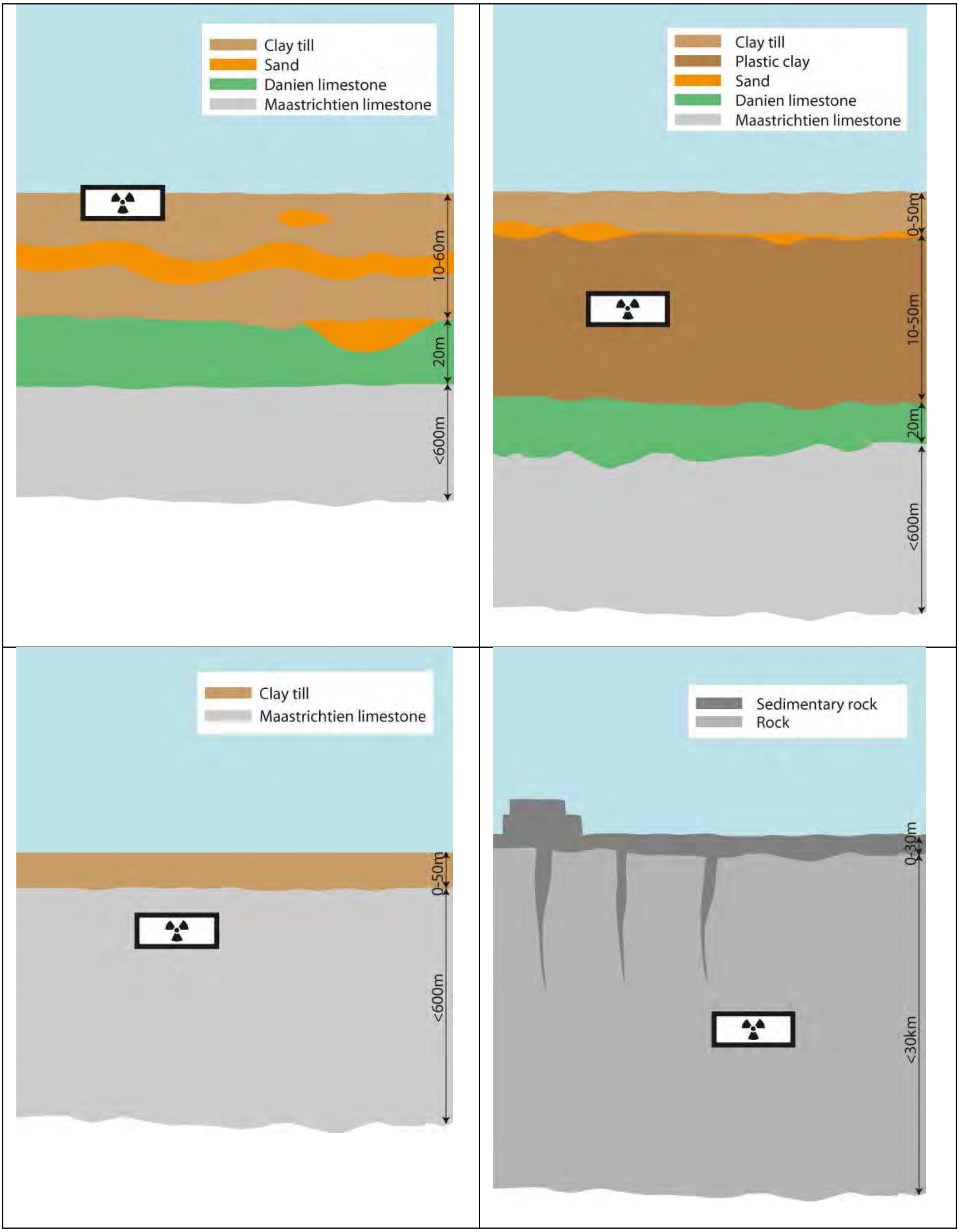

Figure 7.8 *Different deterministic understandings of the geology surrounding the repository. Upper left: clay till. Upper right: fat, plastic clay. Lower left: limestone. Lower right: rock.*

7.4.3 Groundwater models

In order to set up groundwater models based on the above assumptions, the upper hundred metre have been divided into ten metre computational layers and the below two hundred metre (from one hundred to three hundred metres depth) into fifty metre computational layers. In this way, the geological variability can be described sufficiently, and the repository location and size can be reasonably represented. In the model, the geological setting and the location of the main repository and the borehole can be varied.

In order to represent various geologies for the different repository concepts, a number of combinations have been selected that covers the variability in geology. For the fat clay and clay till situations, 6 and 4 different models respectively have been identified based on near surface or medium deep locations of the repository. For the limestone and rock situations, 6 and 5 different models respectively have been identified based on all three locations of the repository. As such, 21 different groundwater models are developed as shown in Figure 7.9. The models are named according to the "column" in which they are placed in this figure for each geology, and according to their depth. The near surface repository located in fat clay is thus named L1M1, while the medium deep repositories in limestone are named SK2M5, SK4M7 and SK7M7 respectively. Repositories located in clay (till) are named starting with ML, while repositories located in rock are named starting with G.

	Depth	Clay, till				
Near surface	0 - 10	ML; NS	ML	S	ML; NS	ML
	10 - 20	Bz	ML	ML	S	ML
	20 - 30	Bz	Bz	ML	ML; NS	S
	30 - 40	SK	Bz	Bz	ML	ML
Medium deep	40 - 50	SK	SK	Bz	Bz	ML; MD
	50 - 60	SK	SK	SK	Bz	Bz
	60 - 70	SK	SK	SK	SK	Bz
	70 - 80	SK	SK	SK	SK	SK
	80 - 90	SK	SK	SK	SK	SK
	90 -100	SK	SK	SK	SK	SK
Borehole	100 - 150	SK	SK	SK	SK	SK
	150 - 200	SK	SK	SK	SK	SK
	200 - 250	SK	SK	SK	SK	SK
	250 - 300	SK	SK	SK	SK	SK

	Depth	Clay, fat				
Near surface	0 - 10	ML	ML	S	ML	ML
	10 - 20	L; NS	ML	ML	S	ML
	20 - 30	L	L	ML	ML	S
	30 - 40	L; NS	L	L	ML	ML
Medium deep	40 - 50	L	L	L	L	ML
	50 - 60	L; MD	L	L	L	L; MD
	60 - 70	Bz	L	L	L	L
	70 - 80	Bz	Bz	L	L	L; MD
	80 - 90	SK	Bz	Bz	L	L
	90 -100	SK	SK	Bz	Bz	L; MD
Borehole	100 - 150	SK	SK	SK	Bz	Bz
	150 - 200	SK	SK	SK	SK	Bz
	200 - 250	SK	SK	SK	SK	SK
	250 - 300	SK	SK	SK	SK	SK

	Depth	Limestone							
Near surface	0 - 10	SK; NS	Bz	Bz	ML	ML	S	ML	ML
	10 - 20	SK	SK	Bz	Bz	ML	ML	S	ML
	20 - 30	SK	SK	SK	Bz	Bz	ML	ML	S
	30 - 40	SK	SK	SK	SK	Bz	Bz	ML	ML
Medium deep	40 - 50	SK	SK; MD	SK	SK	SK	Bz	Bz	ML
	50 - 60	SK	SK	SK	SK	SK	SK	Bz	Bz
	60 - 70	SK	SK	SK	SK; MD	SK	SK	SK; MD	Bz
	70 - 80	SK	SK	SK	SK	SK	SK	SK	SK
	80 - 90	SK	SK	SK	SK	SK	SK	SK	SK
	90 -100	SK	SK	SK	SK	SK	SK	SK	SK
Borehole	100 - 150	SK	SK	SK	SK	SK	SK	SK	SK; B
	150 - 200	SK	SK	SK	SK	SK	SK	SK	SK
	200 - 250	SK	SK	SK	SK	SK	SK	SK	SK
	250 - 300	SK	SK	SK	SK	SK	SK	SK	SK; B

	Depth	Rock			
Near surface	0 - 10	Granite; NS	ML	ML	S
	10 - 20	Granite	Granite	S	ML
	20 - 30	Granite	Granite	Granite	S
	30 - 40	Granite	Granite	Granite	Granite; MD
Medium deep	40 - 50	Granite	Granite	Granite	Granite
	50 - 60	Granite	Granite	Granite; MD	Granite
	60 - 70	Granite	Granite	Granite	Granite
	70 - 80	Granite	Granite	Granite	Granite
	80 - 90	Granite	Granite	Granite	Granite
	90 -100	Granite	Granite	Granite	Granite
Borehole	100 - 150	Granite	Granite	Granite	Granite; B
	150 - 200	Granite	Granite	Granite	Granite
	200 - 250	Granite	Granite	Granite	Granite
	250 - 300	Granite	Granite	Granite	Granite; B

Figure 7.9 The identified locations of the repository are marked with yellow. L = fat clay, ML = clay till, S = sand, BZ is Danien limestone, SK is Maastrichtien limestone and Granite = rock

A number of assumptions apply for the semi-deterministic groundwater modelling approach related to boundary conditions, parameters for the various formations and other geological deposits. These assumptions are discussed and described in Appendix E.

As described above, the results of the calculations in terms of concentrations are extracted at predefined locations in the groundwater model, i.e. at a deep well, a shallow well, in a stream, in the sea water in the coastal zone (approx. 50 m from the coast) and on the beach (due to seepage from a shallow aquifer).

Solute transport parameters

A number of assumptions also apply for the solute transport modelling, and they are also discussed in Appendix E. In short, the modelling takes advection, dispersion, diffusion and retardation into account. The parameters used to model these processes are derived from literature reviews combined with COWIs model experience from similar geological and geo-chemical conditions.

For the 21 combinations of geology and repository location up to 14 K_D combinations have been simulated. In order to determine the maximum dose levels, some of the models simulate 1,000,000 years, some 100,000 years and yet some 10,000 years, depending on the half-life time and retardation of the relevant nuclides.

7.4.4 Model results

As indicated above, a large number of models have been executed to simulate the dilution at various recipients. It is obviously not possible and relevant to present all results of these simulations, also because they only represent a part of the final dose calculations. Some relevant results are presented below and in Appendix E in order to illustrate the differences and similarities of the model calculations.

The medium deep repository location is compared for all four geological formations, see Figure 7.10 to Figure 7.13. The repository depths are 40 - 50 metres, 50 - 60 metres, 50 - 60 metres, and 60 - 70 metres below ground surface, respectively. The K_D used is 2 m^3/kg corresponding to K_D group 3/3 and all models have run for 1,000,000 years. Figure 7.10 to Figure 7.13 show the concentration levels at the recipients based on a unified input concentration to the groundwater model corresponding to 10 kg/year which again corresponds to 1 kg/l.

From the figure it can be seen:

- There are considerable differences between the break through times and the concentration levels. The lowest concentrations and latest break through appears in the rock formation, while earliest break through and highest concentration levels appears in clay till formations.

- Break through is a little later in fat clay than in limestone formations and concentration levels are smaller.

- Steady state with respect to concentration levels is only reached for the clay till location of the repository within the 1,000,000 years simulation time.

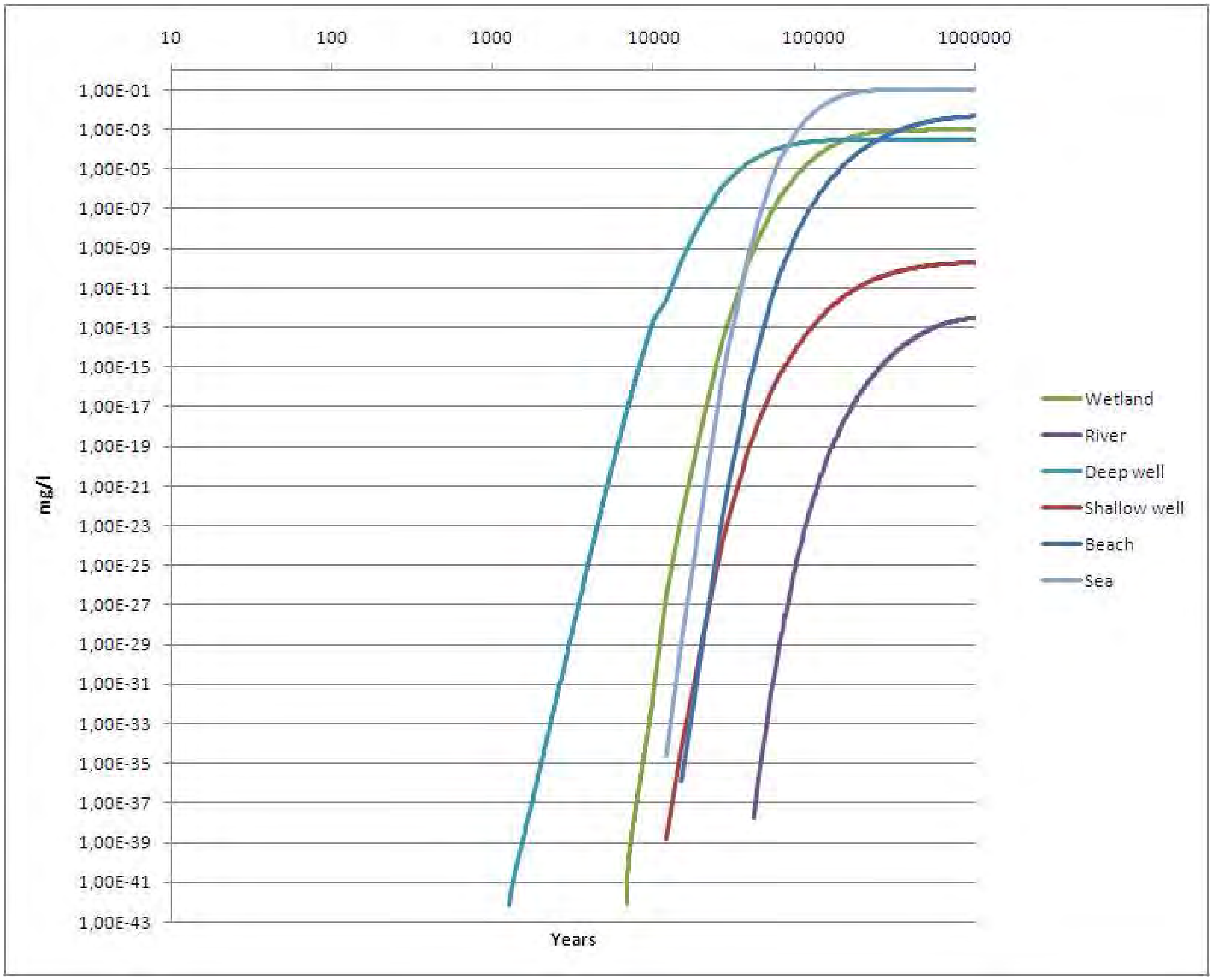

Figure 7.10 Break through curves of concentration versus time for medium deep location of the repository in clay till. ML5M5 (Repository located 50 m below the surface with a sand layer at 20 to 30 metres depth, see Figure 7.9).

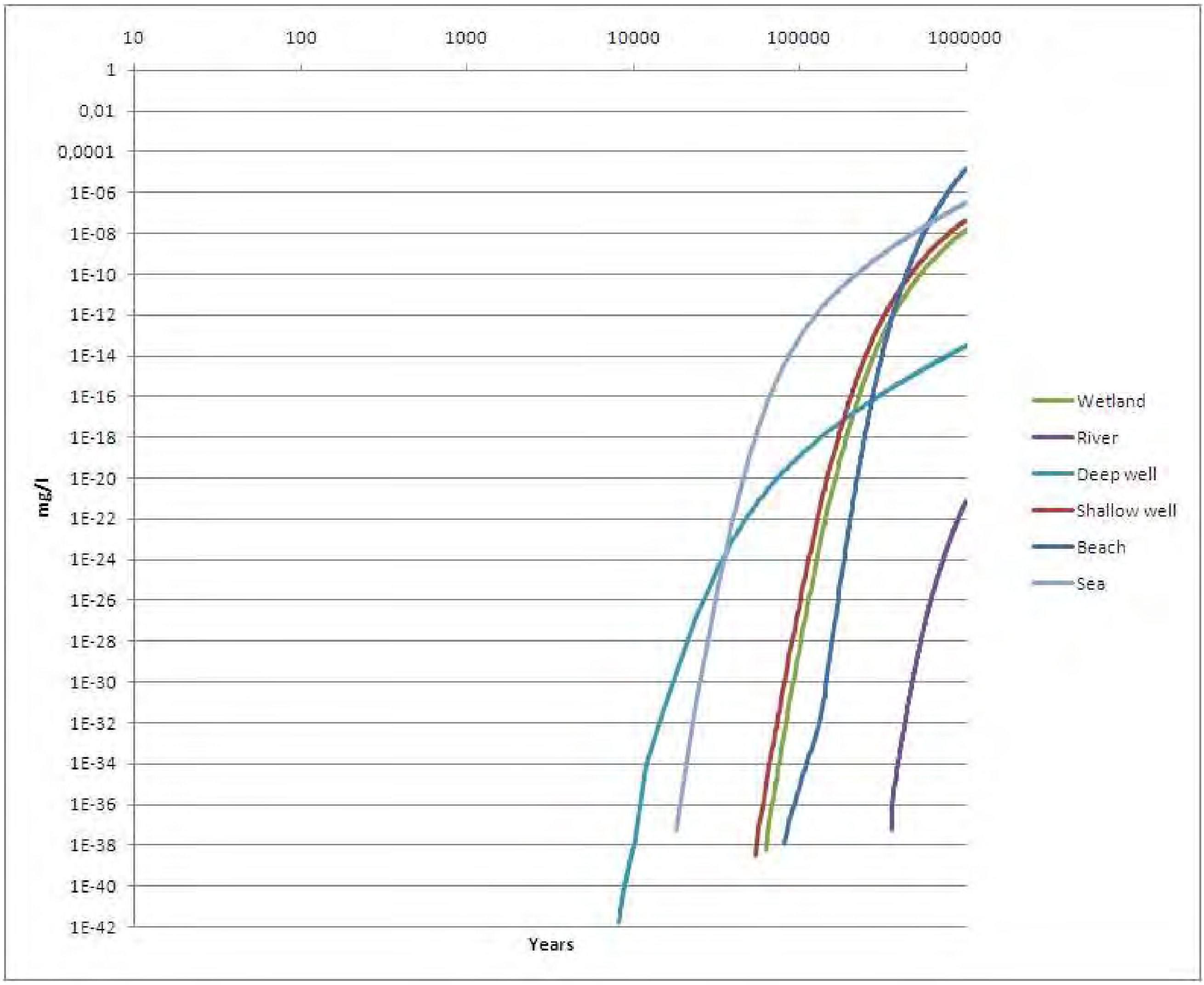

Figure 7.11 *Break through curves of concentration versus time for medium deep location of the repository in fat clay. L5M6 (Repository located 60 m below the surface with a sand layer at 20 to 30 metres depth, see Figure 7.9).*

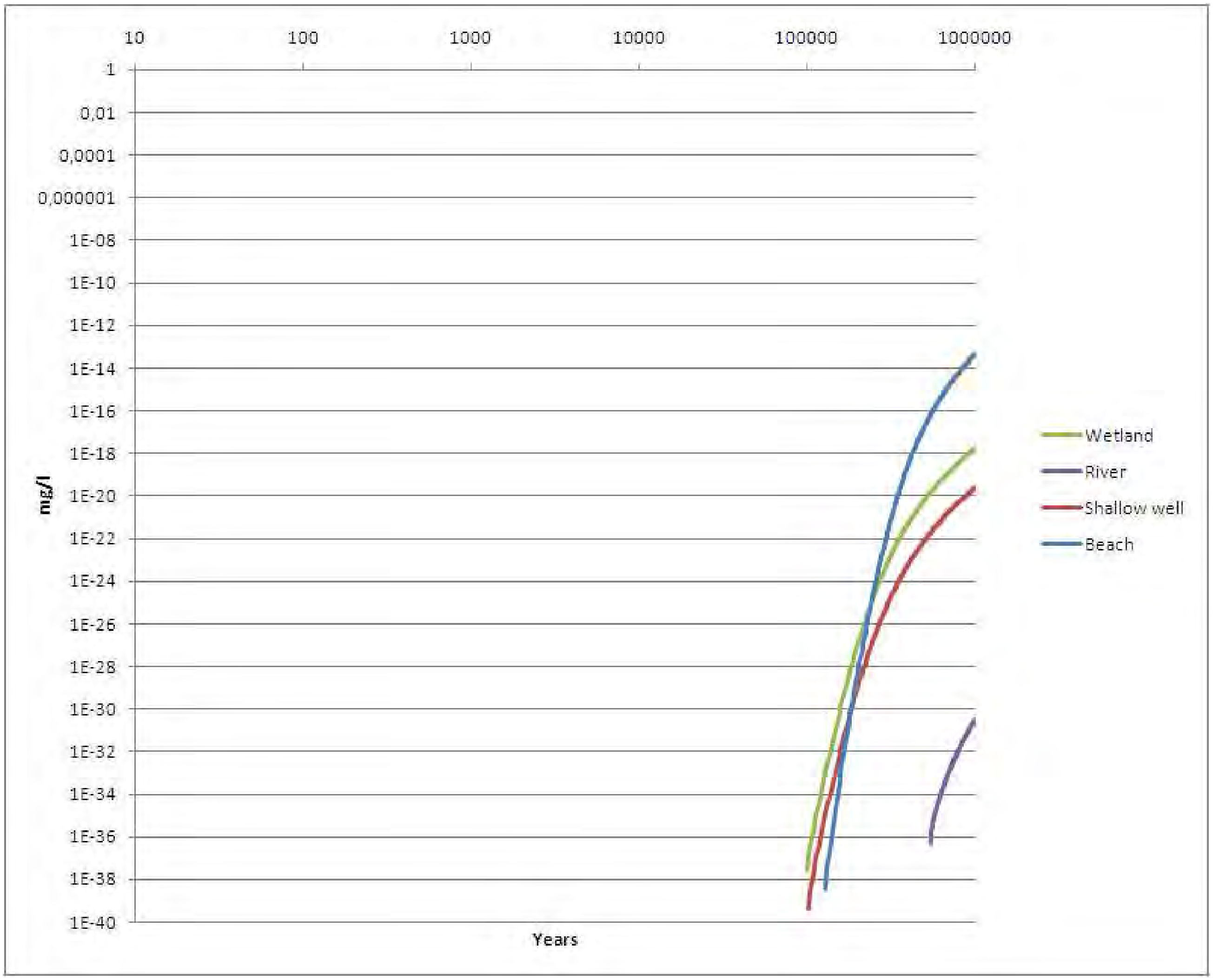

Figure 7.12 Break through curves of concentration versus time for medium deep location of the repository in rock. G3M6 (Repository located 60 m below the surface with a sand layer at 0 to 10 metres and 20 to 30 metres depth, see Figure 7.9).

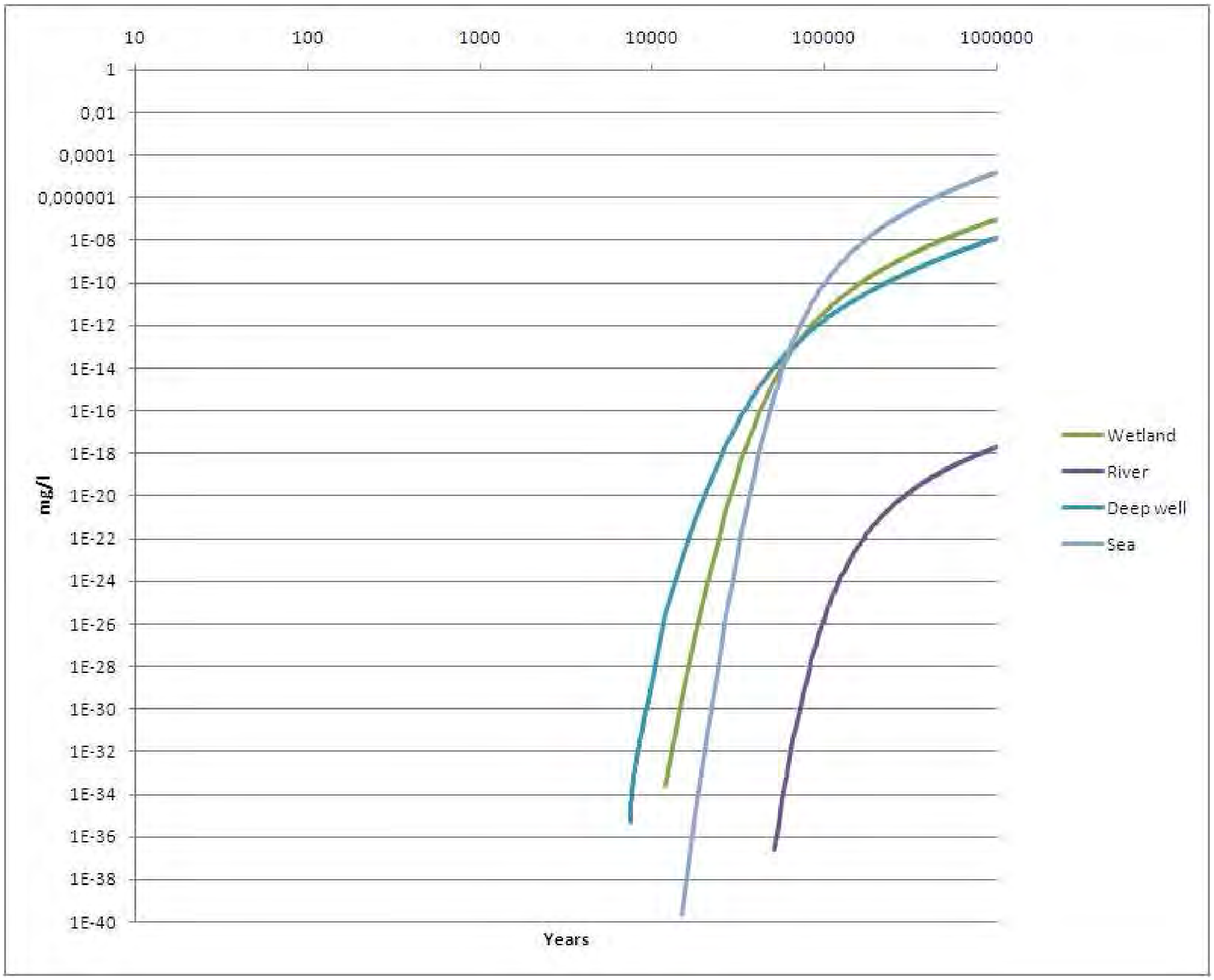

Figure 7.13 Break through curves of concentration versus time for medium deep location of the repository in Maastrichtien limestone. SK4M7 (Repository located 70 m below the surface with Danien limestone at 20 to 40 metres depth and clay till at 0 to 10 metres depth, see Figure 7.9).

Based on the above break through curves and the results presented in Appendix E, it may be concluded:

- There is a considerable variation between the different geologies in relation to break through time and concentrations at steady state; this point to the importance of using local conditions in a final safety assessment.

- There is also considerable variation between the different repository depths (near surface, medium deep and borehole) in relation to break through time and concentration levels at steady state.

- The conditions in geological formations surrounding the layer in which the repository is located are very important and will have a large impact on the spreading of nuclides etc. from the repository. E.g. even though the repository is located in a fat clay environment, a location close to a high permeable Danien limestone layer will eventually result in fast break through in a drinking water well downstream the repository.

7.5 Biosphere model

When performing a dose assessment during a prefeasibility study, where neither location nor repository concept are chosen, and where the initial assessment of the waste is also fairly basic, a number of assumptions need to be made about living habits and exploitation of the landscape by representative individuals inhabiting the site. To secure a conservative/cautious approach, it is assumed that the reference person exploits the contaminated landscape maximally, thus consuming all potentially edible food produced (see for instance SKB, 2006), see description of all assumptions in 7.1.3. The representative person is also assumed impacted by inhalation of dust and gas and by external exposure from the repository. This is a highly hypothetical situation, but since the context is a choice between repository concepts and geologies, it will in principle be the same for all scenarios, and thus comparison will be possible. In future safety assessments of specific locations, etc., the importance of these assumptions should be evaluated.

Exposure can be internal, i.e., through ingestion or inhalation of radionuclides, or external, i.e., dose due to staying in a contaminated environment. It should be noted that dose assessments considering normal operation or release from a deep geological repository often conclude that the external dose is insignificant in comparison with the internal exposure.

The contribution to the total dose from various exposure pathways depends on production of crops, meat, fish, etc., and normal nutritional demands of a representative individual living in the site. This dose model uses Danish intake values wherever available and in other cases primarily values from the similar Swedish modelling of exposure (see for instance Bergström et al, 1999).

The dose coefficients of ICRP (1996) for inhalation and ingestion are used, while external dose coefficients arising from use of contaminated water on land and seepage of contaminated water to the shore are evaluated using the methodology described in Bergström, et al (1999).

The biosphere modelling is described in more detail in Appendix F. The modelling is based on an analysis of possible exposure routes, in the long-term leaching case based on impact caused by leaching of contaminated water from the repository with time. The assessment of impacts from gaseous sources such as radon is described in 7.7.

The biosphere model takes it's input concentration from the groundwater model (and the model for release of gaseous compounds). In the groundwater model release concentrations are calculated at the point shown in Table 7.6, where also the associated exposure routes are listed.

Table 7.6 Potential exposure pathways related to the release points from the groundwater model

Deep well	Intake of water by humans
Shallow well	Intake of milk or meat from domestic animals watered from the well
	Intake of crops irrigated with water from the well
	Inhalation of vapour and resuspended particles
	External exposure from irrigated contaminated land
Stream	Ingestion of freshwater biota (fish, crustaceans, etc)
Sea	Ingestion of biota (fish, crayfish, macroalgae, mussels, etc.)
Sea shore	External exposure from seepage of contaminated water in the beach zone

In the biosphere model, exposure is thus calculated based on the concentrations in each of the release points from the groundwater model. In the framework used, dose conversion factors for each nuclide and outlet point are calculated, that is the total dose resulting from a unit release from the outlet point through all relevant exposure routes related to the outlet point.

In the model, it is assumed that the shallow well is used for watering of produce and cattle, that produce is grown above (or quite near) the repository and grazing of cattle also takes place here. The short well thus gives rise to the most complex exposure routes including ingestion of crops, meat and milk, inhalation of resuspended dust contaminated through watering of the soil, and external exposure also caused by watering of the soil.

The deep well is used for drinking water for humans.

Fish etc. is caught and consumed from both the stream and the sea. Half of the fish consumed is taken from the stream, while the other half is taken from the coastal zone, which is also the origin of all shellfish consumed.

The beach zone only gives rise to external dose.

Thus, the result of the dose model is a matrix consisting of a dose conversion factor for each nuclide and each release point.

Even though concentrations in the water seeping to the defined wetland is calculated in the groundwater model, it has been decided not to include impacts from the wetland in the dose model, since use of a wetland for production of generally used food items is considered of little relevance. These concentrations could be used for evaluating impact on the environment in general. Concentrations in drains are also not used for evaluating impacts on humans. The wetland concentrations can be included in an evaluation of exposure to ecosystems at a later stage if considered relevant.

Evaluation of possible impacts due to leakage of gaseous nuclides from the repository itself or from a contaminated groundwater plume (e.g. radon) is also evaluated, see Chapter 7.8.

Relevant parts of the dose model, e.g. impact through seepage to surface water, can also be used in the modelling of impact from accidents as described in Chapter 8 of this report.

The dose model is run both deterministically and probabilistically, where distributions for the uncertainty of a large number of the parameters included in the model are used in Monte Carlo simulations to generate aggregated uncertainty distributions for the dose conversion results.

7.5.1 Model results

The result of the biosphere model alone is a set of dose conversion factors for each nuclide set up for each of the possible release points from the groundwater model:

- The deep well
- The shallow well
- The stream
- The sea
- The sea shore.

In these dose conversion factors also the relevant dose coefficients have been included dependent on the relevant exposure routes giving rise to ingestion, inhalation or external exposure from the ground. These resulting dose conversion factors are listed in Table 7.7

As can be seen from the table there is a fairly large difference between the nuclides. This is partly due to the difference in dose coefficient (which represents the difference in harm caused by the different nuclides once they have entered the body by a certain route) and partly due to their different properties with respect to adsorption to soil, uptake and transfer in plants and uptake in animals. As it can also be seen there is some difference between the different exposure routes represented by the different release points.

Table 7.7 Dose conversion factors for the different release points from the groundwater model to the biosphere model.

	Deep well	Shallow well	Stream	Coast	Beach
	[Sv/year / Bq/m3]	[Sv/year / Bq/m3]	[Sv/year / Bq/m3]	[Sv/year / Bq/m3]	[Sv/year / Bq/m3]
H-3	1.08E-11	7.02E-11	1.80E-13	3.96E-13	0
C-14	3.48E-10	1.69E-09	2.90E-08	3.36E-08	0
Ca-41	1.14E-10	4.29E-10	2.28E-11	5.70E-11	2.49E-11
Co-60	2.04E-09	3.90E-09	1.02E-08	2.18E-08	2.75E-07
Ni-63	9.00E-11	4.78E-10	1.50E-10	9.30E-10	0
Se-75	1.56E-09	4.43E-09	5.20E-08	2.09E-07	0
Sr-90	1.68E-08	4.25E-08	1.68E-08	2.24E-08	0
Tc-99	3.84E-10	5.87E-10	1.28E-11	3.90E-10	0
Ag-108m	1.38E-09	1.91E-09	1.15E-10	2.67E-08	1.75E-08
Ba-133	9.00E-10	1.17E-09	1.50E-10	3.36E-10	6.43E-11
Cs-137	7.80E-09	4.50E-08	1.30E-06	5.46E-08	6.36E-08
Sm-151	5.88E-11	8.76E-11	2.94E-11	2.55E-10	1.75E-07
Eu-152	8.40E-10	1.24E-09	7.00E-10	5.60E-09	3.03E-05
Eu-154	1.20E-09	1.77E-09	1.00E-09	8.00E-09	2.62E-05
Ir-192	8.40E-10	1.12E-09	1.40E-10	5.60E-10	9.56E-09
Pb-210	4.14E-07	5.10E-07	1.73E-07	1.41E-06	9.14E-11
Rn-222	0.00E+00	8.03E-17	0.00E+00	0.00E+00	4.0E-18
Ra-226	1.68E-07	2.32E-07	1.12E-08	3.36E-07	7.57E-11
Ac-227	6.60E-07	7.97E-07	1.10E-06	4.40E-06	9.56E-13
Th-229	2.94E-07	3.59E-07	2.94E-08	5.78E-06	2.58E-09
Th-230	1.26E-07	1.54E-07	1.26E-08	2.48E-06	3.98E-11
Pa-231	4.26E-07	5.20E-07	7.10E-08	2.84E-07	2.27E-09
Th-232	1.38E-07	1.68E-07	1.38E-08	2.71E-06	2.03E-11
U-233	2.94E-08	3.83E-08	4.90E-09	5.88E-08	7.57E-12
U-234	2.94E-08	3.83E-08	4.90E-09	5.88E-08	4.78E-12
U-235	2.82E-08	3.68E-08	4.70E-09	5.64E-08	1.35E-09
U-236	2.82E-08	3.68E-08	4.70E-09	5.64E-08	7.22E-12
Np-237	6.60E-08	8.01E-08	5.50E-08	2.20E-06	7.22E-10
U-238	3.06E-08	3.99E-08	5.10E-09	6.12E-08	2.04E-13
Pu-238	1.38E-07	1.58E-07	6.90E-08	1.84E-07	8.51E-11
Pu-239	1.50E-07	1.72E-07	7.50E-08	2.00E-07	2.38E-11
Pu-240	1.50E-07	1.72E-07	7.50E-08	2.00E-07	5.96E-12
Am-241	1.20E-07	1.37E-07	6.00E-08	8.00E-07	2.86E-09
Cm-244	7.20E-08	8.29E-08	3.60E-08	3.60E-07	4.77E-12

7.5.2 Uncertainty and variability

As for the other models both variability of the parameters used in the biosphere model is an issue, e.g. due to different soil types in the top layers. Other parameters such as intake by humans and animals are uncertain. Estimates of the possible minimum and maximum values and the relevant distribution functions have been set, and based on these probability calculations can be carried out using @RISK. This has been done for a number of nuclides representing different nuclide groups with e.g. different half-lives.

The results show that the variability/uncertainty varies between the nuclides an the exposure routes. This is partly due to the data available for the setting of parameter spans. For Cs-137 there is quite a lot of data, this nuclide being often of concern in relation to air borne spreading of radioactive material, e.g. in the context of accidental releases from nuclear facilities. This actually gives a larger variability for Cs-137 than what is observed for several of the other nuclides, where less data is available. In order to take this into account, a variation span of a factor 100 is generally assumed in the discussion of probable variation of the overall impact from the groundwater route.

A number of examples of distributions are given in Appendix F.

7.6 Coupling of the models for the groundwater route

Based on the calculation of overall dose from the different pathways, the different repository concepts and geologies can be compared and correlated with the overall dose constraint set for the repository.

In general, the safety assessment model calculates dose based on concentrations encountered at the point of exposure. This concentration will be a function of release and dispersion in the environment. These calculations can in principle be carried out without taking decay into account, and when the time of exposure is known, then compared to the concentration at that time as a result of decay.

The first coupling of the models is done through coupling of the results from the groundwater model and the dose model. The coupling point is the K_D value appropriate for a specific nuclide in a specific geological setting of the repository. This K_D value will, apart from the nuclide, depend on the soil type and the depth of the placement of the repository[38].

Based on a literature study, a set of appropriate K_D values for each calculated repository scenario is chosen for all the nuclides relevant for the waste types placed in the repository. For all nuclides, the dilution calculated in the groundwater model for the repository is defined for the K_D class relevant for the nuclide. This is done to enable the summation of the doses from each outlet point, since breakthrough and dilution will be the same for the different outlet points.

[38] Which influences e.g. the geochemical conditions and thus the K_D value

For each nuclide the dilution is then multiplied with the dose conversion factor for each of the outlet points to achieve a dose conversion factor per outlet point at a given time.

These results are then combined with the results from the repository model for each variation with respect to fill materials and waste types stored in the repository. And finally the decay of the nuclides is taken into account at the given time, including developing and decay of daughters where relevant.

In this way overall results can be obtained per nuclide for each specific geological setting of a specific repository design and a specific set of included waste types and fill materials[39]. For each combination the total dose from all relevant nuclides can now be estimated, giving a total dose for the scenario, which can, in turn, be used as the basis for the comparison of the scenarios.

A number of the barriers and events to be evaluated primarily influence the time when breakthrough and maximum concentrations occur. This is for instance the case with the retention capacity of the different fill materials, and when evaluating the consequence of premature breaching of barriers, such as liners, steel and concrete packaging and construction elements. This is taken into account in the repository model before the model results are coupled.

7.7 Exposure from air borne nuclides

The repository will be designed to contain radionuclides, using several independent engineered barriers dependent on the repository type. Air borne releases from the repository can thus be due to possible accidents etc. and, with time, to degradation of waste packages and the engineered barriers relevant for each repository type. Accidental releases of air borne nuclides are described in Chapter 8.

A long term impact can be the result of release of gaseous compounds to the atmosphere through the top of the near surface repository types or through vaporisation of nuclides from a plume leaking from the repository long-term.

7.7.1 Transport and diffusion

Nuclides released to air (or water with subsequent release to soil air) will be transported and diluted before reaching the general public. This is for long term impact be modelled by

- modelling of continuous flux of gas from the repository to the atmosphere and the subsequent spreading in the atmosphere
- modelling of release of vapours from a contaminated groundwater plume, if relevant.

[39] both in the drums and containers and in the repository as such

7.7.2 External exposure

External exposure is a consequence of radiation emitted by radionuclides in the surrounding environmental media, here air. Exposure to radionuclides in the air and to dust settled on the ground, will be lower indoors due to the shielding effects of the building. An average dose is calculated assuming the reference person being inside 80 % of the time and outside 20 % of the time, including an average shielding factor of 0.35[40]:

$$\bar{D} \cong 0.8 \cdot 0.35 \cdot D_{outdoors} + 0.2 \cdot D_{outdoors} \cong 0.5 \cdot D_{outdoors}$$

7.7.3 Internal exposure

Internal exposure is due to intake of radionuclides into the human body. Inhalation of contaminated air can occur both indoors and outdoors, although indoor exposure for some nuclides can be lower due to the filtering effects of the building. Radionuclide contamination of air can come both from resuspension of soil particles and the release of nuclides bound to particles or aerosols and from elements that exist in gaseous form (e.g. radon and ^{14}C as CO_2, if present). Resuspension of dust is considered in the biosphere model for all nuclides and is thus not included in the separate modelling of impacts from air borne nuclides.

7.7.4 Exposure pathways

Exposure pathways related to air borne nuclides will for the long term impact primarily include:

- Inhalation of gases.
- External exposure[41]

Doses will in the comparison in the pre-feasibility study be calculated for adults only. In further assessments, assessments should be made for all the age groups defined by ICRP.

The relevant nuclides for assessment of long term impact are primarily ^{222}Rn, which is produced through decay of radium present in the waste and potentially in a leakage plume from the repository in the long term. ^{14}C is also present in the waste and can potentially be leaked to the surrounding geosphere. If the ^{14}C is present in degradable material under aerobic conditions, CO_2 containing ^{14}C can be produced. Some of the CO_2 will be dissolved in the groundwater plume, but some of it can in principle be released to the soil air.

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[40] (Bergström, et al, 2001)

[41] External exposure can include exposure from ground deposition or submersion in contaminated air. Since the expected scenarios only include potential emission of radon and $^{14}CO_2$ and the dose coefficients for external exposure for these nuclides equal zero or a very low, external exposure is not relevant in these cases.

7.7.5 Transport and diffusion models

Transport and diffusion of the released air borne nuclides can either occur in the atmosphere as a result of diffusion of gaseous nuclides (radon and CO_2) through the soil covering the repository, or due to vaporisation of gaseous compounds (radon and CO_2) from a contaminated groundwater plume.

Dispersion of airborne radionuclides

Radionuclides can be released to air through a leak in a deteriorated membrane covering the near surface repositories. The subsequent dispersion to the reference person[42] is modelled by a Gaussian plume model. When screening, the most adverse meteorological conditions is used.

Vapour transport from a plume

Volatile compounds will be released from the groundwater plume dependent on the ratio between the vapour pressure and the solubility of the compound (the Henry constant).

The vapours will diffuse upwards through the vadose zone either to the ground surface, where mixing in the lower atmospheric layers will occur, or into buildings. The effect of the radon will depend on the transport time to a possible target in relation to the decay time for the released radon. The transport time will primarily depend on the depth to the repository and the soil types above the repository. The transport in to a house is due to both diffusive transport through the construction (e.g. concrete floor slab and the base of the exterior walls) and advective transport through cracks in the floor construction and will depend on the construction of the floor and the pressure difference between the soil beneath the house and inside the house (due to temperature differences and ventilation of the house).

As stated previously, only the potential additional dose from radon released form the groundwater plume is evaluated here (specifically with relation to possible differences in repository design and setting), although it is acknowledged that the ground itself may be a substantially larger source of radon impact.

7.7.6 Release models

The vaporisation of the volatile nuclides from the groundwater plume is assumed to take place under equilibrium conditions. This is a worst case assumption regarding the release of radon. The production of radon in the plume is a function of the decay of radium, the concentration of radium and time. The concentration of radium at the point evaluated (1000 m down stream from the repository) will depend on the geological setting of the repository and thus the K_D-value and on the decay rate and time.

Release of radon from the repository itself is relevant for repository models placed above the groundwater. The release rate is a direct function of the combined decay functions for radium and radon. With time the concentration of radon and radium will be in secular equilibrium.

[42] Assumed to be the nearest neighbor situated 1 kilometer downwind.

The release of ^{14}C as a gas (as CO_2) will depend on the degradability of the stored carbon (and on the availability of oxygen and the solubility of CO_2 in the groundwater). It is assumed that the degradation rate for carbon is quite small for the type of waste involved. In the calculations of direct release of CO_2 from the repository a maximum of 10 % of the ^{14}C is assumed converted into CO_2, which is very conservative.

7.7.7 Dose models

In the modelling of impacts it is assumed that the representative person is potentially impacted by air borne nuclides as a neighbour due to a leak in the top membrane or through volatilisation of radon from the groundwater plume passing beneath his house is living 1000 meter downstream from the repository. When considering impact from volatilisation of radon, the time spent inside the house is as stated in 7.7.2.

Outdoor impacts

Airborne radionuclides may cause exposure as:

- direct external exposure from radionuclides in air
- direct external exposure form radionuclides deposited on the ground
- direct internal exposure due to inhalation.

Indoor impacts

Airborne radionuclides may cause exposure as:

- direct internal exposure due to inhalation.
- direct external exposure form radionuclides in air
- direct external exposure form radionuclides deposited on the ground

External exposure from air or dust dragged into the house is excluded for simplification reasons in this pre-feasibility study.

Specific dose models

The method for dose calculation follows the principles recommended by IAEA for radiological impact assessment (IAEA, 1994).This gives the following general equations for dose via inhalation and external dose:

Annual inhalation dose:

$$D_{inh} = C_a \cdot IH \cdot H_i \cdot DC_{inh},$$

where

C_a = Concentration of radionuclides in air [Bq/m³]

IH = Inhalation rate [m3/h]

H_i = Exposure time [h/year].

DC_{inh} is the dose coefficient for inhalation [Sv/Bq] according to ICRP.

External annual dose from a plume:

$$D_{ext,cl} = C_a \cdot DC_{ext} \cdot H_i,$$

where

$DC_{ext,cl}$ is the dose coefficient for external dose (cloud shine) [43].

External annual dose from the ground:

$$D_{ext.gr} = q_{dry} \cdot DC_{ext,gr} \cdot H_i,$$

where

$DC_{ext,gr}$ is the dose coefficient for external dose from the ground[44].

Detailed descriptions of the models used for dispersion etc. are given in Appendix G.

7.7.8 Model results

Long term radon impact from direct release

For the possible long term of release of radon from the above surface and near surface repositories (in this context assumed to be placed a few meters below the surface) calculations have been made of the possible release of radon (due to decay of radium), and of C-14 as CO_2 due to aerobic degradation of 10 % of the radioactive carbon in the waste. This has been done for the year 800, where the HDPE membrane in this context has been assumed to be totally degraded. The releases of nuclides are calculated to:

Above surface repository:

Rn-222: $6.8 \cdot 10^{-6}$ GBq/s   C-14 (as CO_2): $3.7 \cdot 10^{-60}$ GBq/s

Near surface repository

Rn-222: $2.2 \cdot 10^{-6}$ GBq/s   C-14 (as CO_2): $2.0 \cdot 10^{-77}$ GBq/s

The release of C-14 as CO_2 is so low that no calculation of dose after spreading has been carried out.

For Rn-222 calculations have been made for possible dose for the reference person living 1 km downstream from the repository. The dose will be due to the radon daughters and include external dose from ground- and cloudshine and internal dose from inhalation.

The resulting total doses will be:

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[43] (National Council on Radiation Protection and Measurements, 1996 and Karlsson & Aquilonius, 2001), see Working Report 14, Appendix B for values used

[44] (National Council on Radiation Protection and Measurements, 1996 and Karlsson & Aquilonius, 2001), see Working Report 14, Appendix B for values used

Above surface repository: Rn-222: $3.3 \cdot 10^{-4}$ mSv/year

Near surface repository: Rn-222: $1.1 \cdot 10^{-4}$ mSv/year.

To evaluate if radon could be an issue if emitted from a groundwater plume under the house of the reference person, initial calculations of the resulting radium concentration in the soil near the different repositories were carried out and compared to typical natural concentrations.

The release of radium from the different repositories will result in pore water concentrations near the repository of maximum 0.2 GBq/m^3. This can be compared to measurements from e.g. Swedish well water samples that vary between 0.2 and 2455 GBq/m^2 with a median value of 12 GBq/m^3 (IAEA, 2005a). The pore water concentrations can thus be seen to be quite low compared to natural background. On this basis, it has been decided not to carry out calculations of release of radon from the groundwater plume.

7.8 Overall results

In this chapter the overall results of the modelling encompassing the preliminary safety assessment is presented.

All in all, 7 repository models have been compared, see Table 7.5, in 4 scenarios, see section 7.3.1, where more or less of the waste[45] has been placed in the borehole (repository type 11). This has been done for:

- 4 different geologies for the above surface repository (type 1)
- 7 different geologies for the near surface repository (type 2)
- 9 different geologies for the medium deep repositories (type 3 - 10), where the combinations of the repositories and geologies are dependent on the depth of the repository.
- The borehole (type 11) has been evaluated for 2 geologies and 2 depths for each geology.

7.8.1 Combined results for the groundwater route

Tritium

A specific check of resulting tritium concentrations in the deep and shallow wells has been carried out against the drinking water quality criteria for tritium of 100 Bq/l = 10^{-4} GBq/m^3.

Maximum doses of tritium in the wells are in the repositories placed in clay (till) near permeable Danien limestone or sand layers quite near or equal to the tritium quality criteria, if no fill is used in the repository. The rest of the repositories are from 2 to several decades below the criteria.

[45] Particularly the special waste

Heavy metals

As mentioned previously, the waste also contains: the following major amounts of metals:

- 2 tons of uranium
- 50 to 70 tons of lead
- 200 kg of cadmium
- 80 kg of beryllium.

Spreading of uranium and lead has been calculated as part of the calculation of the radioactive nuclides of these metals[46]. And based on these, possible resulting concentrations of the metals can be estimated in the wells and compared to drinking water quality criteria. This can similarly be done for cadmium and beryllium, when their solubilities are known.

Cadmium is, based on a literature search, assumed to have a solubility of $2 \cdot 10^{-6}$ M, and beryllium a solubility of $1 \cdot 10^{-4}$ M. K_D values for the two metals are set to 0.1 m^3/kg and 1 m^3/kg[47] respectively, also based on literature (Allen et al, 1995; Anderson & Christensen, 1988; ATSDR, 2008; Buchter et al, 1989; Del Debbio, 1991; Garcia-Miragaya, 1980; US Department of Health and Human Services, 2002).

Based on the groundwater modelling carried out for the different K_D groups, it can be seen that lead and beryllium will be retained in the soil matrices for a very long time and that the dilution before reaching the well will be at least 10^{10} times. This means that the resulting concentrations of lead and beryllium will much lower than the 5 µg/litre that is the Danish quality criteria for lead (Ministry of the Environment, 2006). There is no criterion for beryllium in Denmark, but the US EPA has set a drinking water quality criteria of 4 µg/litre (US Department of Health, 2002).

Similar calculations for cadmium and uranium will result in maximum concentrations for the worst case repository of 0.002 µg/litre and 0.000002 µg/litre respectively. The Danish quality criterion for drinking water for cadmium is 2 µg/litre (Ministry of the Environment, 2006). There is no criterion for uranium, but natural concentrations in Sweden in water samples from quaternary deposits lay around 0.001 µg/litre (IAEA, 2005a)..

The metals present in the waste thus do not constitute a risk.

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46 Since the releases has been calculated as mass concentrations and only later been converted to radioactivity.

47 This is a very conservative value for beryllium.

7.8.2 Variability of the results due to parameter variability

As can be seen from the chapters regarding the models for the overall exposure via the groundwater route, there is variability and uncertainty related to the results. For the repository model this is typically in the order of 2 decades, which is also true for the dose model. For the groundwater model the variability can be quite large, when not looking at a specific location, but will probably be within 3 decades at an actual site (which has not been investigated in great detail). On this basis, it is estimated that the actual variability on the results lay within at least 7 decades. It is thus suggested, that the calculated result for the overall dose in the prefeasibility study shall be at least 5 decades lower than the dose constraint of 0.01 mSv per year to ensure that there is not a great probability of exceeding the criteria. This suggestion is also based on the generic nature of the modelling carried out, which on the one hand calls for quite conservative assumptions for the reference person, and on the other hand gives no differentiation with respect to affected age group or the number of persons actually affected.

7.8.3 Overall results for the dose modelling

As described previously, the safety assessment has looked at exposure from two types of near surface repositories:

- One type placed on the surface (ASR or Repository type 1)
- One type placed up to 30 meter under the surface (NSR or Repository type 2)

These two types can be placed on or in either fat clay, clay (till), limestone or rock. For the repositories beneath the surface, a location just beneath the surface and a location 30 meter under the surface have been investigated. Different relevant adjacent layers have been taken into account.

Medium deep repositories (MDR of Repository type 3 to 10 dependent on the design of the repository and the reversibility of the repository[48]) have been assessed for all 4 geologies. The highest repository type (bottom app. 50 meter below surface) has been evaluated for all 4 geologies. For this repository type, both repositories filled from above and from the inside have been assessed.

For the deeper repositories, only operation from above has been included, since too much room will have to be taken up of the overall repository volume to allow for inside access. Also a repository constructed as a cavern in fat clay, limestone or rock has been assessed. The repository located with a bottom depth of app 70 meter has been located in fat clay, limestone and rock. This is based on the likely depths of the different geologies. Also here, probable adjacent permeable layers have been taken into account.

[48] See Table 7.5

The borehole solution has been located in limestone and in rock in depths of 100 to 150 meter and 250 to 300 meter. The results will also cover depths in between.

The assessment model can of course be used to assess other geological locations in the future if necessary.

The assessed geologies are shown in Figure 7.14, which is also shown in the groundwater modeling chapter. The models are named according to the "column" in which they are placed in this figure for each geology, and according to their depth. The near surface repository located in fat clay is thus named L1M1, while the medium deep repositories in limestone are named SK2M5, SK4M7 and SK7M7 respectively. Repositories located in clay (till) are named starting with ML, while repositories located in rock are named starting with G.

All dose results for all repository types with no fill and no backfill are shown in Appendix K, while a few examples of results with and without fill and backfill are given below. The dose curves that are not shown, give no dose in any recipient even after 1,000,000 years.

Some of the dose curves show up as "broken" in the figures around year 100,000, which is because the calculation times for the respective nuclides have not been sufficiently long with small time steps that can be combined directly with the decay calculations. Comparing the residual activity of these nuclides and their dose conversion factors to the nuclides, for which the calculation time has been longer, has given assurance that the dose in later periods will not exceed that of any of the nuclides, for which calculations for a very long time have been carried out.

Dose calculations have only been carried out for fill alone or backfill alone, but extrapolating from the results to combinations of the two is straight forward. It should be noted that the calculations for fill materials does not fully take the absolute retention capacity of the fill material into account. This may make the reduction look to great for the combinations of fill materials and nuclides with high sorption potential. This is another reason for the suggested "safety gap" of 5 decades to the dose constraint. Previous calculations only taking the effect of the fill due to lower hydraulic conductivity and altered geochemistry into account have shown reduction values of the order of 2 to 4 decades dependent on the combination of fill/backfill type, geology and nuclide. The overall recommendations are based on a combination of these results.

It can be seen from the results for repositories without fill and back fill that the dose for some of the nuclides are close to 0.00001 mSv per year, which is above the suggested value taken the uncertainties into account. On the other hand almost all repositories are well below this level, once backfill and fill has been introduced. This is, together with the results from Chapter 8 regarding assessment of potential accidents, the basis for the recommendations given in Chapter 11.

	Depth	Clay, till				
Near surface	0 - 10	ML; NS	ML	S	ML; NS	ML
	10 - 20	Bz	ML	ML	S	ML
	20 - 30	Bz	Bz	ML	ML; NS	S
	30 - 40	SK	Bz	Bz	ML	ML
Medium deep	40 - 50	SK	SK	Bz	Bz	ML; MD
	50 - 60	SK	SK	SK	Bz	Bz
	60 - 70	SK	SK	SK	SK	Bz
	70 - 80	SK	SK	SK	SK	SK
	80 - 90	SK	SK	SK	SK	SK
	90 -100	SK	SK	SK	SK	SK
Borehole	100 - 150	SK	SK	SK	SK	SK
	150 - 200	SK	SK	SK	SK	SK
	200 - 250	SK	SK	SK	SK	SK
	250 - 300	SK	SK	SK	SK	SK

	Depth	Clay, fat				
Near surface	0 - 10	ML	ML	S	ML	ML
	10 - 20	L; NS	ML	ML	S	ML
	20 - 30	L	L	ML	ML	S
	30 - 40	L; NS	L	L	ML	ML
Medium deep	40 - 50	L	L	L	L	ML
	50 - 60	L; MD	L	L	L	L; MD
	60 - 70	Bz	L	L	L	L
	70 - 80	Bz	Bz	L	L	L; MD
	80 - 90	SK	Bz	Bz	L	L
	90 -100	SK	SK	Bz	Bz	L; MD
Borehole	100 - 150	SK	SK	SK	Bz	Bz
	150 - 200	SK	SK	SK	SK	Bz
	200 - 250	SK	SK	SK	SK	SK
	250 - 300	SK	SK	SK	SK	SK

	Depth	Limestone							
Near surface	0 - 10	SK; NS	Bz	Bz	ML	ML	S	ML	ML
	10 - 20	SK	SK	Bz	Bz	ML	ML	S	ML
	20 - 30	SK	SK	SK	Bz	Bz	ML	ML	S
	30 - 40	SK	SK	SK	SK	Bz	Bz	ML	ML
Medium deep	40 - 50	SK	SK; MD	SK	SK	SK	Bz	Bz	ML
	50 - 60	SK	SK	SK	SK	SK	SK	Bz	Bz
	60 - 70	SK	SK	SK	SK; MD	SK	SK	SK; MD	Bz
	70 - 80	SK	SK	SK	SK	SK	SK	SK	SK
	80 - 90	SK	SK	SK	SK	SK	SK	SK	SK
	90 -100	SK	SK	SK	SK	SK	SK	SK	SK
Borehole	100 - 150	SK	SK	SK	SK	SK	SK	SK	SK; B
	150 - 200	SK	SK	SK	SK	SK	SK	SK	SK
	200 - 250	SK	SK	SK	SK	SK	SK	SK	SK
	250 - 300	SK	SK	SK	SK	SK	SK	SK	SK; B

	Depth	Rock			
Near surface	0 - 10	Granite; NS	ML	ML	S
	10 - 20	Granite	Granite	S	ML
	20 - 30	Granite	Granite	Granite	S
	30 - 40	Granite	Granite	Granite	Granite; MD
Medium deep	40 - 50	Granite	Granite	Granite	Granite
	50 - 60	Granite	Granite	Granite; MD	Granite
	60 - 70	Granite	Granite	Granite	Granite
	70 - 80	Granite	Granite	Granite	Granite
	80 - 90	Granite	Granite	Granite	Granite
	90 -100	Granite	Granite	Granite	Granite
Borehole	100 - 150	Granite	Granite	Granite	Granite; B
	150 - 200	Granite	Granite	Granite	Granite
	200 - 250	Granite	Granite	Granite	Granite
	250 - 300	Granite	Granite	Granite	Granite; B

Figure 7.14 Assessed repository and geology combinations.

The dose results for the above surface repositories show very high doses for repositories placed on fat clay and rock. This is due to the impermeability of the soil, causing high concentration water to run of through drainage systems directly to the stream. This is a quite crude representation, which should be nuanced in further modeling efforts.

In general, the dose curves show that location near a permeable layer gives the risk of higher dose and that is more important than depth and geology (apart from location in rock). For rock this is partly due to the assumption that there are no substantial cracks. This assumption is made due to the generic nature of the modeling. The actual direction of the cracks would be very important with respect to the recipients potentially impacted, and should be included in the modeling of a specific rock location.

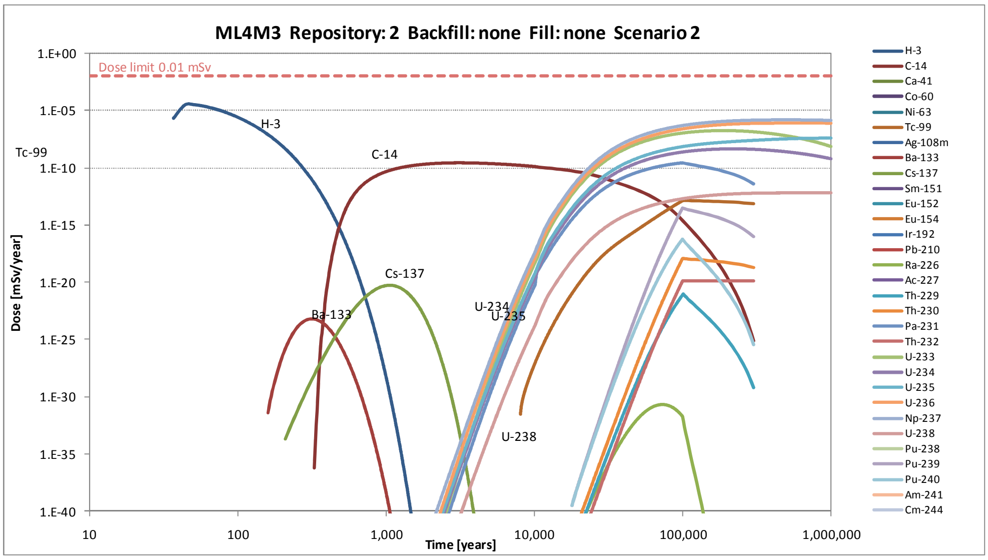

Figure 7.15 Dose results for a near surface repository located in clay (till)

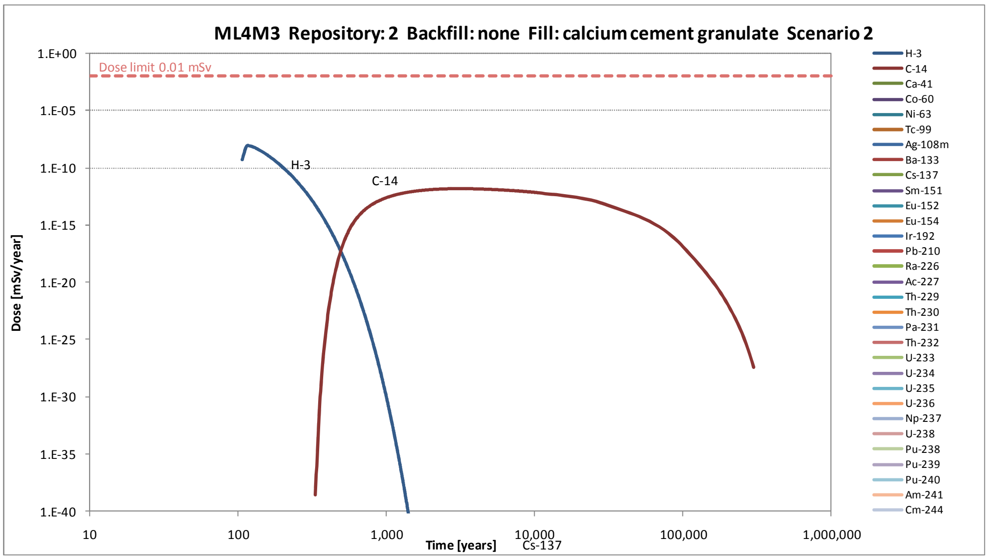

Figure 7.16 Dose results for the less retained nuclides for the same repository as in Figure 7.12 with calcium cement granulate used as fill. Since uranium etc. will first be released after the calculation period for the release of 10,000 years, the effect on the radionuclides cannot be seen fully from the curves.

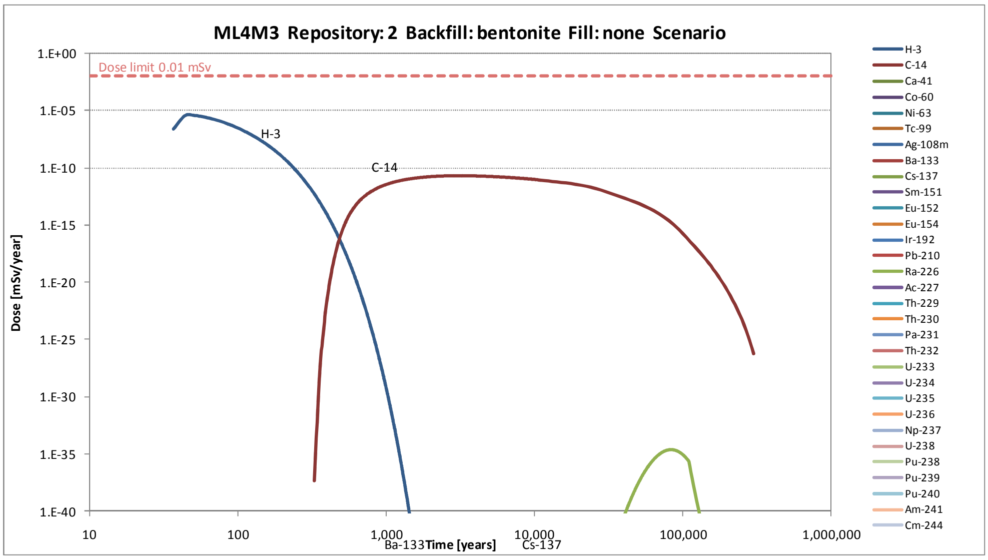

Figure 7.17 Dose results for the less retained nuclides for same repository as in Figure 7.12 with bentonite used as backfill. Since uranium etc. will first be released after the calculation period for the release of 10,000 years, the effect on the radionuclides cannot be seen fully from the curves.

8 Assessment of potential accidents

8.1 Basic assumptions

8.1.1 Repositories

The following types of repository have been considered in relation to possible accidents etc.:

- Above surface repository (ASR) in combination with a borehole (BORE) for the most radioactive waste, and as the only repository without a borehole.
- Near surface repository (NSR) always with combination with BORE for the most radioactive waste. This repository is located at a depth of 0-10 m below ground surface.
- Medium deep repository (MDR) with BORE for the most radioactive waste. There are three types of MDR, all located at a depth of 10-100 m below ground surface. The borehole is located at 100-300 m depth.
- Medium deep repository (MDR) for all waste. There are three types of MDR, all located at a depth of 10-100 m below ground surface.

Repository safety design

Concrete walls of the near surface and medium deep repositories will be designed to prevent cracking caused by earthquakes. It is assumed that they will be designed to withstand earthquakes with a return period of up to 1000 years.

ASR will have a drainage tank of 10 m^3 for possibly contaminated drainage water collecting water that has entered into the repository. The removal of water from the drainage tank will be by pumping, i.e. water cannot by itself run out because of a leaking valve. The drainage tank will be fitted with a level indicator and alarm. A probe will be measuring the radioactivity level inside the tank. The tank is located such that leakages may be visually observed and collected.

ASR will have an electronic leak detection system on the top membrane.

The drainage tank for possibly contaminated drain water is during operation emptied regularly in a safe manner, when there is 1 m^3 in the tank or at least each quarter.

Repository location

It is assumed that repositories will be located where not vulnerable to flooding by sea water at the current sea water level.

It is also assumed that a reasonably high sea water level rise as a result of global warming is taken into account, and that the repository will not be vulnerable to flooding by mean sea level rise of less than 5 m.

Air control / ventilation

Medium deep repositories operated from within the repository (i.e. not from ground level) during the filling of the repository, will have filtering of the air inside the repository before the air is released to the atmosphere. The filter is expected to withhold all particles except Rn. The filters are assumed effective until the repository is sealed.

Operation of repository

The following is assumed concerning the operation of the repository, regardless of type of repository:

- Waste from Danish Decommissioning is filled into the repository during the first year after opening.
- The repository is open for filling in additional waste until 30 years after the opening of the repository.
- The repository will be closed and sealed after 30 years.

8.1.2 Other

Knowledge of the repository

It is assumed that the presence of the repository will be known for the first 300 years. After 300 years the repository will be forgotten.

Dose criteria

A reference dose of 1 mSv per event should be used for accidents considered in the planning and approval of the repository, and it is generally accepted that risk reduction measures are not required. Compared to the reference dose of 10 mSv per accident, below which measures to reduce accident probability or accident consequences are not justified, a reference dose of 1 mSv includes a safety factor of 10 to account for the uncertainties present in these types of analyses.

Persons

The following terms are used in this chapter:

- Worker: Person working at the repository site, if not stated otherwise in the text.

- Neighbour: the reference person living 1 km from the repository.

Major hazards

As stated in the "Beslutningsgrundlag for et dansk slutdepot for lav- og mellemaktivt affald" (Ministeriet for Sundhed og Forebyggelse, 2008), the risks related to large natural hazards, where either the likelihood is extremely low, or where the consequences from the natural hazard is significantly larger than the additional consequences related to the radioactive material, shall not be part of the present analysis. Examples of such are major earthquakes, large meteor strikes, ice age, and volcanic activity.

In the context of the present report earthquakes measuring 7.0 on the Richter scale or above are considered major earthquakes.

Toxicity of chemicals

In relation to accidents, it is evaluated that the consequences related to the toxicity of the waste are inferior to those related to the radioactivity. Hence only the radioactivity is considered in relation to accidents.

8.2 Methodology

The overall methodology of assessment of the accident scenarios is illustrated in Figure 8.1 and is further described below.

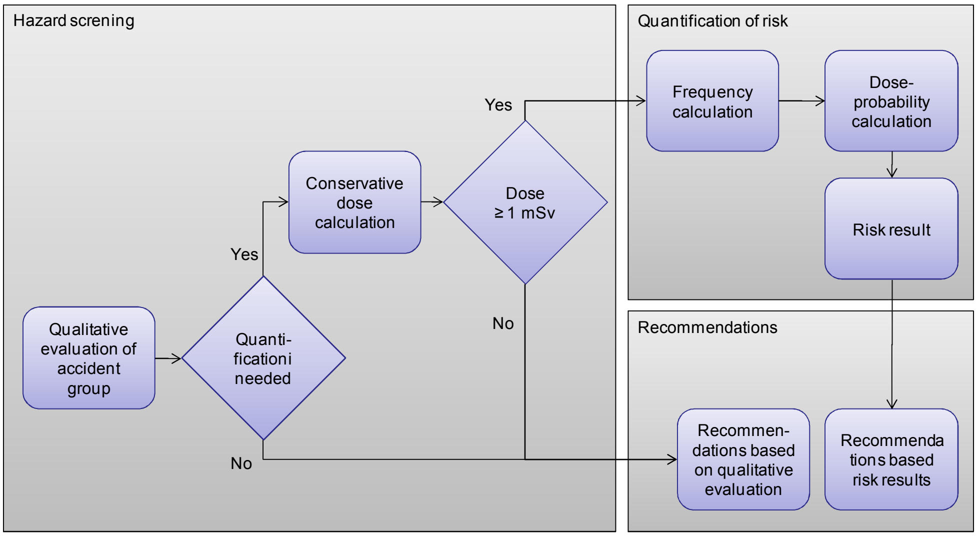

Figure 8.1 Overall methodology of accident modelling and assessment

Based on the hazard identification, a number of hazards have been identified as accidents. These hazards have been grouped into a number of types of accident, e.g. *handling accident* and *fire*. A hazard screening was carried out to identify the more important hazards. This was done by a qualitative analysis, in order to assess if there is a realistic potential for an accident affecting a neighbour followed by a conservative quantitative analysis of the consequences to assess the potential of imposing a dose of 1 mSv to a neighbour. This evaluation was carried out for dispersion calculation using Pasquill weather class F and wind speed 0.5 m/s. If there is not sufficient activity left to create a dose of 1 mSv for a neighbour, or if an unrealistically large release of waste is required for the dose to occur, then no further quantification is done. If on the other hand, a dose above the reference dose may realistically be reached the frequency of the occurrence of the accident is estimated together with the probability distribution of doses.

8.3 Hazard screening

The hazards related to accidents have been sorted into groups according the nature of the initiating event. These groups are listed below, and are discussed in the following sections:

- Handling accidents
- Mechanical damage to repository
- Aircraft crash / meteorite impact
- Deterioration of packaging
- Fire
- Chemical reactions
- Malicious damage and acts of war
- Intrusion by living organisms
- Natural hazards
- Worker safety.

8.3.1 Handling accidents

The main scenarios related to these accidents are:

- Deterioration and disintegration of packaging during handling
- Ineffective radiation shielding due to faulty packaging.

Physical damage to waste unit during handling

This section covers the accidents where part of the waste is being released when handled at the repository site and include the risk related to:

- Opening or physical damage of packaging due to drop from a height
- Puncturing the packaging by a fork lift or lifting spreaders
- Disintegration of packaging during handling due to corrosion or local structural failure.

These handling accidents may lead to release of waste from the waste unit, though only the waste that can be dispersed as dust constitutes a threat of radioactive exposure to the neighbour. Identified waste types that may produce dust are: waste type 1 (graphite), waste type 3 (steel and lead incl. vacuum cleaners), waste type 4 (concrete and heavy concrete incl. concrete dust), waste type 8 (sand and paint dust from sandblasting), waste type 9 (evaporator residues but only those not bituminized), and waste type 10 (mainly contaminated soil).

Handling accident scenarios are presented in form of the generic barrier diagram in Figure 8.2. The barrier diagram is supplemented by a discussion for the individual types of repository.

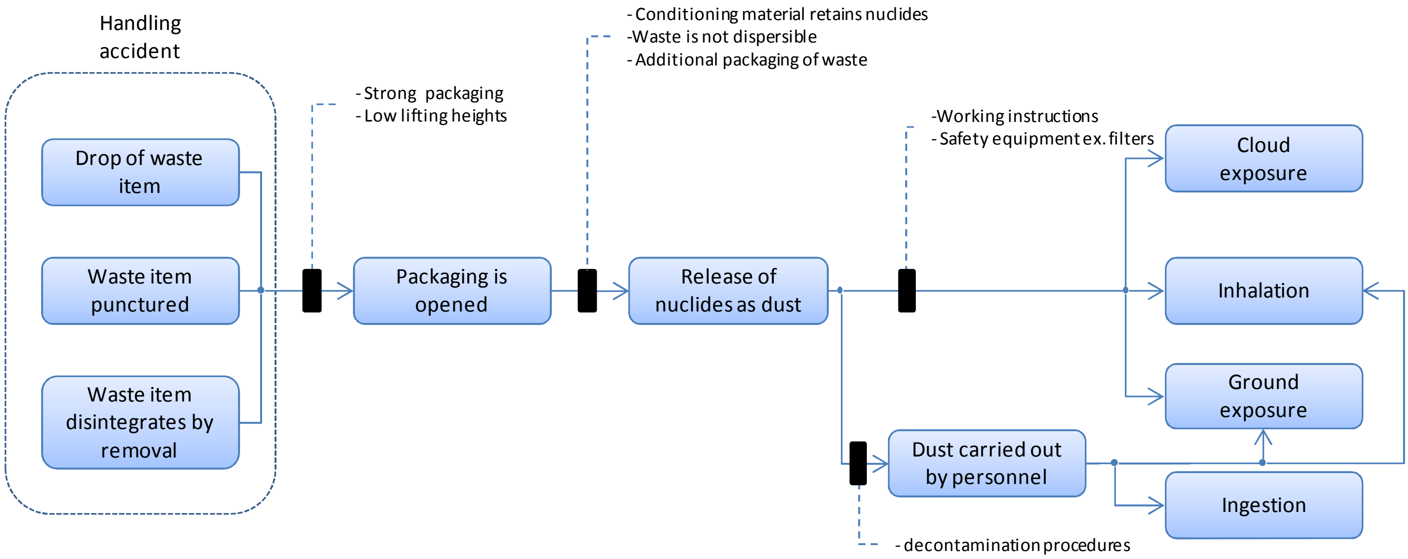

Figure 8.2 Barrier diagram for handling accidents where neighbour is exposed for radioactive waste in form of dust

Phases

The scenarios are relevant for different phases for the different repository types:

- ASR - operations of filling the repository can take place until closure of the repository, removal can take place as long as the existence of repository is not forgotten.
- NSR - operations of filling the repository can take place until closure of the repository, removal can take place as long as the existence of repository is not forgotten.
- MDR - operations of filling repository can take place until closure of the repository. Removal of the waste can take place as long as the existence of repository is not forgotten if the repository is of the reversible type. A reversible repository is designed taking into account the maximal stacking load that can be absorbed by the containers and drums.
- BORE - operations of filling repository and removal is possible until the closure of the repository. Removal of waste from the borehole is not considered.

Above surface repository (ASR)

Filling the ASR will take place by means of fork lifts or cranes. The containers/drums are assumed to be stacked: max 4 ISO containers stack or up to 3 m high steel drum stack. This means that the lifting height for an ISO container is up to approx. 4 m and for a steel drum up to approx. 3 m. All handlings for this type of repository occur in the open if no temporary building is established.

This means that there are no barriers like ventilation system filters to detain radioactive dust. The only possible barriers left are the conditioning materials in the waste units if applicable for the particular type of waste.

The waste released to the open in form of dust is being dispersed towards the neighbours. As a consequence the neighbour is exposed via: inhalation, cloud exposure and ground exposure.

In case of failure of the procedure for decontamination of the personnel dust is carried away from the repository site by personnel and this may lead to exposure of other members of the public (e.g. family etc.) by ingestion, inhalation and ground exposure. E.g. a member of the personnel exposed to dust, prepares a meal at home, and ingestion of the radioactive dust takes places.

Near surface repository (NSR)

A NSR can be operated from the ground level by means of a crane or it can be operated from the inside by means of a fork lift and/or a crane just like in the ASR case. In the first case a temporary building may be set up, just like in the ASR case. If the temporary building is not established there is no other barrier except for the conditioning material in the waste unit (if present).

The containers are going to be stacked: e.g. max 4 ISO containers stack or up to 3 m high steel drum stack. In case of a repository operated from the ground level it means that the packaging drop height is up to approx. 10 m while in case of a repository operated from the inside the lifting heights for ISO containers is up to approx. 4 m and for steel drums up to approx. 3 m. The release modes and consequences are the same as described for the ASR.

Medium deep repository (MDR)

A MDR can be operated from the ground level (SOFA – Shaft Operated From Above) by means of crane or from the inside (SOFI – Shaft Operated From Inside ; COFI – Cavern Operated From Inside) by means of cranes and fork lifts. For SOFI and COFI a ventilation system with a filter that will detain all radioactive dust. In all repository types containers are stacked e.g. max 4 ISO container stacks or up to 3 m high steel drum stacks. The drop height from the crane may be between 10 - 100 m, 0 - 100 m and 0 - 4 m for the SOFA, COFI and SOFI repositories respectively. The consequences of release to the atmosphere are the same as described for the ASR.

Borehole

The hazards are not relevant since there is no dispersible waste to be stored in the borehole. A canister dropped in a borehole might break when reaching the bottom of the hole. In such a situation it may be necessary to recover and repack the waste.

Assessment

While it is hardly likely that a significant dose is carried out by personnel having forgotten to be decontaminated, it cannot be ruled out that dust released and dispersed by the wind will create a dose above the limit of 1 mSv for a neighbour. Therefore, quantification of the risk for this hazard for all relevant repository types is carried out in section 8.4.3 of this report.

Quantitative screening

The minimum required amount dispersed to give a dose of 1 mSv to a representative neighbour have been calculated for the waste types identified above as being dispersible by air. The results are shown in Table 8.1, and it may be evaluated from this, that only waste type 4, 8, and 21 may have the potential to cause a release resulting in a 1 mSv dose.

Table 8.1 Results of calculations of the required activity released for the neighbour to receive a dose of 1 mSv. Grey shaded waste types have been evaluated as not having the potential of giving the reference dose.

Waste type	Critical release, GBq	Fraction of total activity of waste type %	Corresponding number of waste units	Packaging type
1	831	35	5.5	Steel container
3	866	1.8	1.8	ISO container
4	55	0.2	0.4	ISO container
8	2.8	0.1	0.02	Steel container / 210 l drums inside
9a	38	457	N/A	210l drums
10a	110	16,277	N/A	210l drums
10b	2.0	0.1	4.5	210l drums
21	0.2	0.1	0.1	ISO container

Ineffective radiation shielding

This hazards includes the risk related to:

- The packing process was not performed to standards
- Physical damage to waste packaging during transport or handling
- Corrosion of the packaging due to humidity or water in the temporary storage at DD.

These events result in ineffective radiation shielding, so the personnel of the repository may become exposed to high doses of radiation. A safety barrier against this is that all waste is checked before leaving DD for transportation to the waste repository. This type of hazard is the same for all types of repositories, and although the safety of the workers at the repository is important, it is outside the scope of this report.

8.3.2 Mechanical damage to the repository due to human activities

The main scenarios related to these accidents are:

- Drilling activities at or near the repository site when the repository has been forgotten (assumed to happen after 300 years)
- Building or excavation activities at the repository site when the repository has been forgotten
- Damage to the repository during construction or operation of the repository.

Excavation and drilling activities

The excavation and drilling scenarios concern the situation where the repository has been forgotten, i.e. after 300 years. The scenarios are presented in form of the generic barrier diagram in Figure 8.3.

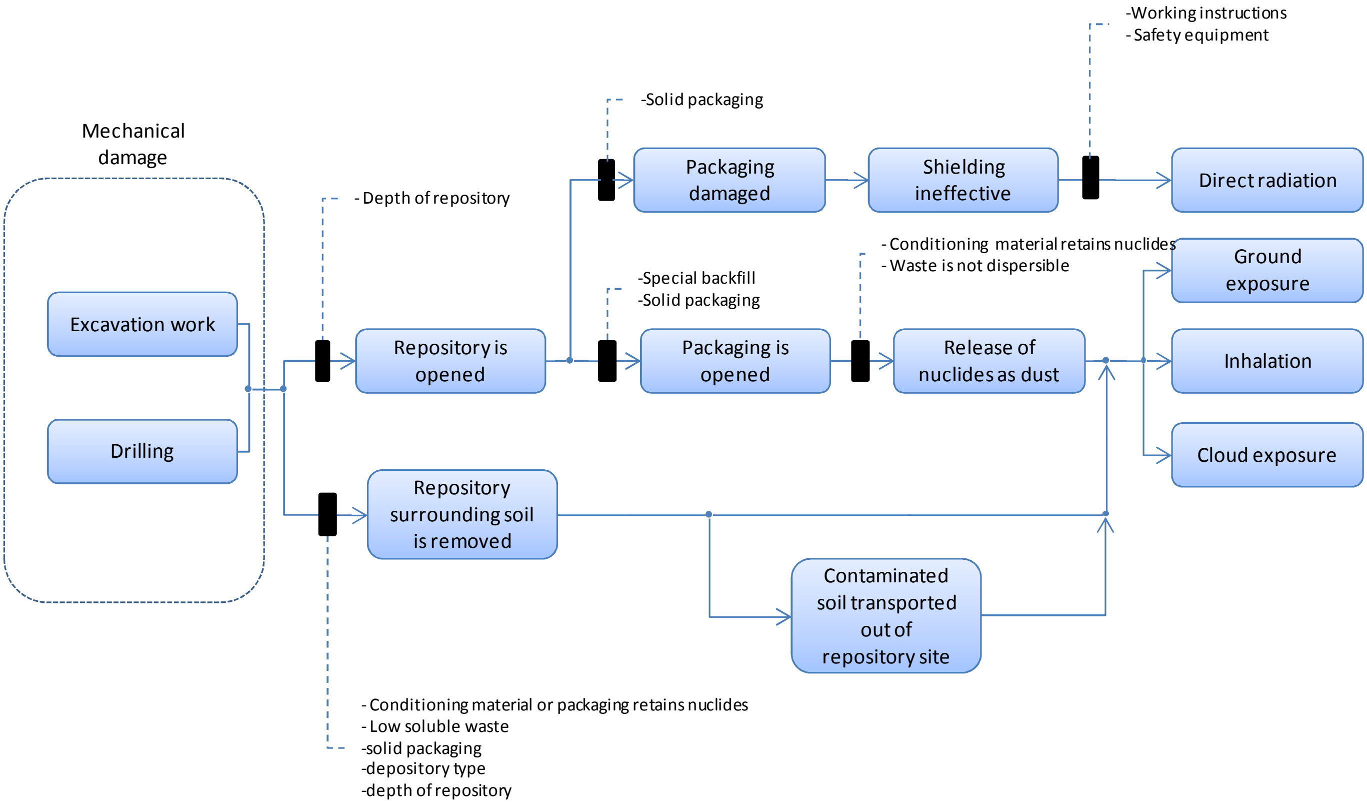

Figure 8.3 Barrier diagram for mechanical damage to repository where neighbour is exposed for radioactive waste in form of dust and drilling/ excavation worker is exposed to high doses of radiation.

The barrier diagram is supplemented by a discussion for the individual types of repository. When drilling or excavation work is performed at or near the repository, the repository may become opened and waste and/or contaminated soil may be released as dust, e.g. when loading to a truck that should transport the soil away. If the waste units are still identifiable, the excavation/drilling workers may realize something is wrong, and activities may be stopped. If not the work may go on for days. Both neighbours and drilling/excavation workers may become exposed to radionuclides as dust, and additionally excavation/drilling workers may become exposed to direct radiation from the radionuclides in the contaminated soil.

In general one may say that the deeper the repository is located, the less is the risk of the repository being damaged by excavation or drilling activities.

It has conservatively been assumed that after a longer time period all waste types may be dispersible by wind due to disintegration over time. Backfill may potentially also be contaminated and have dispersible elements.

Operational modes

Drilling and excavation activities are only relevant a long time after the repository was closed, and its existence has been forgotten.

Above surface repository

Excavation work at or near the repository will be likely to result in release of nuclides as dust, since the waste is located only a few metres under ground level, depending on the level of erosion and landscape changes occurred from the repository was made until the excavation work was performed.

Excavation work leaving the sides of the repository open may lead to runoff of waste with radionuclides to the surrounding areas. Should the excavation be limited to the central flat part of the repository, runoff is not expected. In that case it is assumed that rain water will pass down to the underground instead.

Drilling may also result in a release of radionuclides. The amounts of soil and thereby the amount of dust released may be less than during excavation activities.

Near surface repository

Regarding excavation and drilling activities the discussion under above surface repository generally apply. The concrete top slab on the NSR may though give a better protection of the repository and a possible warning to the drilling/excavation workers.

Medium deep repository and borehole

Excavation activities are not assumed to reach the depth of the waste units in a medium deep repository.

Drilling may result in a release of radionuclides. The amounts of soil and thereby the amount of dust released may be less than during excavation activities. Drilling may go through canisters.

Assessment

It cannot be ruled out that excavation or drilling activities at the repository once it has been forgotten will create a dose above the limit of 1 mSv for a neighbour.

Quantitative screening

The minimum required amount dispersed to give a dose of 1 mSv to a representative neighbour have been calculated for the waste types identified above as being dispersible by air. The results for the year 300 (conservative evaluation when considering decay of radionuclides) are shown in Table 8.2 and it may be evaluated from this, that waste types 3, 4, 8, 9d, 10b, 12, 13b, 15, 16, 17, 19, and 21 may have the potential to cause a release resulting in a 1 mSv dose, depending on type of activity (excavation/drilling).

Table 8.2 Results of calculations of critical release when the dose for a neighbour reaches 1 mSv, nuclides distribution for 300 years after opening repository. EX - excavation, DR – drilling. Grey shaded waste types have been evaluated as not having the potential of giving the reference dose.

Waste type	Critical activity release, GBq	Fraction of total waste type activity, % in year 300	Corre-sponding number of waste units (in year 300)	Packaging type	Relevant human activity	Time period of relevance, years
1	10,991	9,497		steel container		
2	0.83	141		ISO container		
3	1,315	63	60	ISO container	EX	300 - 1,000
4	41	0.91	2	ISO container	EX	300 - 10,000
8	0.63	0.86	0.2	steel container with 210 l drums inside	EX	300 - 10,000
9a	7.7	2,061		210l drums		
9b	7.4	1,130		210l drums		
9c	0.55	115		210l drums		
9d	0.65	1.2	0.25	210l drums	EX	300 - 10,000
10a	398	58,780,449		210l drums		
10b	0.64	0.93	37	210l drums	EX	300 - 10,000
11	49,021	2,501		steel container		
12	0.65	1.1	1.3	steel container with 210 l drums inside + canister	EX	300 - 10,000
13a	21.2	10	0.30	steel container or canister		300 - 10,000
13b	0.89	0.04	0.0011	steel container or canister	DR	300 - 10,000
15	0.64	0.38	0.023	steel container or canister	DR	300 - 10,000
16	0.65	0.10	0.0071	steel container or canister	DR	300 - 10,000
17	0.68	0.005	0.00045	steel container or canister	DR	300 - 10,000
18	0.87	43	2.6	steel container or canister		300 - 1,000
19	7.40	15	0.59	steel container	EX	300 - 10,000
21	0.70	0.51	0.41	ISO container	EX	300 - 10,000

Damage to the repository during construction

Damage to the repository during construction includes perforation of HDPE (High Density Poly Ethylene) membrane and cracks in concrete. The consequence of such damage may be increased flow of water through the repository.

Operational modes

Damage during construction of a repository will mainly have effect during the first 100 years of the repository lifetime. After that the damage is not expected to have significant effect of the flow through the repository.

Above surface repository

The top HDPE membrane may be locally perforated during the construction of the repository. It is less likely that this should occur after applying 2 m of soil above the membrane. A damaged top membrane may lead to water entering a dry type repository. This may again lead to increased corrosion rates of containers/drums and water seeping out of the repository through the drain pipes above the concrete layer and above the lower HDPE membrane. It is considered though, that the effect of this will be less that the effect from the water that entered the repository during the initial filling.

The bottom HDPE membrane may be locally perforated during the construction of the repository. During infill it will be protected by the concrete top slab and sand layer. If the top membrane is intact this would have no effect after filling and closing the repository since the waste units are considered dry. However during the infilling period water may enter the unprotected waste units and seep through the waste dissolving radionuclides. Normally this would be collected by the drainage pipes, but with a perforated bottom membrane, some of the water may seep through. It is hardly possible that significant amounts may pass this way, however care should be taken to reduce the amount of water reaching the waste units during the infill period.

Should both top and bottom HDPE membrane be locally damaged water may pass through the repository. The process will be very slow though, due to the low hydraulic conductivity of the concrete liner above and the clay liner below the HDPE membrane. Most of the water would flow to the drain pipes above the HDPE membrane.

Near surface repository

Damage to the repository during construction may be in form perforation of the HDPE membrane or cracks in the concrete of the top slab. The discussions under the ASR apply in general, except that it is already envisaged that there will be a roof structure above the repository during infilling. Cracks and lower quality of the concrete in local areas is considered included in the uncertainty of the hydraulic conductivity of the concrete used in the calculations relating to the long term consequences.

Medium deep repository

Damage to repository during construction may be in form of cracks in the concrete. This may increase the flow of groundwater into the repository, and this will most likely be discovered during the construction and/or the following period of filling waste units into the repository, and repair work may be carried out if evaluated necessary. If not discovered, the result may be an increased flow of water through the repository.

The effect of this will be larger if the surrounding soil has a large hydraulic conductivity. Cracks and lower quality of the concrete in local areas is considered included in the uncertainty of the hydraulic conductivity of the concrete used in the calculations relating to the long term consequences.

Borehole

The scenario for damage to the repository during construction work is not considered relevant for the borehole.

Assessment

it is not considered likely that damage to the repository during construction will lead to leak of significant amounts of radionuclides from the repositories.

8.3.3 Accidental release of drain water

The ASR has a drainage that leads water from inside the repository, should there be any, to a tank. The tank is assumed to have a volume of 10 m^3.

The cause of water entering the repository may be deterioration of the top membrane or damage to the membrane during construction water may in small amounts seep into the repository. The amount passing through the repository in such situations has been set to 4.7 m^3 per year. Once sufficient water has entered it will start flowing to the drainage tank.

The water in the tank will be measured for contamination before being disposed safely. The following safety measures have been assumed to be installed.

- The tank will be placed in basin with higher volume than tank
- The basin edge higher than concrete edge in repository, and basin protected from rain. This prevents overflow of basin.
- Water inside tank must be pumped out of top of tank, i.e. no seeping to recipient from
- Tank level indicator and alarm to control room
- Procedure for regular emptying of tank both on time and volume
- Electronic leak detection system in top membrane
- If evaluated necessary a layer under top membrane reducing the flow may be introduced, should there be a hole in the membrane, may be considered. Such a membrane could be a bentonite membrane.
- The ASR top is build with inclination, i.e. practically no flat area at the top, increasing run off and reducing the risk of concave areas creating ponds on top of the repository.
- Automatic measurement of contamination of water in combination with manual measurement.
- Automatic lock preventing opening of a valve to lead contaminated water to a recipient. Lock may be overridden manually.

Considering these safety measures, the two most likely scenarios in relation to such operational errors is considered to be

- Water drained from inside the repository is erroneously led to recipient.
- Water is not drained through the drain system either because the drain system is not working or the tank is not being emptied. Water will enter the ground at the side of the repository

Water drained from inside the repository is erroneously led to recipient.

It may be envisaged that contaminated water drained from the inside of the repository will be led to a recipient by error. This could happen at any time once water starts to show up in the drains. Given the procedure to empty the tank every quarter or every time there is 1 m^3 in the tank, it may conservatively be considered that a maximum of a full year volume of 4.7 m^3 is sent directly to a recipient. This may cause a dose at a neighbour. The radionuclides in the water will be a mixture of the different waste types, and thus a quantitative screening of waste types is not done.

Water is entering the ground

If the drainage system is not working or if the tank for some reason is not emptied, for example due to that the repository has been abandoned before and not sealed properly. In such situations the water will enter the ground at some point at a rate of 4.7 m^3 per year. Since stoppage of the drainage may happen in the early years of the repository lifetime, the short lived radionuclides that do not have an effect in the long term evaluations may show up in the ground water. Depending on the geology that the repository is located the contaminated water may flow to a recipient, and thus may result in a dose of 1 mSv or more at a neighbour. The radionuclides in the water will be a mixture of the different waste types, and thus a quantitative screening of waste types is not done.

8.3.4 Aircraft crash or meteorite impact at repository site

These scenarios are in general meant to represent objects falling from the sky with a possible subsequent explosion, fire, or heat development. Large meteorite impact resulting in damage more severe than the damage from the release of radionuclides is outside the scope of the analysis (Ministry of Interior and Health, 2008).

Not taking into account the fatalities directly related to an aircraft crash, the consequences of the two scenarios are similar. Therefore they have been treated as one in the following sections.

Aircraft crash or meteorite impact

An aircraft crash and a meteorite impact may have the potential of damaging the repository and the waste units, causing dispersion of the waste as dust, as also illustrated in the safety barrier diagram in Figure 8.4.

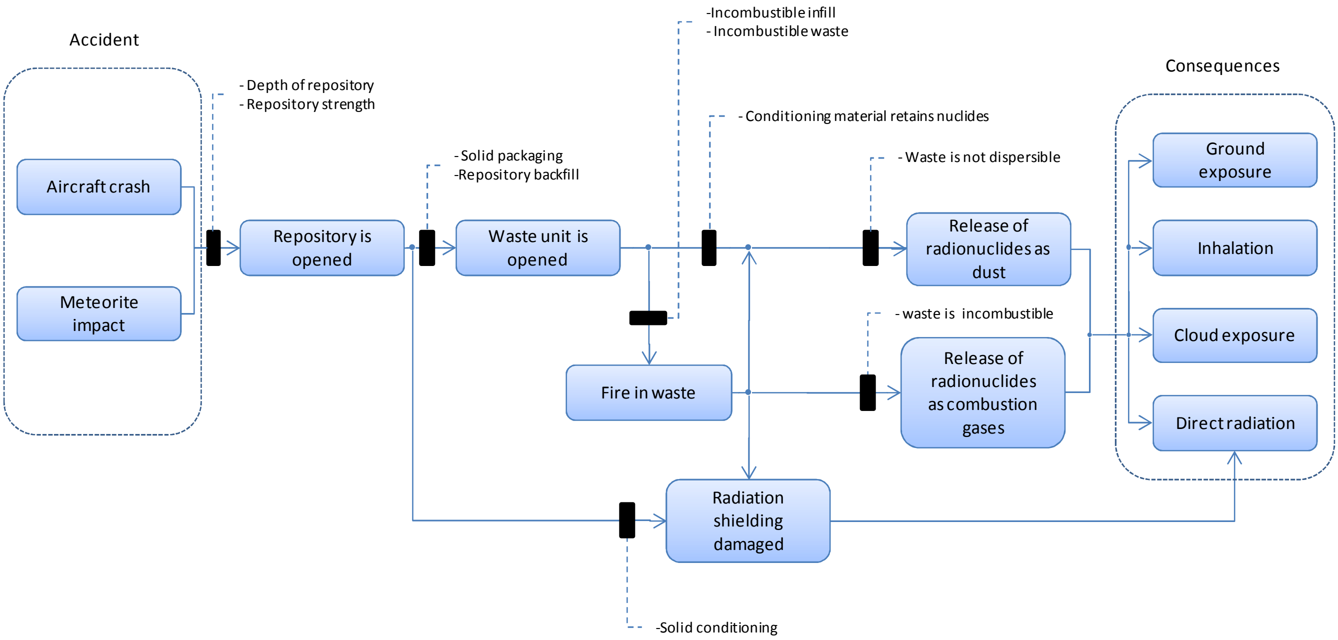

Figure 8.4 Safety barrier diagram for aircraft crash and meteorite impact

A fire subsequent to an aircraft crash may occur and develop due to ignition of the jet fuel from the aircraft, and the fire may spread to combustible waste (e.g. graphite) or conditioning (e.g. bitumen) potentially releasing radionuclides as combustion gas.

A fire in the waste may also occur subsequent to a meteorite impact as a result of the heat from the meteorite, which may have a temperature of several thousand degrees.

With or without a subsequent fire, it may occur that the radiation shielding effect of the repository structure, the backfill, and the container/drum in which the waste is packaged is impaired. A rescue team or persons interested in seeing the crater may become exposed to direct radiation from radionuclides at the location of the impact.

Both types of incidents may occur at any time of the time period covered by the analysis.

During the first year of the operational phase while the repository is being filled with waste from DD, an aircraft crash or a meteorite impact would have the potential of causing release nuclides from the intermediate storage building. However considering the short period of time (one year) and the limited area of the intermediate storage building, it is considered highly unlikely that an airplane crash related to the in-flight phase or a meteorite impact should occur. This risk contribution is the same for all repositories, and is evaluated to be negligible, and is therefore not discussed in the sections below.

Above surface repository

The above surface repository may be subject to mechanical damage and release of nuclides from an aircraft crash or meteorite impact at the repository.

Certainly not all aircraft crashes and meteorite impacts will have the potential of penetrating through the layer of earth and backfill on top of the repository. However, that it is possible for this type of repository may be illustrated by the pictures from fairly recent incidents in Figure 8.5 and Figure 8.6.

Figure 8.5 Crater from airplane crash in Iran, 16 July 2009 (CNN, 2009)

Figure 8.6 Crater resulting from impact of a 1-m wide meteorite in Peru, 16 September 2007 (Miguel Carrasco/La Razon/Reuters, 2010)

Both an aircraft crash and a meteorite impact at the repository site are events with a very low probability of occurrence. However, given the long period of time covered by the analysis (10,000 years) and the potentially large amounts of radionuclides being released to the atmosphere, it has been decided to quantify the risk for the above surface repository.

If the crater is located at the inclined side of the repository, and has a shape that makes it possible to have runoff of nuclides to the ground outside the repository, this runoff may also continue to nearby rivers etc. Such a scenario is considered possible if a large commercial aircrafts crashes into the side of the repository, e.g. by sliding into and damaging the inclined repository side.

Near surface repository

As shown in the section above, a repository located at ground level or just below, such as the near surface repository, may become damaged from an aircraft accident or a meteorite impact.

The near surface repository will be better protected from impact from above than the above surface repository due to the steel reinforced high quality concrete top slab, and therefore the probability that an impact would affect the waste is lower. However, after a certain amount of time, say 300 years, the concrete will have deteriorated and the steel corroded, and the risk will then be the same for the two repositories.

Based on this and the argumentation for the above surface repository in the section above, it has been decided also to quantify the risk for the near surface repository.

Medium deep repository and borehole

It is assumed that once the waste is stored inside the repository, it will neither be affected by an aircraft crash nor by a meteorite impact.

The risk of release of radionuclides is for these scenarios and repository types considered negligible, and is thus not quantified.

Quantitative screening

Calculations of the critical activity release for achieving 1 mSv dose for the neighbour have been calculated for waste types present in above and near surface repositories, and the results are shown in Table 8.3. It may be seen that for most waste types it will be possible to have a release that result in a critical dose for the neighbours. While some waste types may not be relevant for the full time period covered by the analysis (or not at all), the fraction of waste types relevant for the full period is large. Based on this, and the fact that it is the aggregated dose from the release of different waste types that is to be considered in the dose calculation, it is conservatively assumed, that a release caused by an airplane crash or a meteorite impact result in a critical dose for the neighbours.

It may have been assumed that for the first 100 years the concrete structure of the NSR will prevent damage from an airplane crash, but not from a meteorite impact. However, considering the period of 10,000 years, a risk reduction relating only to the first 100 years, will not be significant, when only considering the risk of a neighbour receiving a dose of 1 mSv.

Table 8.3 Calculations of critical activity release after repository damage caused by fall of object from the sky causing a 1 mSv dose. Grey shaded waste types have been evaluated as not having the potential of giving the reference dose.

Waste type	Critical activity release, GBq	Fraction of total waste type activity, %	Number of released waste units	Container type	Time period of relevance, years
1	876	36.5	5.83	Steel container	0-30 years
2	190	3.32	0.332	ISO container	0-10,000 years
3	913	1.95	1.85	ISO container	0-10,000 years
4	58.0	0.162	0.373	ISO container	0-10,000 years
8	2.93	0.118	0.024	Steel container with 210 l drums inside	0-10,000 years
9a	40.3	481		210l drums	-
9b	12.9	311		210l drums	-
9c	3.05	32.6	61.3	210l drums	0-10,000 years
9d	2.31	0.155	0.0309	210l drums	0-10,000 years
10a	116	17,160		210l drums	-
10b	2.07	0.117	4.69	210l drums	0-10,000 years
11a	2,314	13.4	2.01	Steel container	0 - 100
12	2.47	0.153	0.168	Steel container with 210 l drums inside	0-10,000 years
18	4.92	5.01	0.301	Steel container or canister	0-10,000 years
19	2.16	4.32	0.173	Steel container	0-10,000 years
21b	0.204	0.150	0.120	ISO container	0-10,000 years

8.3.5 Fire

Fire may occur inside or outside a repository. The identified causes of fire are fire in truck, forklift or crane inside/outside the repository and fire in installations inside the repository.

A fire may also occur after an aircraft crash (see section 8.3.4) and after an uncontrolled release of Wigner energy from non-annealed graphite (see section 8.3.6).

The fire may develop if no effective fire fighting is performed, and the fire may spread to combustible waste or conditioning, such as graphite, bitumen, plastic and paper. Additionally, the waste unit may become damaged by the heat and crack open. Radionuclides may be released as dust dispersed with hot smoke or as combustion products from the fire. The hot smoke and potentially also dust particles will be dispersed by air and may expose a neighbour to radiation from ground exposure, cloud exposure, and inhalation.

- Waste type 1 contains graphite with radioactive nuclides, and this may burn and form gaseous combustion products.
- Waste type 8 contains paint dust that may burn and combustion products may contain dust particles of radioactive nuclides. The paint dust is mixed with sand, and will not burn easily. The waste is located in concrete lined drums placed inside ISO containers.
- Waste type 9 contains bituminised evaporator residues. Bitumen may burn and combustion products may contain dust particles of radioactive nuclides.
- Waste type 10 contains plastic rubber and paper, amongst others. This may burn and combustion products may contain dust particles of radioactive nuclides.

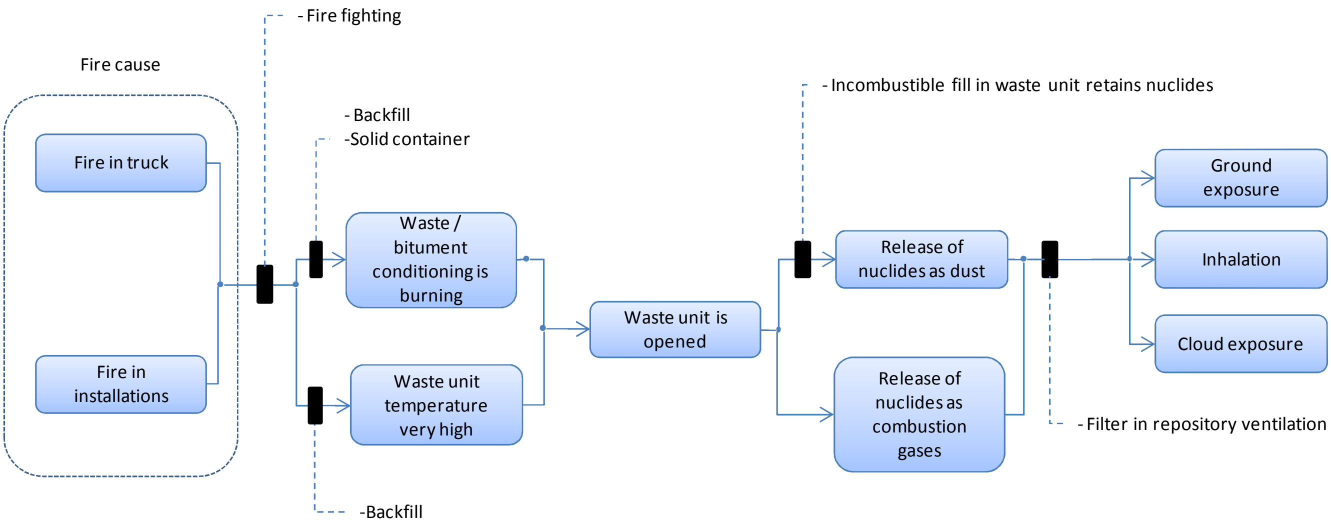

Figure 8.7 Safety barrier diagram for initiating event "Fire".

Above surface repository and near surface repository

The ASR and NSR will most likely not have electrical installations, and therefore the only source of fire considered is a truck, a forklift, or a crane. For the fire to propagate to the waste, it has to have a certain duration and the vehicle at fire must be located near the waste unit with combustible materials. This, however, cannot be ruled out.

Fire fighting may be done with portable fire fighting equipment, and it may be assumed that a quick reaction on fire fighting will prevent the propagation of the fire. A fire spreading to the waste may be limited by repository backfill covering the waste units.

Medium deep repository

A fire inside a medium deep repository may be caused by a fire in a forklift or in installations inside the repository. The repositories operated from the ground level by cranes will be safe against fires of these types.

Fires in electrical installations may spread to the waste if there is sufficient combustible material present. With a proper design (see e.g. recommendations below) it is assumed that the amount of combustible materials relating to electrical installations is not sufficient to sustain a fire long enough to affect the waste.

A forklift running on diesel may catch fire and have the potential to develop sufficiently to affect the waste units and the combustible waste, if located near these.

A fire inside a medium deep repository, especially the cavern solution, may reach higher temperatures for a larger area than a fire outside, just as is seen for fires inside tunnels, where it is difficult for the heat to be vented away. At some point though, the oxygen content of the air inside the repository will become too low to sustain the fire, unless fed sufficiently by a ventilation system.

Given that there is a ventilation system providing fresh air into the repository, and filtering the air leaving, this may prevent dust and gases from being released to the air. It may be questioned though if such a filter will be able to hold the possible large amounts of particles and gas from a fire. Therefore it is recommended that a procedure is made of how to shut down the ventilation in case of fire. This will also reduce the supply of oxygen to the fire.

It cannot be ruled out that a fire may develop in the medium deep repository and result in release of nuclides as gas or combustion products. Therefore this scenario is quantitatively analysed.

Borehole

Fire is not considered for the borehole since there are no combustible materials present.

Quantitative screening

Calculations of critical activity release for achieving 1 mSv dose for the neighbour are made for the combustible waste types specified above. The results are shown in Table 8.4.

Table 8.4 Calculations of the critical radioactivity release when doses for neighbour reaches 1 mSv. Grey shaded waste types have been evaluated as not having the potential of giving the reference dose.

Waste type	Critical activity release, GBq	Fraction of total waste type activity, %	Number of released waste units	Container type
1	2,847	118		Steel container
8	9.5	0.38	0.077	Steel container with 210 l drums inside
9b	41.9	1012		210l drums
9c	9.9	106		210l drums
9d	7.5	0.50	0.10	210l drums
10b	6.7	0.38	15	210l drums

It seems unlikely that all the nuclides from 15.3 drums (waste type 10b including contaminated glass, plastic, rubber, paper, rags, and aluminium steel) should be released in a fire, considering that it would have to be as dust particles carried with the smoke out through leaks in the containers, and that not al the waste inside the drums is combustible. Neither is it likely that the mixture of sand and old paint will burn and carry nuclides as dust particles out of the drums (waste type 8). However, the drums containing bituminized waste (waste type 9d) may be leaking or heated sufficiently to catch fire. Only 10 % of the radionuclides in one drum being dispersed by air (or 1 % from each of 10 drums) it can not be ruled out that this may happen.

8.3.6 Chemical reactions, development of gas, and energy release

No accidental scenarios related to chemical reactions were identified.

Rn gas will develop from Ra sources in the tailings in waste type 21. This is a foreseen development, and no accident scenarios have been identified in relation to this.

The following hazard is discussed in the following:

- Release of Wigner energy from non-annealed graphite in waste type 3.

Release of Wigner energy

Waste type 3 includes graphite from DR1, DR2, and DR3. The graphite has served to absorb neutrons, and includes Wigner energy accumulated during this process. The Wigner energy may be released by heating up the graphite to temperatures 50 ^{o}C above the temperature that the Wigner energy was accumulated (International Atomic Energy Agency, 2006), normally at least 250 °C.

The scenario considered is an unexpected release of Wigner energy from graphite in the waste, and the graphite is heated up in combination with a fire near the graphite waste. This may cause very high temperatures and result in CO and CO_2 gases being released due to combustion of graphite in the presence of air. Such gases when produced from the graphite in waste type 3 will contain ^{14}C. The gases may be dispersed and reach the neighbours to the facility, as illustrated in Figure 8.8.

Possible causes for the release of Wigner energy are described in the fault tree in Figure 8.9 with the top event "Wigner energy release".

The scenario is in the following discussed with respect to the different repository types and phases to evaluate the need for quantification of the scenario.

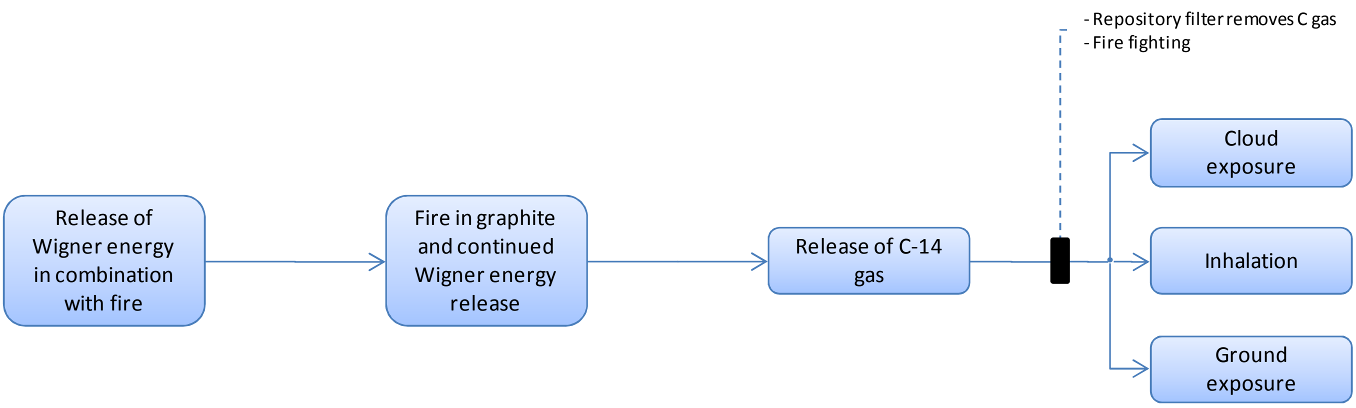

Figure 8.8 Barrier diagram for consequences following release of Wigner energy

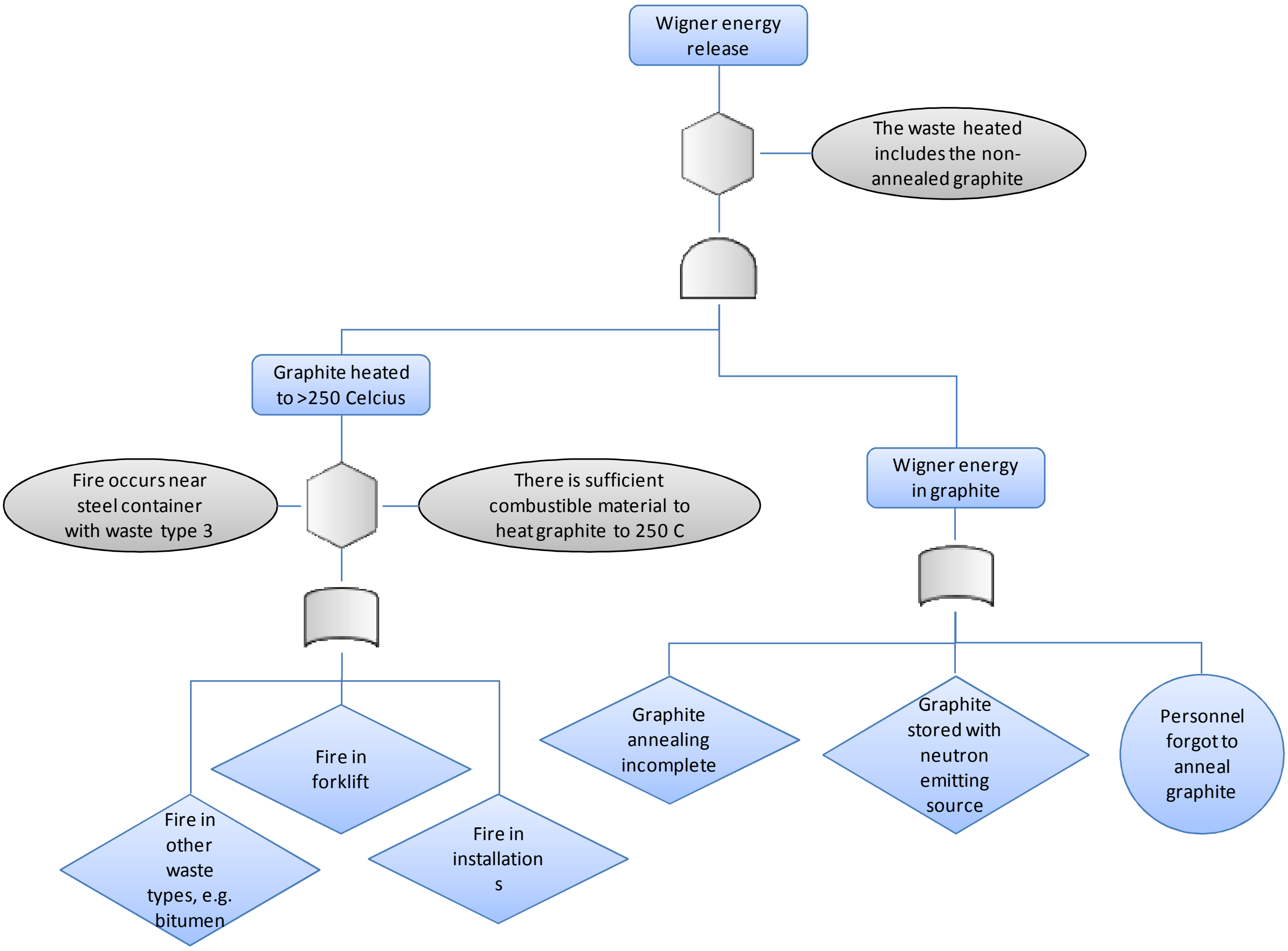

Figure 8.9 Fault tree for release of Wigner energy from graphite in waste type 3

Operational modes

The scenario is relevant for the operational modes until the closure of the repository, since after that time there will be no source of fire, except if an unauthorized person breaks in to the repository and starts a fire.

Above surface repository

The ASR will not be very prone to fires with the potential of heating up the graphite since there will be only few sources of fire and ignition. It is not very likely that other types of waste will auto ignite. In case of a fire at the ASR, there will be good ventilation, although nurturing the fire, also having a cooling effect.

However, if there is a fire and combustion products containing ^{14}C are developed, there will be no barriers preventing them from being dispersed towards the neighbours.

Near surface repository

The NSR will not be very prone to fires with the potential of heating up the graphite since there will be only few sources of fire and ignition. It is not very likely that other types of waste will auto ignite.

Should a fire occur and combustion products containing ^{14}C be developed, the temporary building above the repository will act as a limited barrier in preventing the combustion products from being dispersed towards the neighbours. The damage to the concrete walls should be investigated and if necessary repaired.

Medium deep repository

In the medium deep repository there may be electrical installations and the use of forklifts and both may serve as a source of ignition and fire. It is questionable though if these fires may become large enough to cause heating of the graphite to 250 °C.

Should the graphite be heated and combustion products, i.e. CO and CO_2, containing ^{14}C be developed, it is assumed that there is a filter at the venting of the repository. It is questionable though if this filter will be able to catch the amount of combustion products produced. A closing mechanism may be provided for the ventilation as recommended in the fire scenario. As also discussed in the fire scenario, the heating from the fire will be larger for enclosed spaces such as in the cavern type.

Borehole

Graphite will not be stored in the borehole and therefore not relevant.

Assessment

Although not very likely, it cannot be ruled out that a release of Wigner energy will happen and cause a release of radionuclides. Thus a quantitative screening is carried out below.

Quantitative screening

The required fraction of the total waste to obtain a critical dose at a neighbour is shown in Table 8.5. It may be seen that more than 20 times (2,356 %) the full amount of radionuclides available as ^{14}C is required for the neighbour to receive a dose of 1 mSv. Hence this scenario is not considered further.

Table 8.5 Calculation of critical radioactivity release for reaching 1 mSv dose for the neighbour

Waste type	Critical activity release (GBq)	Fraction of total waste type activity (%)
1	56,590	2,356

8.3.7 Malicious damage and acts of war

This section touches briefly on the risk of malicious damage and acts of war.

Examples of hazards are:

- Theft of nuclides without knowing the potential danger
- Theft of nuclides with the intention of using them either for terror or other purposes
- Intentional/unintentional bombing of the repository during war.
- Explosions inside the repository as an act of terror.

Generally, it can be said that the deeper in the ground the repository is situated, the more difficult it will be to steal from the repository or damage it to a degree where radionuclides are spread to the surroundings.

8.3.8 Intrusion by living organisms

Intrusion by living organisms includes the risk related to:

- Animals/organisms entering the repository and consuming waste with radio nuclides.
- Animals/organisms entering the repository and carrying waste with radio nuclides outside on their fur, skin, etc.
- Plants causing decay of structures, their roots entering the waste absorbing radionuclides, and transporting it outside.

These scenarios are only considered to be relevant for the ASR and NSR.

Animals/organisms entering the repository

While hazards related to animals may cause harm to these, it is considered highly unlikely that these scenarios will result in significant exposure to neighbours, and is thus not considered further.

Plants causing decay of structures

It is likely that over time, plants will gain access to the ASR and the NSR type repository. The plants may absorb radionuclides dissolved in the water inside the repository and transport them out to the other side of the protective barrier. This will be a slow process, but may however continue for hundreds of years thus resulting in accumulation within the plants.

While the repository is known, it is assumed that this will be a controlled process and that plants with the potential to penetrate the membrane will be removed. Should misjudgement lead to plants reaching the repository, it is assumed that the dead wood is handled with the knowledge that it may contain radionuclides.

After the repository is forgotten, trees may grow on top of the repository and roots may reach water with dissolved radionuclides, resulting in absorption and accumulation of radionuclides in the trees. Penetration of the barrier (membrane) protecting the radioactive waste will also cause increased water flow. This will most likely happen for the ASR and the NSR at some point after the repository has been forgotten, and is therefore not considered an accidental scenario.

The scenarios are not relevant for a medium deep repository or a borehole.

The risk has not been quantified.

8.3.9 Natural hazards

Natural hazards include the risk related to:

- Flooding by surface water
- Flooding by groundwater
- Earthquake, and earth settlements
- Sea level rise.

As stated in (Ministry of Interior and Health, 2008), the risks related to large natural hazards, where either the likelihood is extremely low, or where the consequences from the natural hazard is significantly larger than the additional consequences related to the radioactive material, shall not be part of the present analysis. Examples of such are large earthquakes, meteor strike, ice age, and volcanic activity.

The risk related to natural hazards is not quantified. The argumentation for this is given in the following.

Flooding by surface water

Flooding of the repository by surface water may happen either as a result of severe amounts of rainwater, melting snow or as sea water level rise.

It is assumed that a future repository is located in an area not prone to flooding by sea water, and as such flooding by sea water is not considered a possibility unless severe changes in the sea water level should occur. That scenario is discussed later in this section under the heading Sea level rise.

In the following flooding of a repository by heavy rainfall is considered, assuming that a drainage system is provided around the ASR and NSR repository.

Above surface repository

The top of the ASR is made of soil above an HDPE membrane. The top soil may be washed away by heavy rainfall, especially in the early years before vegetation is strong. While the repository is under operation or is still known, the consequence of this is limited, and it is assumed that it will be restored after the water has disappeared.

An extreme rain fall during the period of filling the repository i.e. before the HDPE top membrane has been placed to cover the waste units, may cause problems with excessive amounts of rainwater passing through the waste containers and subsequently flowing over the low concrete wall at the side of the repository area. However this scenario can be ruled out by design, considering the short time period it is relevant. It is recommended that the side wall of the ASR is without openings and has a height that reduces the probability of the scenario such that the risk is negligible.

Within the first 100 years when the membrane is assumed to be water tight, the water will not enter the repository, unless the membrane is damaged. The consequences of such damage have been considered as part of mechanical damage to the repository analysed in section 8.3.2.

After 100 years it is assumed that the plastic membrane is no longer water tight. The scenario of rain water passing through the repository after 100 years is thus not considered an accidental event.

After 300 years, when the repository is forgotten, the membrane and concrete is no longer considered to be water tight. Repeated significant rainfall over many years may cause erosion and thereby reveal the repository and though less likely also the waste. While the frequency of this occurring is hardly quantifiable, it will be higher the higher in the ground the repository is located.

Near surface repository

During the first years while the repository is being filled with waste from DD, extreme rainfalls in combination with damage to the temporary structure above the repository may cause large amounts of water to enter the repository. Such damage to the structure above the repository may e.g. be caused by the load from large amounts of snow. This may create problems with removing the water that has entered the repository, and in terms of increased corrosion of waste units. While the removal of water inside the repository may pose a risk the workers and include clean-up cost, this will not have consequences for the neighbours.

After the repository is closed, but within the period while the repository is still known, the near surface repository is not expected to be damaged by rainwater due to the size, strength and weight of the repository, including the top slab. Large amounts of snow may increase the weight on the top slab significantly and cause cracks. Cracks in the top slab including the membrane will increase the flow of water through the repository into the groundwater. The consequences of this will be a variation of the long term conseqeunces.

Surface water may enter the repository under heavy rainfalls through cracks in the concrete if the membrane is damaged also. This may also happen during normal rainfalls or snow melting, though at a lower rate.

After 300 years, when the repository is forgotten, the membrane and concrete is no longer considered to be water tight. Repeated significant rainfall over many years may cause erosion and thereby reveal the repository and though less likely also the waste. While the frequency of this occurring is hardly quantifiable, it will be higher the higher in the ground the repository is located.

Medium deep repository

During the first years while the repository is being filled with waste from DD, extreme rainfalls in combination with damage to the temporary structure above the repository may cause large amounts of water to enter the repository. Such damage to the structure above the repository may e.g. be caused by the load from large amounts of snow. This may create problems with removing the water that has entered the repository, and in terms of increased corrosion of waste units.

While the removal of water inside the repository may pose a risk the workers and cost of clean-up and water treatment, this will not have consequences for the neighbours.

After the repository is closed, it is considered a wet repository.

Borehole

The borehole will be located at a depth where it will not be affected by rainwater. Even if rainwater should enter during the filling of the repository, it would not be considered a problem, since the repository is not considered a dry repository.

Flooding by groundwater

The repository may become flooded by groundwater before closure. This may cause faster deterioration of the waste containment, and thereby wash out of nuclides will occur at an earlier time than if the repository was kept dry. The worst scenario is that the groundwater level fluctuates, i.e. enters and leaves the repository frequently, since this may cause increased flow of water through the repository and nuclides to be washed out earlier than expected.

Above surface repository

If the above surface repository is flooded by groundwater fluctuating at the ground level wash out of nuclides from the waste may be increased. This may happen both before and after the closure of the repository, although most likely it will be detected in the period while the repository is being monitored. The frequency of this occurring is not quantifiable without knowing the location of the repository, and would be related to climatic change. The consequences of the scenario will be a variation of the long term consequences.

Near surface repository

Fluctuating groundwater level around the level of the repository may wash out nuclides. However the hydraulic conductivity of the concrete walls will make the water flow very low, and thus this will be a variation of the groundwater scenarios covered as part of the long term consequences. The frequency of this occurring is not quantifiable without knowing the location of the repository, and would be related to climatic changes.

Medium deep repository and borehole

These repositories are expected to be located below groundwater level. Therefore this scenario is not relevant.

Earthquake and earth settlements

Earthquakes and earth settlements may theoretically cause the walls of the repository to crack or even fail.

According to GEUS (2010), the number of registered earthquakes in Denmark, including Danish waters, averages to 99 per year for the period 2000-2009. The size distribution of the earthquakes is shown in Figure 8.10 for 1930-2009 and 2000-2009. The largest earthquake measured in Denmark was 5.2 on the Richter scale, registered in 1980.

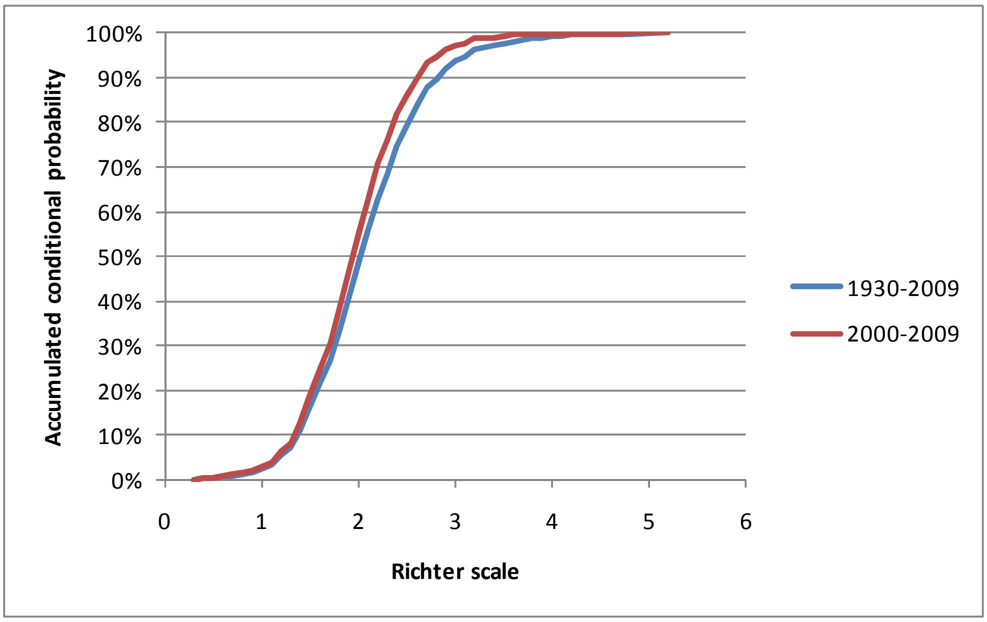

Figure 8.10 Size distribution of earthquakes registered in Denmark (GEUS, 2010)

In general it is not considered likely that smaller earthquakes will be able to cause additional consequences in terms of increased release of nuclides from the repositories. This is argued below for each of the repositories. An earthquake happening during the first year where the waste is being placed in the repository might result in a dropped container/drum and dust may be released. This is considered to be a sub-component of the risk evaluated in section 8.3.1, and is the same for all repositories.

Large scale earthquakes are not considered according to (Ministry of Interior and Health, 2008).

Although the risk of release of nuclides from inside the repositories is not expected to be increased by earthquakes, it is recommended, due to worker safety, that earthquake design requirements are established for a medium deep repository.

Above surface repository

The membrane of an above surface repository may become damaged due to an earthquake. The increased water flow through the repository due to this may cause increased corrosion and wash out of nuclides, once the containment is deteriorated. The risk from this is considered covered by the calculations of deteriorated membranes for long term consequences.

Near surface repository

In case of cracks in the repository wall, the repository may be filled with groundwater faster than normally expected, if the repository is located below groundwater level. This will cause faster deterioration of the waste containment. The water filling will still be slow though, and depend on the water flow through the soil surrounding the repository.

If this happens during the operation phase while the repository is kept dry, this will result in increased amounts of water that needs to be drained, however this will not increase the risk of radiation.

If the earthquake happens after the closure of the repository, water filling of the repository will be increased. However, the flow through the repository will not increase, since the limiting factor for the water flow through the repository after 30-100 years will be the soil surrounding the repository. Therefore, there will be no additional consequences from this scenario.

Medium deep repository

A smaller earthquake will not be able to damage the walls of a medium deep repository.

Borehole

A borehole will not be affected by an earthquake, since there are no walls that may be damaged.

Sea level rise

It is assumed that a future repository is located in an area not prone to flooding, and as such a minor rise in sea level will not produce a risk.

In case of major a sea level rise, the above surface repository may be at risk of being flooded or even submerged with possible increased flow of water through the repository. In such a scenario the consequences of the sea level rise itself would be much more extensive for the population of the earth than the damage to the repository would be.

Sea level rise is a slow process and is followed closely by scientists and governments, and hence there will be sufficient time to take precautions, should the level rise significantly during the period while the repository is known. Based on this, and an assumption that the location of the repository is chosen with consideration to a reasonably foreseeable increase in sea water level, this scenario is only relevant for the distant future say 300 years from now, when the repository may be forgotten.

Above surface repository

The above surface water may be flooded with sea water after the repository has been forgotten, and with the flooding water may flow through the repository, washing out nuclides. The consequences of an increased flow due to surface water are considered as part of the long term consequences. Worst case the repository may be washed away by the sea after the repository is forgotten, dispersing the radionuclides with the sea water. While this may have severe consequences, the probability of this occurring is hardly quantifiable.

Near surface repository

During the first 30 years of operation until closure of the repository, it is not considered possible that the sea level will rise to a level reaching the repository.

After closure it is not likely that the repository will be damaged by sea water rising above the level of the surface of the repository.

Medium deep repository, and borehole

During the first 30 years of operation until closure of the repository, it is not considered possible that the sea level will rise to a level that it will reach the repository. After sealing of the repository, there will be no additional consequences from a sea level rise.

8.3.10 Worker safety

This hazard group covers all types of normal accidents at a working site

The risk of such accidents is handled by normal safety precautions at work sites. If the working conditions are approved by the authorities, the risk to the workers is deemed acceptable. Therefore the risk to workers has not been quantified as part of this study.

Examples of such risks are:

- Trips and fall from heights
- Accidents during construction work
- Handling accidents
- Fire inside or outside repository
- Inadequate radiation shielding.

Trips, falls, and accidents during construction

The risk from trips, falls, and accidents during construction are not related to the nature of the substances stored at the repository, and may happen at any working location with large heights.

The consequence of trips, falls, and accidents during construction will be higher for the medium deep repository than for the others due to the depth of the repository.

Handling accidents

Drop of waste units from heights may result in release of radionuclides as dust, and thus in inhalation of radionuclides for the workers near the accident.

It is not foreseen that there will be a significant difference in the risk to the workers in the different types of repositories, although the escape from the accident may be more difficult in the medium deep repositories.

Procedures on how to act in case of damage to a wate unit and appropriate training in relation to these should be provided for the workers at the repository.

Fire inside repository

A fire in a medium deep repository operated from inside the repository is more likely to have severe consequences than a fire in the above or near surface repository. Workers may be trapped inside the repository, and provisions should be made for escape routes and fire fighting according to applicable laws and regulations.

The amount of flammable material inside a repository is limited as described in section 8.3.5, and the most likely source of ignition is a forklift catching fire.

8.3.11 Conclusion of the hazard screening

Based on the hazard screening carried out in section 8.3 the following hazards were selected for quantitative analysis based on their expected potential to result in a dose of 1 mSv or more at a representative neighbour:

- Handling accidents
- Fire accidents
- Drilling and excavation
- Aircraft accidents and meteorite impact
- Leakage of contaminated drain water from ASR to recipient

8.4 Quantitative analysis

This section covers the quantitative analysis of the accidents concluded in section 8.3.11 to have the potential to cause at least 1 mSv dose at a neighbour. For each accident the frequency of initiating events and the probability function for the dose to the neighbour has been estimated.

The total risk to the representative neighbour is estimated using a Bayesian tool. This tool combines the quantitative and probabilistic dose functions for the individual types of accident to an overall dose function for the facility.

The risk related to accidents has been quantified for the years 10, 30, 100, 300, 1000, 3000, and 10,000 related to the opening of the repository (year 10 is considered the opening year). The risks from handling accidents and fires are considered for year 10 when filling the waste in the repository, and year 30 as an example of a year of removing the waste from the repository. At year 300 it is assumed that the repository has been forgotten.

8.4.1 Bayesian model

A Bayesian model is set up for calculation of the overall dose of activity received by a representative neighbour due to accidental events at the repository.

The Bayesian model is capable of generating dose functions for a large number of scenarios varying type of repository, depth of repository, operational mode, time, backfill, fill in waste units and soil.

Accidents of the following types are modelled:

- Handling accident
- Mechanical damage to the repository
- Fire
- Natural hazards
- Chemical reaction
- Intrusion by living organisms
- Malicious damage and war.

Main net

The net "Main" includes a node for each of the parameters determining the scenario, i.e. "Repository", "Depth of repository", "Geology", "Operational mode", "Time", "Backfill", and "Fill".

These "input" nodes influence the hazards. When these nodes represent a number of different hazard scenarios, these are represented by a sub net, e.g. "Handling" and "Aircraft" The sub nets will calculate the total dose probability function for the accidents of the type in question and return this to "Main".

"Main" then combines all dose probability functions to one overall dose probability function.

"Main" analyses the dose functions received from the sub nets and calculates the probability of receiving a dose of 1 mSv or more. "Main" also calculates this parameter for the individual types of accidents.

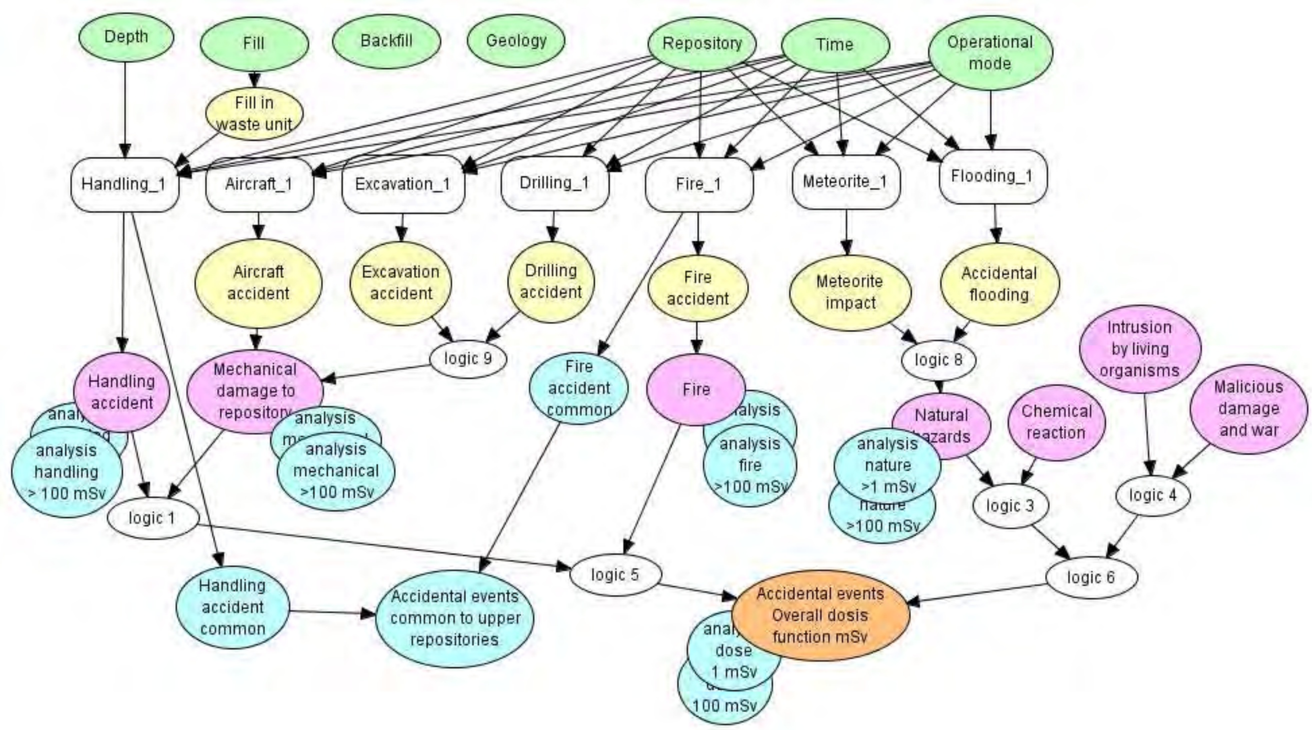

Figure 8.11 Bayesian model, main net "Main"

Sub nets

Sub nets are prepared for each type of accident. The sub nets calculate the probability function for the dose received by the neighbour and returns the function to "Main".

The layout of the sub net's are determined by the type and complexity of the hazards. The hazards are described in the sections below. These sections also describe the @RISK simulations prepared to generate the dose functions for each specific hazard.

Handling

The sub net "Handling" includes nodes for the specific handling accidents capable of exposing the neighbour to doses above 1 mSv.

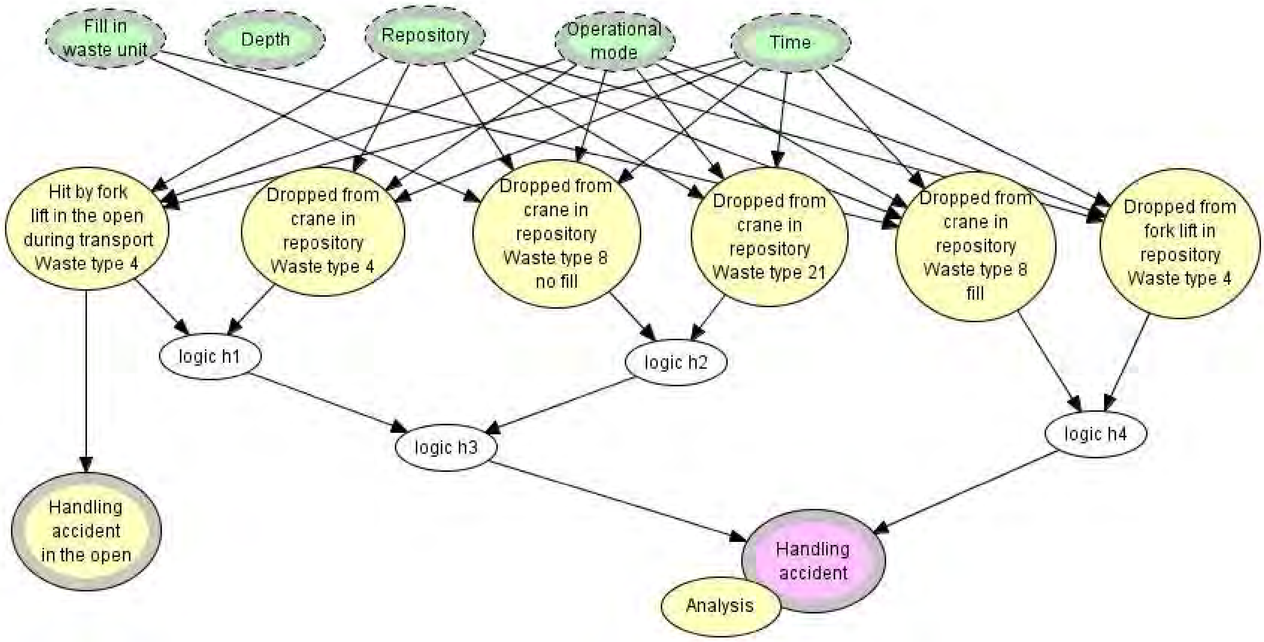

Figure 8.12 Bayesian model, Sub net "Handling"

Excavation

The sub net "Excavation" includes a single node for the hazard "excavation" capable of exposing the neighbour to doses above 1 mSv.

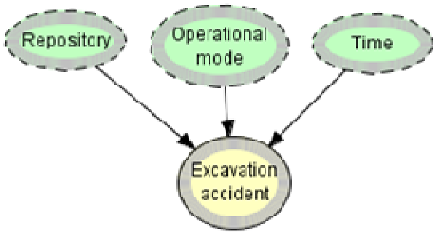

Figure 8.13 Bayesian model, Sub net "Excavation"

Aircraft

The sub net "Aircraft" includes nodes for the specific accidents with aircraft capable of exposing the neighbour to doses above 1 mSv.

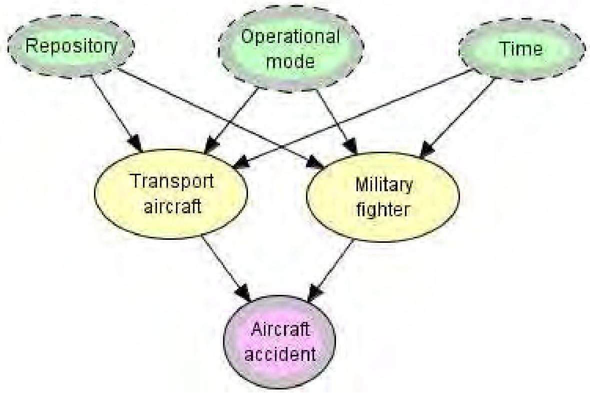

Figure 8.14 Bayesian model, Sub net "Aircraft"

Fire

The sub net "Fire" includes nodes for the specific fire accidents capable of exposing the neighbour to doses above 1 mSv.

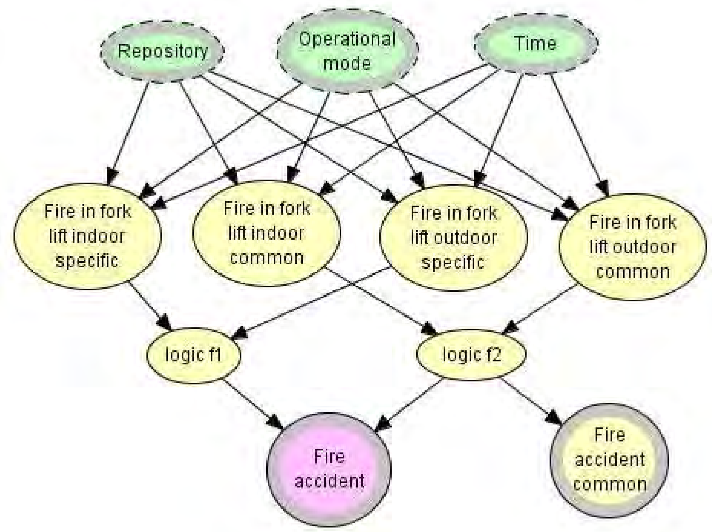

Figure 8.15 Bayesian model, Sub net "Fire"

8.4.2 Dose functions

The Bayesian model requires a list of probability of exposure versus dose for each accidental event, i.e. a dose function. The overall span of dose intervals are derived from the dose criteria indicated previously in this chapter. Minimum is 0 mSv, maximum is >100 mSv. To facilitate speed of calculation the number and span of individual intervals are detailed considering the dose levels relevant. An example of a dose probability function is presented in Table 8.6.

Table 8.6 Example of input for Bayesian model. Probability of exposure versus dose

Dose interval (mSv/year)	Probability of exposure
0	0.999968
0 - 1	2.77E-05
1 - 2	2.98E-06
2 - 3	1.1E-06
3 - 4	9.36E-09
4 - 100 mSv	0
> 100 mSv	0

The hazard screening aimed at identifying accidental events able to expose neighbours to a dose >1mSv. Thus the Bayesian model only considers these events. This means that although the probability of receiving an overall dose below 1 mSv is calculated by the model thise probability does not include all accidents capable of exposing the neighbour to these low doses.

The Bayesian model considers a period of one year. The model requires the overall probability of the dose "0 mSv" for each accidental event. This dose may occur:

a) Because the accidental event does not take place (within the one year) and

b) Because the accidental event takes a course not resulting in exposure of the neighbour.

a) is derived from the frequency of the initiating event, e.g. a waste unit is dropped. b) is derived from the conditional probabilities of exposure given occurrence of the accidental event.

Frequency estimates

The frequency estimation includes estimating the frequency of occurrence of the initiating events and the probability that they will develop into a release of waste that may cause a dose of at least 1 mSv at a neighbour. The estimates are made as best estimate. In general there is large uncertainty related with the frequency estimation, at least a factor of 10 must be expected.

The frequency estimates are based on:

- Available statistics
- Fault tree analysis
- Engineering judgement.

Dose probability calculation

The probability function of the dose given occurrence of the accidental event is in general generated by Monte Carlo simulation using the EXCEL add-in @RISK. The simulation considers the variation in conditions and development of the accidental event, e.g. weather conditions, drop height, damage to waste unit. The programme @RISK provides detailed list of results from which the probability function on the intervals used in the Bayesian model are generated.

For the same accident the probability function may vary. The main parameters causing variation are the type of repository, operational mode and the moment of occurrence. However, also parameters like soil type, depth of repository and packaging may have an impact.

8.4.3 Handling accidents

Frequency of initiating events

It is assumed that an accident involving a fork lift happen with a probability of $3 \cdot 10^{-5}$ per move of one waste unit by forklift. The frequency of a waste unit being hit by a forklift is estimated at $1 \cdot 10^{-4}$ per move of waste unit. The frequencies are based on forklift operations statistics at a Danish production company (COWI Rådgivende Ingeniører, 1986). Drop of waste unit from a crane is assumed to occur with frequency 7.8 per 1 mil hours of operation (F P Lees, 1980). This may be conservative since the number in the reference relates to crane failure of any kind. Summary of frequencies of handling accidents used in the calculation are given in Table 8.7.

Table 8.7 Frequency of dropped waste item

Accident	Frequency
Drop from fork lift	$3 \cdot 10^{-5}$ per move of waste unit
Drop from crane	$7.8 \cdot 10^{-6}$ per hour of operation
Hit by fork lift	$1 \cdot 10^{-4}$ per move of waste unit

Dose calculation

The parameters used in the dose probability calculation are shown in Table 8.8.

Table 8.8 Parameters in dose calculation and how the uncertainty is represented in the calculations

Parameter	**Representation in dose calculation**
Nuclides in waste type	Best estimate of activity, see Appendix A
Nuclides in individual waste unit	Conservative assumptions collecting the most active waste in few units
Fraction of dust in waste unit	When precise conditioning of waste is not known a large fraction of dust is assumed
Weather conditions	@RISK simulation with representation of stability classes D and F and wind speed 0.5, 2, 5 and 10 m/s
Drop height	@RISK simulation using triangular distribution covering relevant intervals (very large drop heights for cranes in some repositories)
Damage to waste unit condition to drop height	@RISK simulation using binomial distribution.
Fraction of dust released from waste unit when damaged	@RISK simulation using triangular distribution.
Dose reduction factor due to retention of dust in repository shaft or building	@RISK simulation using triangular distribution.

Figure 8.16 and Figure 8.17 are examples of simulation results for handling accidents.

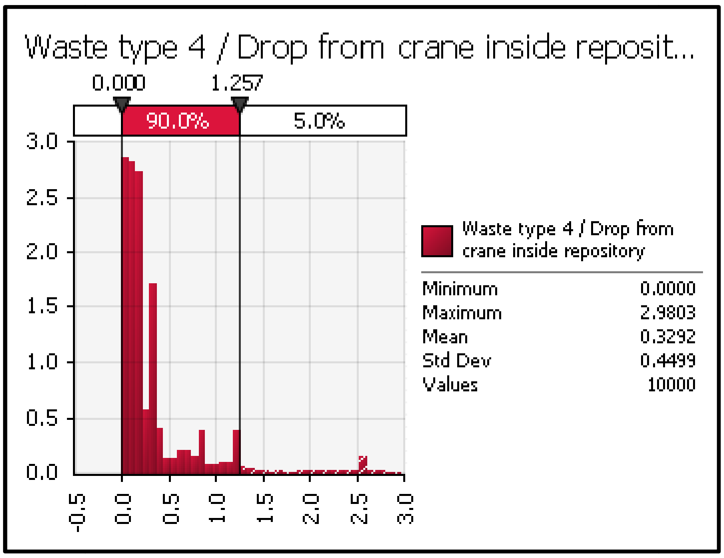

Figure 8.16 Simulation results of dose received by neighbour due to drop of waste unit of type 4 from crane inside a repository of type 3.

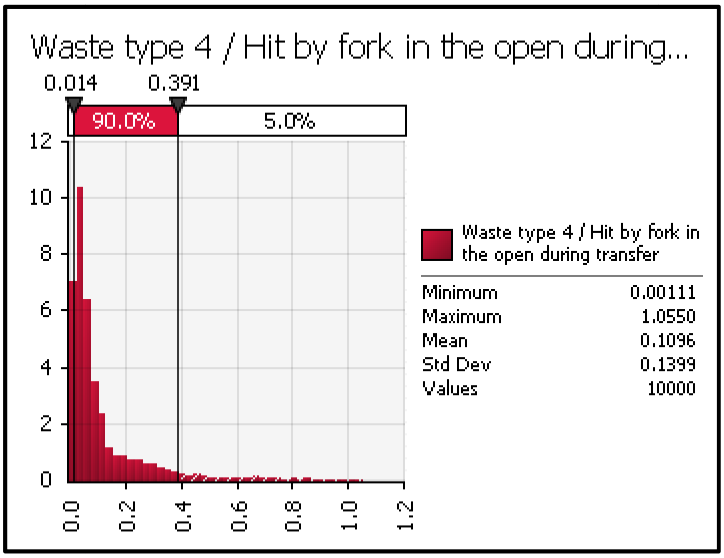

Figure 8.17 Simulation results of dose received by neighbour due to damage to waste unit of type 4 during transportation by fork lift in the open

Table 8.9 is an example of the probability functions simulated for a waste unit of type 4 dropped from a crane inside the repository.

Table 8.9 Probability functions (per year) simulated for drop of a waste unit of type 4 from a crane inside the repository

mSv	REP1*)	REP2**)	REP3 and REP6	REP4 and REP7	REP5 and REP8	REP9	REP10*)
0	1	1	0.999963	0.999956	0.999954	0.999971	1
0 - 1	0	0	3.33E-05	3.82E-05	4.22E-05	2.79E-05	0
1 - 2	0	0	2.49E-06	4.25E-06	2.72E-06	1.07E-06	0
2 - 3	0	0	9.87E-07	1.73E-06	7.25E-07	1.31E-07	0
3 - 4	0	0	0	0	0	0	0
4 - 5	0	0	0	0	0	0	0
5-100	0	0	0	0	0	0	0
> 100	0	0	0	0	0	0	0

*) No crane

**) Functions are only reported when doses above 1 mSv are possible

8.4.4 Fire accidents

Frequency of initiating events

To support the calculation a safety barrier diagram for the situation of a significant truck fire leading to a dose of 1 mSv to the neighbour has been constructed and is shown in Figure 8.18.

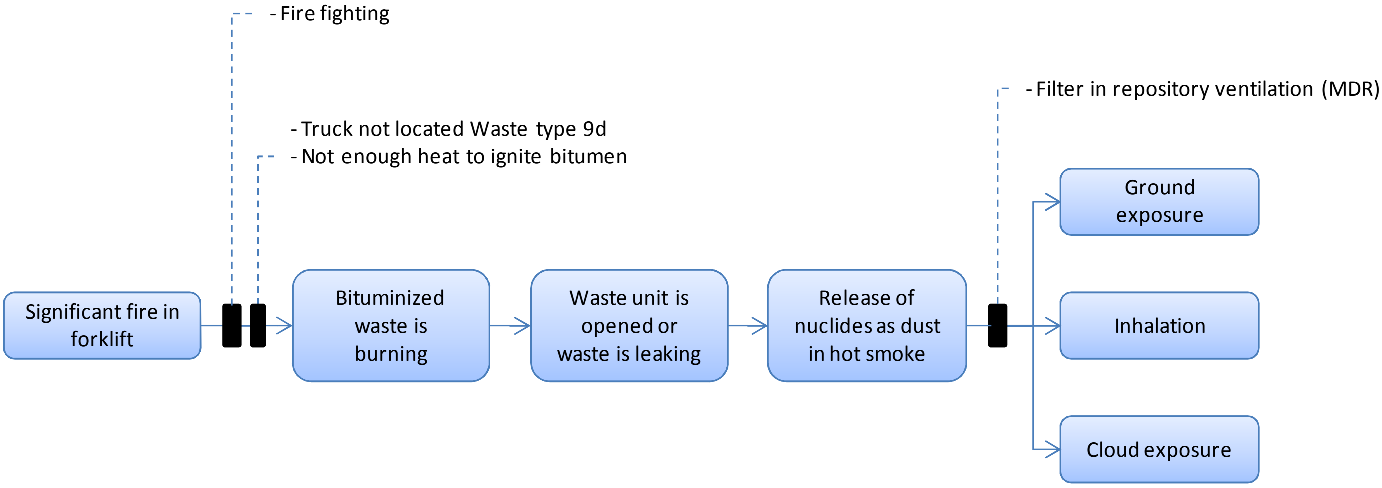

Figure 8.18 Safety barrier diagram for significant fire in forklift

The frequency of forklift fires has been evaluated from the frequency of fires in road vehicles in Denmark, since no statistics for fires in forklifts were found. Only significant fires, defined as those were 2 nozzles were used by the fire fighters, are counted, since small fires will probably not be able to heat the bitumen sufficiently to make a large fire. The calculation and the assumptions used are shown in Table 8.10. For ASR and NSR the frequency the operations are carried out above surface, while for a cavern MDR (COFI) they will be carried out twice, both above surface and inside the repository. For a cavern MDR the double frequency is therefore assumed during the loading.

Table 8.10 Calculation of forklift fires per year

Fire on Danish roads 2007-2009 per vehicle-km (Redningsberedskabet, 2011), (Vejdirektoratets, 2011), (Danmarks Statistik, 2011)	$4.0 \cdot 10^{-9}$
Fire on Danish roads 2007-2009 per vehicle-hour assuming 50 km per hour	$2.0 \cdot 10^{-7}$
Forklift hours per day	8
Forklift fires/vehicle fires	2
Workdays per year	246
Forklift fires per year anywhere inside the repository or during operation at the intermediate storage	$7.9 \cdot 10^{-4}$

Based on Lees, F.P. (2005) a probability for failure of fire fighting is set to 0.1, considering that personnel is likely to be trained in the fire fighting equipment, but on the other hand may be under stress in the situation.

The probability that the truck is located near the waste of type 9d is roughly evaluated to 5% based on volumes of waste type 9d and the total waste. This is conservatively assuming that the forklift when operating at all times is located next to at least one waste unit.

The probability that there is not enough heat to liquefy and ignite the bitumen is set to 50 % for the ASR and NSR, and 10 % for MDR's, since the heat build up is expected to be higher inside the MDR.

The resulting frequency fires in bitumen per year are shown in Table 8.11.

Table 8.11 Frequency of fire in bitumen containing radionuclides.

Repository type	Frequency of bitumen fire per year during operation of repository
REP1-REP9	$2.0 \cdot 10^{-6}$
REP10	$3.6 \cdot 10^{-6}$

Dose calculation

The parameters used in the dose probability calculation are shown in Table 8.12.

Table 8.12 Parameters in dose calculation for fire and how the uncertainty is represented in the calculations

Parameter	Representation in dose calculation
Nuclides in individual waste unit	Conservative assumptions collecting the most active waste in few units
Fraction of dust in waste	When precise conditioning of waste is not known a large fraction of dust is assumed
Specific waste units hit and damaged during fire	The fraction of waste able to give high doses.
Amount of radionuclides retained inside repository (only relevant for REP9)	Conservatively no retention in cavern or filters is assumed
Weather conditions	@RISK simulation with representation of stability classes D and F and wind speed 0.5, 2, 5 and 10 m/s

Figure 8.19 to Figure 8.21 show the simulation results for a fire in a forklift engulfing drums containing waste type 9.

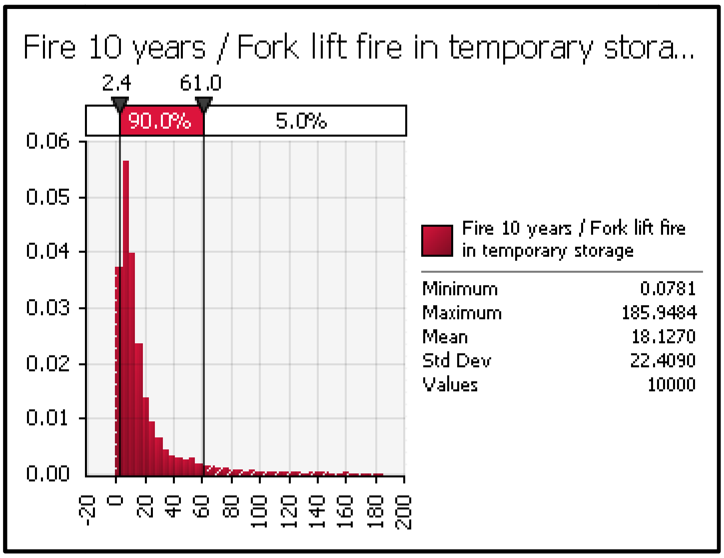

Figure 8.19 Simulation results of dose received by neighbour due to fire in fork lift in the open engulfing drums of waste type 9. Results are valid for the time 10 years.

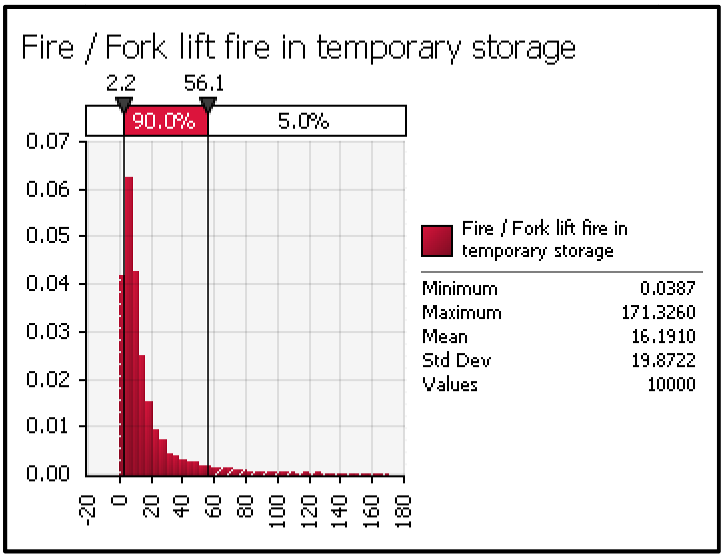

Figure 8.20 Simulation results of dose received by neighbour due to fire in fork lift in the open engulfing drums of waste type 9. Results are valid for the time 30 years.

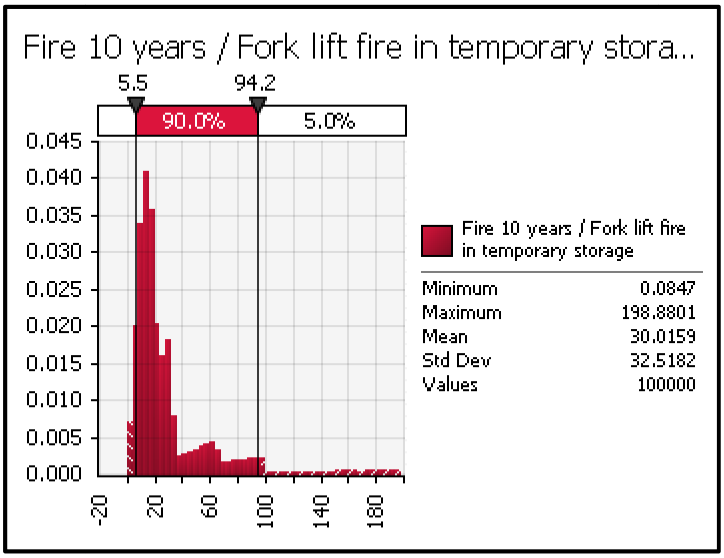

Figure 8.21 Simulation results of dose received by neighbour due to fire in fork lift inside temporary storage or repository engulfing drums of waste type 9. Results are valid for the time 10 years

8.4.5 Drillling and excavation

Frequency of initiating events

Calculation of the frequency of excavation activities in Denmark that will impact the repository is based on assumptions of how much excavation activity that will be done in the future. This is obviously associated with large uncertainty when looking 10,000 years ahead.

As shown in Table 8.13 the fraction of Denmark that was used as urban area was approximately 10 %. For the calculations it is assumed that in year 12000 this fraction is 30 %. This means that 20 % of the area of Denmark will be exposed to excavation activities of one kind or another during the period considered. Each excavation is assumed to be of a size of 1,000 m^2, and if this area overlaps with the repository it is assumed that radionuclides are released, also if it is not a full overlap.

Table 8.13 Development of urbanization in Denmark

Year	Fraction of Danish urban area
2000	App. 10 %
12000 guess	30 %

Calculation of the frequency of excavations damaging the repository is further based on the following assumptions and estimates:

- Excavation activity will conservatively always reach a depth where waste is located (notice that excavation is only considered for above surface and near surface repositories).

COWI

The resulting frequencies are shown in Table 8.14.

Table 8.14 Frequency of release of nuclides from repository caused by excavation.

Years from construction of repository	Repository type	Frequency of excavation on repository site per year
300	REP1	$1.2 \cdot 10^{-4}$
1,000-10,000	REP1	$1.2 \cdot 10^{-4}$
300	REP2	$7.1 \cdot 10^{-5}$
1,000-10,000	REP2	$7.1 \cdot 10^{-5}$

Figure 8.22 is an example of simulation results for excavation of the repository area.

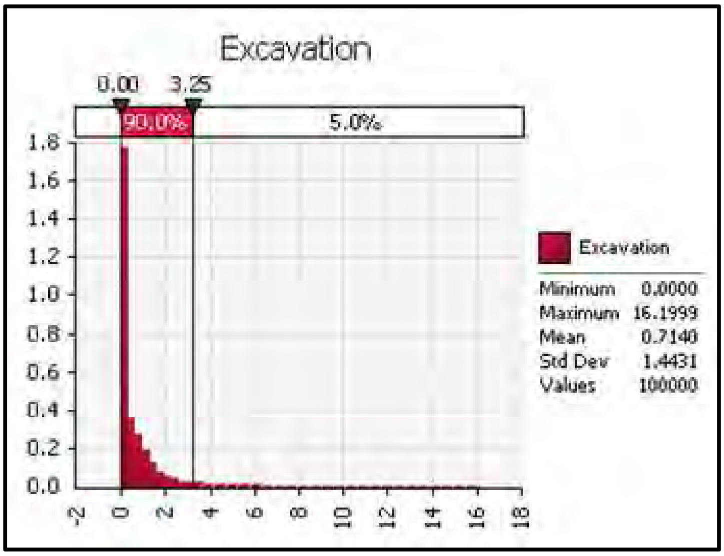

Figure 8.22 Simulation results of dose received by neighbour due to excavation of above surface or near surface repository after 300 years

Drilling activities

Calculation of the frequency of excavation activities in Denmark that will impact the repository is based on assumptions of how much drilling activity that will be done in the future. Although the repository shall be located in an area not classified as a potential water catchment area, in the future, say a thousand years from now, the status of the area in which the repository is located may change.

The calculations are based on the following assumptions:

- 100 groundwater exploration drilling activities per year
- The fraction of Denmark's area relevant for groundwater drilling is 0.9
- The drilling angle is 70 degree.

In Table 8.15 the parameters for calculating the cross sectional area of the canisters content is shown together with the resulting effective area given that the canisters have been placed in a borehole. The same effective are has been assumed should the canisters be located in a medium deep repository.

In Table 8.16 the values and assumptions used for calculating the frequency of exploration drilling are given together with the resulting frequencies.

Table 8.15 Calculation of effective cross section of stuck of canisters

Number of canisters	Canister inner height, m	Inner diameter of canister, m	Drilling angle ,º	Effective cross-section, m^2
78	0.8	0.3	70	6.8

Table 8.16 Frequency of drilling through the canister

Estimated number of borings in Denmark per year	Fraction of area of Denmark relevant for drilling	Area of Denmark, km^2	Probability of drilling through canister per boring	Frequency of drilling through canister per year
100	0.9	43,094	$1.8 \cdot 10^{-10}$	$1.8 \cdot 10^{-8}$

Dose calculation

The parameters used in the dose probability calculation are shown in Table 8.17.

Table 8.17 Parameters in dose calculation for an excavation event and how the uncertainty is represented in the calculations

Parameter	Representation in dose calculation
Nuclides in waste type	Best estimate of activity prepared
Nuclides in individual waste unit	Conservative assumptions collecting the most active waste in few units
Fraction of dust in waste	When precise conditioning of waste is not known a large fraction of dust is assumed
Specific waste units hit and damaged during excavation	Conservative large fraction of waste is assumed able to give high doses.
Weather conditions	@RISK simulation with representation of stability classes D and F and wind speed 0.5, 2, 5 and 10 m/s
Excavation process continues without observation of danger	@RISK simulation using binomial distribution.

8.4.6 Aircraft and meteorite impact

Frequency of initiating events

Aircraft crash frequency for an above surface and a near surface repository has been calculated based on accident statistics on aircraft crashes in Denmark. The results are shown in Table 8.18.

For the calculation of the frequency of aircraft accidents opening the repository for runoff it has been assumed that

- The scenario considered is an aircraft sliding towards the side of the repository and removing the protecting soil on the side of the repository
- This scenario is only relevant for the commercial aircrafts. Military fighter crashes will be more likely to crash at an angle closer to 45 degree creating a crater rather than an opening of the side
- The crater from the aircraft must have a 10 m overlap with the repository.

Table 8.18 Aircraft crash frequencies at the repositories

		Frequency of aircraft crash at repository (per year)	Frequency of opening sides of repository enabling runoff (per year)
Above surface repository	Transport aircraft	$7.0 \cdot 10^{-9}$	$7.0 \cdot 10^{-10}$
	Military fighters	$3.9 \cdot 10^{-8}$	N/A
	Total	$\mathbf{4.6 \cdot 10^{-8}}$	$\mathbf{7.0 \cdot 10^{-10}}$
Near surface repository	Transport aircraft	$4.7 \cdot 10^{-9}$	N/A
	Military fighters	$2.0 \cdot 10^{-8}$	N/A
	Total	$\mathbf{2.5 \cdot 10^{-8}}$	**N/A**

Meteorites

The meteorite impacts considered in the analysis are those large enough to damage an above or near surface repository and small enough that the consequences to the neighbour from the impact itself is not larger than from the release of radionuclides form the repository. Based on the information in Figure 8.23 showing meteorite impact frequencies, and references (Miguel Carrasco/La Razon/Reuters, 2010), (Collins, G. et al, 2005), (Marcus, R. et al, 2010) these limits have been roughly determined and the impact effect in terms of crater size has been roughly estimated. The results are shown in Table 8.19.

Table 8.19 Meteorite impact frequencies and effect

	Small	Medium	Large
Impactor size, m	1	3	10
Number of meteorite impacts on earth, per year	8720	365	1
Crater diameter*, m	3	15	100
Crater depth* , m	3	5	20

*Transient crater dimensions, reflecting the volume of soil/repository affected

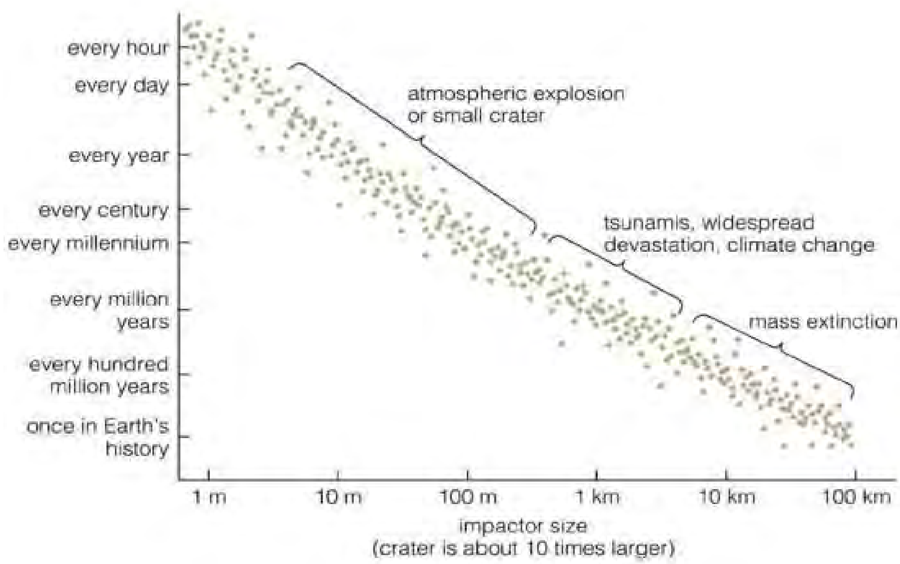

Figure 8.23 Frequency of meteorite impactor size (Dr. Nicholas Short, 2010)

The frequency of meteorites within the upper and lower size limit set for the present analysis have been calculated assuming an evenly distributed density of meteorite crashes on the Earths surface. The results are given in Table 8.20. It may be seen from the table, that the frequency of this occurring is quite remote.

Table 8.20 Frequency of meteorite impact at ASR, REP1 and NSR, REP2

	Area, km^2	Fraction of Earth's surface area	Impact frequency per year		
			Small	Medium	Large
Earth	$5.1 \cdot 10^{8}$	1	8,760	365	1
REP1	$4.7 \cdot 10^{-3}$	$9.2 \cdot 10^{-12}$	$8.1 \cdot 10^{-8}$	$3.4 \cdot 10^{-9}$	$9.2 \cdot 10^{-12}$
REP2	$2.5 \cdot 10^{-3}$	$4.8 \cdot 10^{-12}$	$4.2 \cdot 10^{-8}$	$1.8 \cdot 10^{-9}$	$4.8 \cdot 10^{-12}$

Dose calculation

An impact on the repository may reach any kind of waste, and large amounts are assumed to be dispersed. Due to the relatively low accident frequencies, it has conservatively been assumed that dispersion of dust caused by a meteorite or an aircraft impacts will result in a dose of at least 100 mSv at a neighbour. In relation to aircraft crashes, any surviving passengers, and if the existence repository has been forgotten (after 300 years) also the rescue workers, will additionally receive a significant dose. Additionally, runoff from the side of the repository may cause a dose to a neighbour, however since the maximum dose of >100 mSv has been assumed, no additional contribution is considered.

8.4.7 Leakage of contaminated drain water from repository

These scenarios are only relevant for the above surface repository, REP1.

Frequency of initiating events

The initiating events considered are

- Repository is damaged during construction and drain system is not working
- Repository is damaged curing construction, drain system is working, but the drain tank is emptied erroneously to a recipient.

Drain system

Water may enter the repository before expected, if the repository top membrane was damaged during construction, or if it deteriorates significantly faster than expected. If there is a hole in the bottom membrane, or if the drain system is not working properly, contaminated water will either flow to a recipient or seep into the ground (or both), depending on the geology beneath the repository. The release is set to 4.7 m^3 per year, and is conservatively assumed not to be detected once started, since the repository may be abandoned, before the release is detected.

Table 8.21 Frequency of continuous release due to damage to top membrane

	Time (years)			
	10	**30**	**100**	**300**
Water to recipient due to drainage error	0	$1.7 \cdot 10^{-6}$	$2.8 \cdot 10^{-5}$	$4.8 \cdot 10^{-4}$

Erroneous operation

If the repository top membrane is damaged during construction and water may start to show up e.g. after 20 years in the drainage from inside the repository. Due to several errors the water can erroneously be led directly to a recipient. It is for this incident assumed that most likely 1 m^3 is led to the recipient, corresponding to the amount for the normal emptying of the tank. More seldom, conservatively set to a 25% probability, one full year volume of 4.7 m^3 is led to the recipient. The frequency of 1 or 4.7 m^3 led to recipient is analysed by the fault tree shown in Appendix G. The frequency of potential erroneous discharge is shown in Table 8.22. It can be seen from the table that this scenario is estimated to occur with a very low frequency and only for the limited period of time, while the repository is under operation.

Given the low amounts of contaminated water (one time release, not continuous release) and the low frequency, this scenario is not considered further.

Table 8.22 Frequency of erroneous discharge following

	Time			
	10	**30**	**100**	**300**
Discharge of contaminated water to recipient	0	$7.6 \cdot 10^{-7}$	$1.9 \cdot 10^{-5}$	$8.4 \cdot 10^{-6}$

Dose calculation

The dose calculation for drainage errors in combination with water seeping through the top membrane is carried out with the same models as for long term consequences, except that a volume of 4.7 m^3 is used, and the release is expected to occur at an earlier time of the repository lifetime.

8.5 Risk results at repository level

This section represents the aggregated risk results for accidents calculated by the Bayesian model. The results are for each repository presented as the frequency of accidents leading to a dose to a neighbour as a function of time. The results are shown in the following sections. The years indicated are years since 2008.

8.5.1 Above surface repository

The risk from accidents is shown as the frequency of events giving a dose of a given size or higher (FN type diagram) in Figure 8.24. Only accidents leading to doses at or above the criterion of 1 mSv are shown.

From the figure it may be seen that the frequency of accidents, where the dose to a neighbour exceeds the limit of 1 mSv, is less than 10^{-4} per year. The dotted curves represent handling accidents and fires, i.e. accidents during the filling of the repository or removal of the waste. For these activities, the probability of exceeding the criterion of 1 mSv dose is estimated at approximately 10^{-6} per year. Assuming that each operation takes 1 year in total, it corresponds to that the probability that a neighbour will receive a dose of 1 mSv or more is one in a million for the filling of the repository, and likewise for emptying the repository.

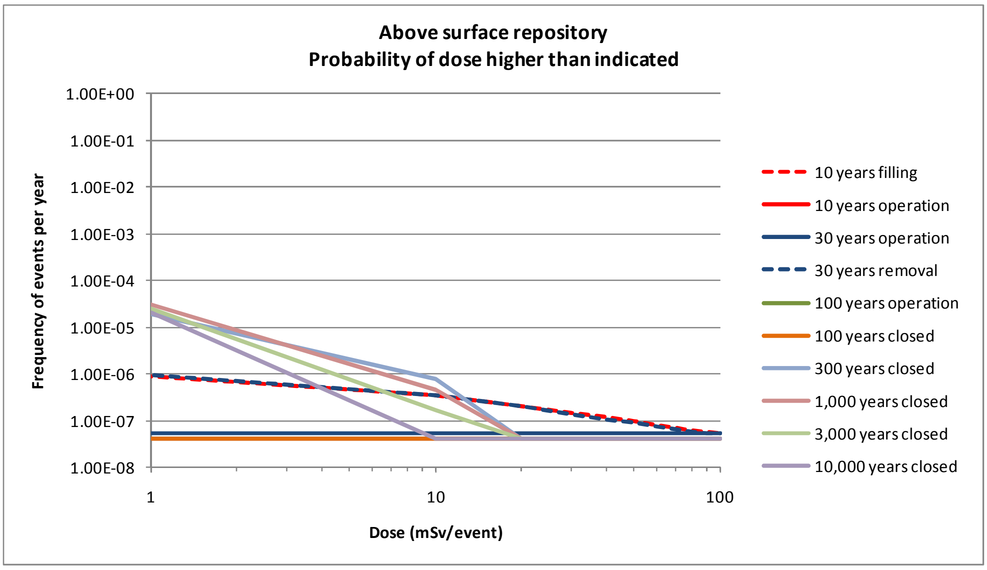

Figure 8.24 Frequency-dose diagram short term exposure events for above surface repository.

The accidents related to contaminated water leaking from the repository due to problems with drainage of water accidentally passing into the repository, e.g. due to a construction error, will not be an event based dose, but will continue for a longer period of time. These types of accidents should be compared with the dose criterion of 0.01 mSv/year, i.e. not the 1 mSv/event used for instantaneous releases. In Figure 8.25 and Figure 8.26 the results for accidental events leading to long term exposure are shown for an ASR placed on granite and clay till respectively. The figures show the dose that a neighbour receives from different radionuclides as a function of time, assuming a release at year 10. Should the release occur at a later point in time, the curves would be unchanged, although cut off at the year of the water entering the repository. The delay in dose for some of the radionuclides is related to the deterioration of the waste unit (e.g. drum or container). This deterioration is assumed to happen regardless of whether there are holes in the top membrane. A dramatic difference is seen between placing an ASR on top of granite (Figure 8.25) compared to clay till (Figure 8.26) in favour of the clay till. For granite the dose limit of 0.01 mSv/year is severely exceeded, should the incident occur. This is because the contaminated water, when reaching the granite, most likely will move horizontally until meeting a recipient, whereas in clay till, the water will pass vertically through the ground resulting in a significantly longer time span, before radionuclides will enter a recipient or a groundwater reservoir. Results for fat clay and another type of clay till geology are shown in Appendix H.

It may be recalled from section 8.3 that the probability of these scenarios occurring is fairly low, especially in the early years of the repository lifetime. The probabilities are for convenience repeated in Table 8.23

Table 8.23 Frequency of continuous release due to damage to top membrane

	Time			
	10	30	100	300
Water to recipient due to drainage error	0	$1.9 \cdot 10^{-6}$	$1.7 \cdot 10^{-4}$	$5.3 \cdot 10^{-3}$

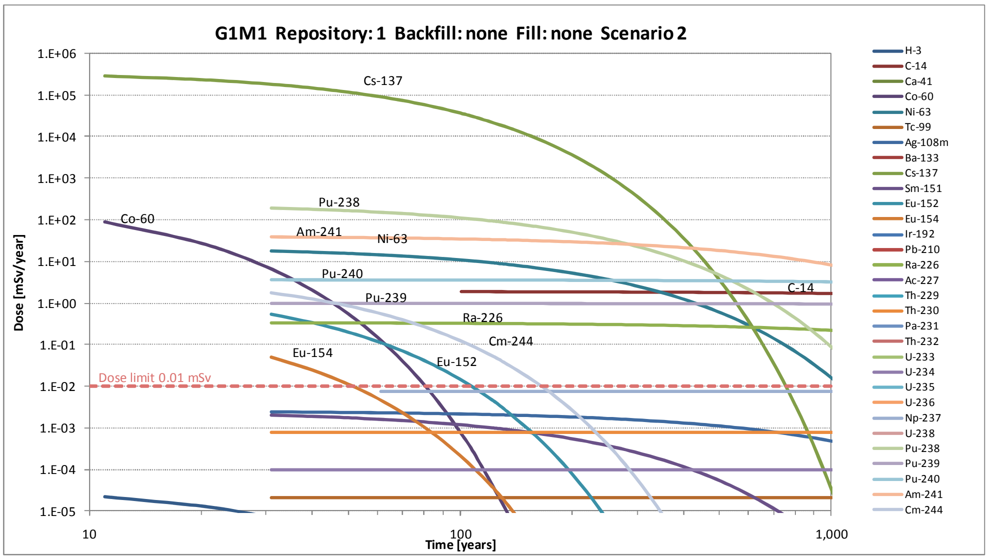

Figure 8.25 Frequency-dose diagram for events with long term effects, ASR placed on granite.

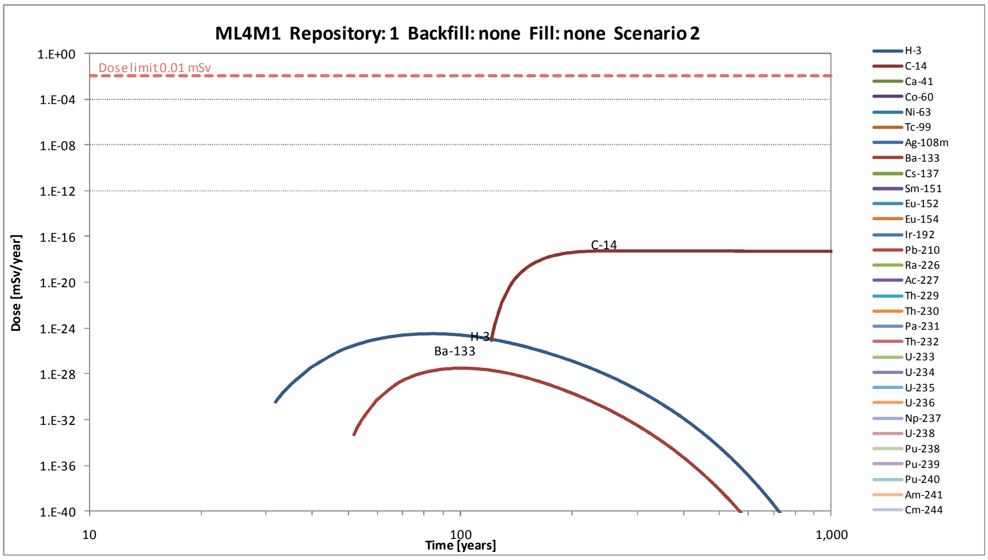

Figure 8.26 Frequency-dose diagram for events with long term effects, ASR placed on clay till.

8.5.2 Near surface repository

The risk from accidents is shown as the frequency of events giving a dose of a given size or more (FN type diagram) in Figure 8.27. Only events leading to doses at or above the criterion of 1 mSv are shown. It may be seen by comparison with Figure 8.24 that the risk is much the same as for the above surface repository. This is because the same probability of damage from aircraft accidents, meteorite impacts, and excavation in a near surface repository as a above surface repository has been assumed, which is conservative. The near surface repository may be slightly better protected by the concrete slab on the top of the repository.

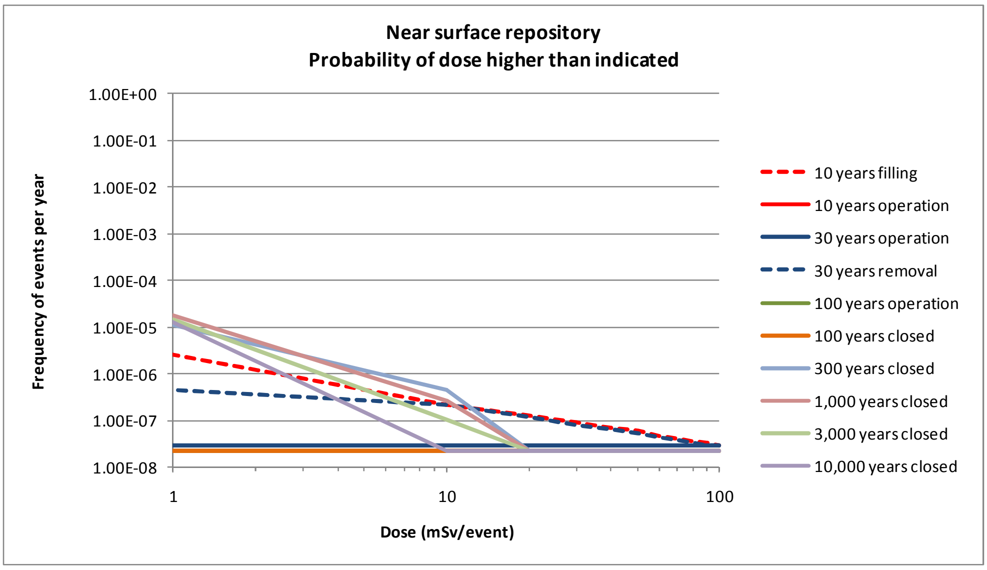

Figure 8.27 Frequency-dose diagram for near surface repository.

8.5.3 Medium deep repository

The risk from accidents is shown as the frequency of events giving a dose of a given size or more (FN type diagram) in Figure 8.28 - Figure 8.30 for the different types of medium deep repository. Only events leading to doses at or above the criterion of 1 mSv are shown. It may be observed that only accidents relating to filling in and removing of waste from the repository were identified. These accidents are handling accidents and fire accidents. The risk is higher than the risk related to handling and fire accidents for the above and near surface repositories. This is due to higher drop heights for the repositories operated from above.

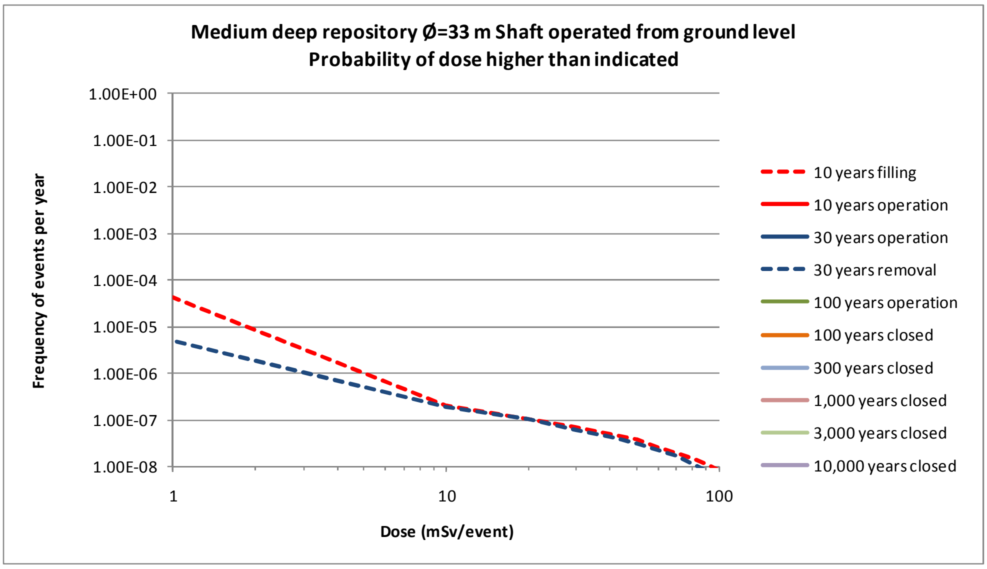

Figure 8.28 Frequency-dose diagram for medium deep repository (diameter 33 m) shaft operated from ground level.

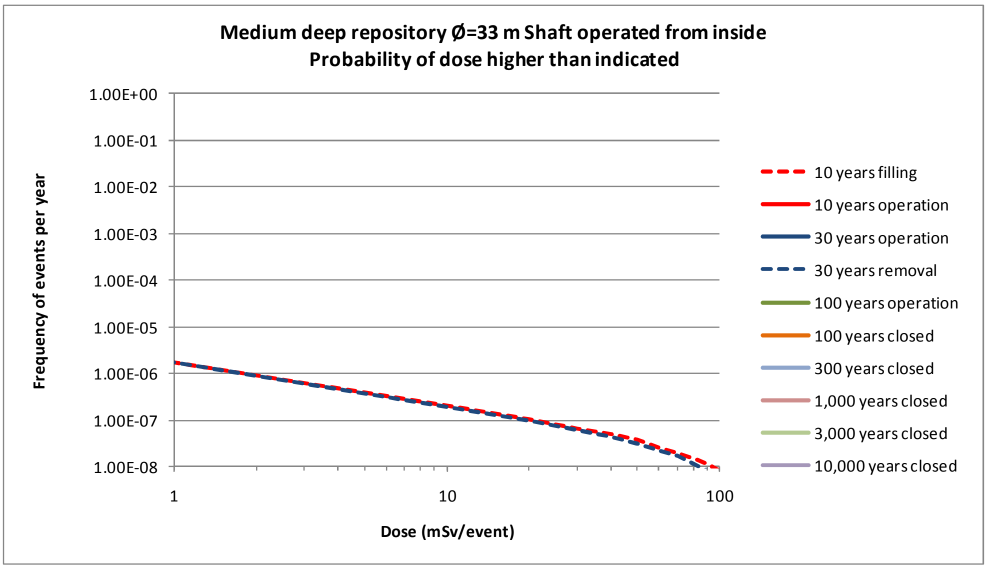

Figure 8.29 Frequency-dose diagram for medium deep repository (diameter 33 m) shaft operated from inside.

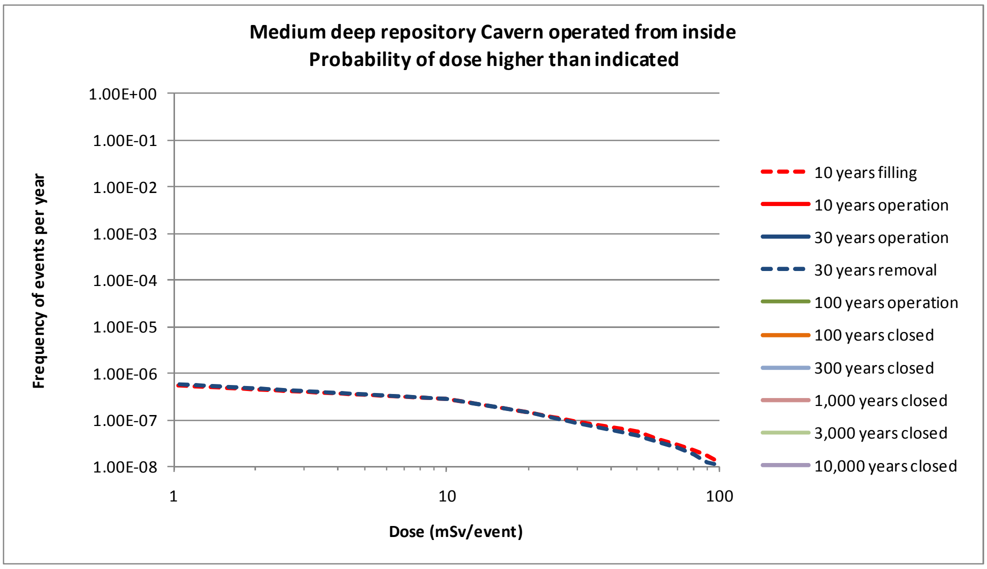

Figure 8.30 Frequency-dose diagram for medium deep cavern operated from inside.

It can be seen from the figures for the different types of medium deep repositories that the risk of impact on a neighbour is slightly higher for a repository operated from above, due to the greater possibility for spreading of dust to the neighbour from this type of repository.

8.5.4 Borehole

The risk related to accidents to waste stored in canisters in a borehole was found to be insignificant.

9 Consequences for conditioning

In this chapter, all results with respect necessary conditioning of the different waste types based on the preliminary safety assessment have been compiled. This includes both issues relating to possible consequences of possible accidents occurring at the repository (based on the safety assessment related to accidents) and possible consequences relating to the long term impact from a repository. In general, there may be differences due to the type of repository for several reasons:

- Placing of the special waste in a near surface repository will typically require extra conditioning of the waste to ensure the retention of the nuclides for sufficient time, and to ensure that accidents will not release them in unsafe amounts to the surroundings.

- Placing of the waste in a borehole requires very special conditioning, so when a borehole is part of the overall solution, such conditioning is required for the waste types in question.

Apart from that, the difference between repositories is mainly due to difference in geological setting and closeness to groundwater sources. Based on the general principle of optimisation required by ICRP and due to the generic nature of the prefeasibility study, it has been chosen not to differentiate further in the suggestions in this chapter for conditioning for different repository types.

In general, it should be avoided to repack already packed waste, if the packaging is sufficient based on the preliminary safety assessment, since repacking will always generate additional doses to the personnel involved and may also generate additional secondary waste. Repackaging may be necessary, either due to the condition of the waste package rendering it dangerous to transport and handling in general, or if supplementary conditioning is necessary to reduce long term impact from the waste or impact related to potential accidents. In that case, it is suggested to apply the method already used by Danish decommissioning and place the package in an outer package. The suggestions below are based on this principle.

9.1 Packing of individual waste types

The waste types listed below are further described in Chapter 2 and in Appendix A.

9.1.1 Waste from DR1, DR2 and DR3

Type 1 - Graphite

Due to risk of superheating of the graphite, in case of fire during disposal or possible later removal of the waste, it is recommended that the graphite items can be packed in steel containers that are filled with bentonite around the graphite, when the containers are filled. Alternatively, the graphite bars may be surrounded by bentonite in the containers. For the same reasons, once the containers have been placed in the repository, bentonite (in case of a reversible repository) or concrete should be backfilled around the containers.

Type 2 - Aluminium

It is suggested that the rest of the aluminium waste will also be packed in ISO containers. Ten ISO containers are needed in total. As mentioned in Chapter 3, it is recommended to use sand as fill material in between the waste items in the containers, in order to reduce corrosion of the aluminium and thereby limit gas formation.

By processing (melting for volume reduction or free release) or by compaction of most of the material, the number of containers may be reduced by a factor two or more. However, this option is not considered in the pre-feasibility study.

Type 3 - Steel, stainless steel and lead

Based on the preliminary safety assessment it is suggested that the lead bricks (50-70 tons corresponding to 4.4 - 6.2 m^3) are packed in containers (11 tons lead in each) and backfilled with cement-calcium granulate due to its easy flowing properties, relatively low density, and if reversibility is requested. However, due to the high density, the containers cannot be completely filled with lead bricks. It is assumed that as a maximum, one ISO container can contain 11.3 tons of lead, if it is to be backfilled with cement-calcium granulate, and that the gross weight of the container should not exceed 21 tons. At a later stage, it may be decided to pack the heavy waste together with other waste with less density, in order to optimise the number of containers.

By processing (melting for free release), the number of containers may be reduced by a factor five or more. However, this option is not considered in the pre-feasibility study.

For the steel waste, it is recommended that remaining empty space in the containers shall be filled with cement-calcium granulate, before they are transported for final disposal. Other fill material such as bentonite or concrete may also be used depending on the need for reversibility. By processing (melting for volume reduction or free release) or by compaction of most of the material, the number of containers may be reduced by a factor three or more. However, this is not considered in the pre-feasibility study.

Type 4 - Heavy concrete and concrete

Based on the preliminary safety assessment, it is suggested that the heavy concrete is packed in steel containers, which is then filled with bentonite. This is primarily based on the risk of dust spreading related to accidents.

The larger pieces of concrete are suggested filled into ISO containers, which are then filled with cement-calcium granulate to ensure that the overall weight of the containers will not be too great and to render the waste reversible, if requested. Concrete gravel/dust from the demolition should be filled into drums with concrete lining that are placed in ISO containers, which is then filled with cement-calcium granulate. The extra conditioning of the dust is due to the potential release of dust in case of an accident.

9.1.2 Waste from Hot Cell

Type 5 - Stainless steel, steel and lead

The waste is assumed decontaminated before disposal in order to decrease the activity. The residual activity is not known at present. On the basis of the available information on the waste, it is recommended this waste type is packed in 1-2 ISO containers, which are then filled with cement-calcium granulate due to its easy flowing properties and if reversibility is requested,.

Type 6 - Concrete

Like for Type 5, the exact activity is unknown, but expected to be very low. This is to be checked before final choice of packing. Currently, it is recommended that the estimated 40 m^3 concrete waste shall be packed in ISO containers and filled with cement-calcium granulate[49]. Six ISO containers are needed.

Type 7 - Various components

Like for Type 5, the exact activity is unknown, but expected to be very low. This is to be checked before final choice of packing. Currently, it is recommended that the waste, which is made up by all kinds of waste items, shall be packed in ISO containers and filled with cement-calcium granulate. It is estimated that 3 ISO containers are needed.

Type 8 - Secondary waste

The waste consists of sand and paint dust from the decontamination (sandblasting) of the Hot Cell, and it contains PCB-paint. The waste will be put into 210 l steel drums with concrete lining. It is further recommended to fill cement-calcium granulate in between the drums after the drums have been placed in the steel containers. Once the containers have been placed in the repository, it is recommended that they are backfilled with bentonite in case of a reversible repository. In case of an irreversible repository, backfilling can be done with concrete.

9.1.3 Waste from waste water treatment

Type 9 - Waste from wastewater treatment

The waste consists of treated concentrate and salts from distillation. To a large extent, the waste has been enclosed in a bitumen-matrix and filled into 210 l concrete-lined drums. A few of the drums are leaking due to the bitumen being hygroscopic, which has caused to expand when taking up water. Since this can also occur for the rest of the drums and due to the combustibility of the waste, it is recommended that this is further investigated before final conditioning. For all drums deemed to be vulnerable, it is recommended that they are put in ISO containers that are then filled with bentonite or concrete.

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[49] See also waste type

Once the drums have been placed in the repository, it is recommended that they are backfilled with bentonite in case of a reversible repository. In case of an irreversible repository, backfilling can be done with concrete.

9.1.4 Existing compacted waste and soil

Type 10 - Compacted waste and soil

Based on the available knowledge it is suggested that the present conditioning is maintained. If the final classification results in the necessity, or if further handling at the repository may be more efficient, the drums can be placed in ISO containers before final disposal.

9.1.5 Existing waste from DR3

Type 11 - Waste from DR3

It is recommended to pack the containers with CCAs in steel containers. The number of steel containers will be determined based on handling considerations[50], and the final number may vary from 1-10. It is recommended that remaining void in containers is filled with bentonite or concrete depending on the need for reversibility.

This waste type also encompasses 40 galvanized steel drums with steel lining that contain clippings from the CCAs, which are less contaminated than the CCAs in general. It is recommended that the drums shall be placed in steel containers (six in each), which are then filled with cement-calcium granulate, bentonite or concrete.

Two waste items have been identified to be too large to be packed in containers. It is recommended that the two large items are segmented and packed in ISO containers, which are then filled with cement-calcium granulate, bentonite or concrete depending on the need for reversibility. This will ease both handling, transport and placement in the repository.

9.1.6 Existing waste from Hot Cell

Type 12 - Waste from hotcell

This waste consists of 180 drums with various waste items from the Hot Cell, 40 A-bins plus a number of items too long to be placed in drums.

The A-bins contain used fuel-elements with a dose-rate of approximately 1 Sv/h. Therefore, the bins cannot be opened and repacked in a simple way. Even when the bins are to be moved, this must take place by means of remote-controlled equipment in a so-called Hot Cell.

It is recommended that all the waste from the Hot Cell (apart from the A-bins) will be packed into steel containers prior to final disposal. This waste will require special attention due to the presence of long lived isotopes. It is recommended that remaining cavities are filled with cement-calcium granulate, bentonite or concrete depending on the need for reversibility.

[50] And the need for shielding required during handling

It is recommended that the waste in A-bins will be transferred to so-called canisters and hereafter transported for final disposal in a deep borehole. It must be emphasised that the waste in A-bins must be repacked in order to fit into canisters with assumed standard measures.

9.1.7 External waste (radiation sources)

Type 13 - Radiation sources

This waste is for a large part of external origin, including both sources and contaminated waste of a more general nature. The sources are mainly delivered from external suppliers (industry and hospitals). Some of them have been put into 18 units of 210 l concrete-lined galvanized drums. Based on the final characterisation, the waste should either be placed in A-bins and canisters or in concrete lined drums.

It is suggested that the drums is put into three steel containers (six drums in each container) filled with granulate, bentonite or concrete prior to final disposal dependent on the need for reversibility. The drums should also be filled with similar material prior to placement in the containers.

The two large sources require special attention and it is recommended that they shall be placed in a borehole due to the high activity content. A borehole is envisaged to be closed after initial filling, which excludes it as an option for future waste.

It is not foreseen that the repository will receive such large sources in the future, primarily due to producer responsibilities to take the sources back after use. If no borehole is established, it is recommended that the waste is packed in concrete lined drums that are placed in steel containers, which are both filled with concrete or bentonite if a reversible solution is requested. The actual conditioning of these future items will depend on the actual characterisation.

9.1.8 Special waste

Type 14 - Special waste - 20 large sources

The sources consist of small packages (less than 10 cm diameter) containing lead etc. and a significant amount of long-lived nuclides. It is therefore recommended that the waste shall be packed in three canisters for disposal in a deep borehole. If no borehole is established, the waste must be packed in A-bins and then steel containers, which are filled with bentonite (in case of a reversible repository) or concrete, before they are taken for disposal in another repository type.

Type 15 and 18, dissolved irradiated uranium

This waste consists of 1.2 kg irradiated uranium dissolved or solidified in concrete (Type 15) and three flasks with liquid waste (Type 18). The uranium solution should be solidified[51] in for example small metallic buckets/tins with tight lids before it is packed in canisters to immobilize the nuclides and make the waste chemically stable. It is assumed that each of the two waste types will occupy six canisters that may be placed in a borehole.

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[51] E.g. by casting in a concrete mix

The final solution for this waste has to be further assessed at a later stage, and types 16 and 17, which are described below, could be part of this assessment. Criticality should be considered when placing these waste types in a repository.

Type 16 and 17, irradiated fuel

Waste type 16 consists of 12 kg irradiated fuel, corresponding to 40 small cylinders, kept in 20 A-bins. Waste type 17 consists of 222 kg irradiated fuel, kept in 13 A-bins (two cylinders in each). It is of importance that the cylinders with the fuel are air tight.

If the design of the repository will contain a deep borehole, both waste types should be placed there for final storage. For disposal in a borehole, three cylinders can be packed in a canister, which means that for type 16, there will be 14 canisters with waste, and 9 canisters for waste type 17.

For a concept without borehole, the packages should also be made in such a way that intrusion and leakage can be prevented for very long time. It is suggested that the cylinders with the fuel are packed in tight canisters, which are then packed in individual steel containers.

The containers should be equipped with inside concrete lining for shielding and performance as part of the overall barrier. The lined containers are assumed to be filled half with bentonite and then one canister (for type 16), alternatively two canisters (for type 17), are placed in the centre of a steel container. More bentonite should be added and finally the containers should be sealed with a concrete lid. Thus, seven steel containers are needed for the final disposal of type 16, and 9 steel containers are needed for waste type 17.

The estimated number of containers for each waste type is solely based on the activity of the waste. Criticality should be considered when placing the waste in the repository.

Type 19 - Non-irradiated uranium

The 2 tons of uranium appear, according to the information received from Danish Decommissioning, fully or partly, in metallic form in small packages. The uranium is of natural distribution, which is why there is no risk of criticality. Even if a package with this waste type can easily be protected against radiation, since the gamma radiation is very low, and it is potentially possible to pack it in tight packages, it is recommended that disposal of uranium in metallic form is avoided. It is further recommended that alternative use or disposal of this material is carefully studied at a later stage. If there is no alternative use, the uranium should be oxidised and could be packed in drums prior to disposal.

It is estimated that 20 drums will be needed. For the final disposal it is recommended that the oxidised uranium shall be packed in four steel containers, which are then filled with cement-calcium granulate.

9.1.9 Tailings

Type 21 - Tailings

In Ministry for Health and Prevention (2008) it is recommended that the tailings must be kept saturated with water in order to prevent release of the radioactive gas radon. However, due to the relatively low radon activity (approximately 5.2 Bq/kg according to Danish Decommissioning (2010c), the increased release of radon that may occur if the tailings are kept dry is considered less problematic than the increased leaching of contaminants that may occur if the tailings are kept under water. The preliminary safety assessment indicates that long term release of radon will not result in exceedence of the dose constraints; neither will short term potential releases due to accidents.

It is recommended to pack the tailings directly into ISO containers for transport and final disposal without further conditioning. To prevent risk for spreading of dust in connection with possible later excavation or other intrusion in the repository, to further reduce the risk of radon release, and because it is unlikely that it later will be of interest to remove this waste from the repository, it is recommended to place the tailings in the lower parts of a repository.

Type 22 - Contaminated concrete

It is recommended that concrete waste from the tailing basins is placed in ISO containers for transport and for final disposal at the repository. Voids are to be filled with e.g. cement-calcium granulate, primarily to reduce release of dust and reduce water permeability.

9.2 Quantities of fill and backfill material and containers necessary for conditioning

Based on the above recommendations on packing of the different waste types, the fill amounts to be used inside drums and containers have been estimated. The result of this estimate is presented in Table 9.1.

Total necessary backfill amounts at different repository options are shown in Table 3.2. Thus, the total estimate on fill and backfill amounts is made up as presented in Table 9.2. Since final choice of backfill is not possible at this stage, partly due to the question of the need for reversibility not yet being settled, types of fill are not given in this table. With respect to recommendations given for the specific waste types, reference is made to the descriptions earlier in this chapter.

The total number of necessary drums, bins and containers are given in Table 3.1 and repeated in Table 9.3.

The necessary specifications for drums, containers, fill and backfill is given in sections 3.1.1 to 3.1.4 and Danish Decommissioning (2009) for the containers etc. and in Appendix B for the fill and backfill.

The figures and specifications given are based on the preliminary assessment and should be updated, once the choice of overall repository option has been taken and a classification scheme has been set up based on this and on the actual location of the repository.

Considerations based on transport concerns, as investigated in the prefeasibility study carried out by SIS should also be incorporated.

Table 9.1 Estimated amounts of fill material to be used

Type	Waste	Fill amount (m^3)
1	Graphite	2 - 4
2	Aluminium	30 - 50 *
3	Steel, stainless steel and lead	150 - 200
4	Heavy concrete and concrete	35 - 70
5	Stainless steel, steel and lead	3 - 6
6	Concrete	5 - 10
7	Various components	5 - 10
8	Secondary waste	50 - 60
9	Waste from wastewater treatment	5 - 10
10	Compacted waste and soil	0
11	Waste from DR3 (incl. TSP & TSR)	80 - 120
12	Waste from Hotcell	200 - 250
13	Radiation sources	8 - 10
14	Approximately 20 larger sources	0
15	1.2 kg irradiated, dissolved uranium	0 - 20
16	12 kg irradiated fuel	0 - 20
17	222 kg irradiated fuel	0 - 25
18	Nuclear solution from DR1	0 - 20
19	Non-irradiated uranium	15 - 20
21	Tailings	0
22	Contaminated concrete	25 - 50
Total:		600 - 1000

* Sand should be used

Table 9.2 Estimated total amounts of fill and backfill material.

	Above surface	Near surface	Medium deep	Borehole
Fill, m^3	600 - 1000	600 - 1000	600 - 1000	0
Backfill, m^3	814	1357	2383	70
Total, m^3	1400 - 1800	1900 - 2400	2900 - 3400	70

Table 9.3 Total estimated number of containers etc. for conditioning of the waste.

Steel containers	ISO containers (10ft)	210 l drums	Canisters
125-153	518	5500	43 - 78

10 Cost estimates

The purpose of this chapter is to provide a summary of the cost estimates for the conceptual designs of the different repository types described in Chapter 4.

The costs estimates for the different repository concepts are based on bills of quantities for construction, operation and monitoring, and closure of the concepts. The cost estimates are based on a large number of assumptions, which have been established partly by Danish Decommissioning (as described in the terms of reference for the pre-feasibility study) and partly by the consultant (as described in different working reports, prepared during the initial phases of the study).

To provide a basis for comparison, the costs are estimated as total costs for the whole repository with all facilities and a borehole repository[52], while the type and layout of the main repository is varied between an above surface repository, a near surface repository and the different layouts of the medium depth repository.

Apart from the costs estimates for construction, operation, monitoring and closure of the repositories, also costs for fill material, for externalities and for the further process is given at the end of the chapter.

10.1 Structure and contents

According to Danish Decommissioning (2009), the following items should be addressed in the cost estimates:

- Acquisition of area
- Field investigations
- Detailed design and invitation to tender
- Construction
- Operation
- Monitoring during operation
- Closure
- Monitoring after closure

[52] The borehole cost is also given separately

Some of the above listed items are in the cost estimates treated as inseparable components of other costs, e.g. design costs as a percentage of the construction costs, or can be regarded as one item, such as monitoring during operation and after closure.

For the cost estimates the above listed items were broken down, treated and supplemented as described in the below:

1 Acquisition of area

In principle, the costs for acquisition of area are the same for all repository types and are thus treated once for all concepts.

2 Facilities at the repository

The facilities (apart from the actual repository structures) are the same for all concepts and are thus treated once for all.

3 Construction

The construction costs vary largely between the different repository types and are dealt with individually. From the above listed items, construction costs include:

- Detailed design and invitation to tender
- Field investigations[53]

4 Operation

Only the operation costs during the initial filing period (first year) are considered to vary significantly between the different repository types. The additional initial operation costs (i.e. the costs in addition to the basic operation costs during the active period of 31 years) are addressed individually for the different repository structures.

The basic operation costs during the active period of 31 years are treated once for all concepts. The basic operation costs of an above surface repository may be slightly higher than for the other repository types. However, this variation is considered insignificant at the current conceptual design stage.

So, in this report the operation costs are subdivided in:

- Basic operation costs during active period of 31 years (common for all repository types)

- Additional operation costs during first year (individual for each repository type)

[53] Related to the construction of the repository

5 Closure

The costs for closure vary between most of the different repository types and are dealt with individually.

6 Monitoring

Monitoring during operation and after closure does not differ to a large extend and are thus treated together. In principle, monitoring is the same for all repository types (the costs for the variations in the equipment and the procedures are considered negligible) and it will be dealt with once for all concepts.

10.2 General assumptions for cost estimates

There are many unknowns in the project, so certain assumptions need to be made to allow a comparison between the different concepts. The general assumptions (i.e. assumptions for all cost estimates) for the items listed in section 10.3are given in this section.

To express the estimative character of the determined prices, the totals and sums are rounded to the nearest full thousand DKK. All prices in the cost estimates are given as 2011 net present values.

The uncertainty in the estimated costs is considered as total min and max percentage of the estimated most likely costs or as percent plus/minus variation of the estimated most likely costs. The individual uncertainties for the items listed in section 10.3 are given in this section.

The effect of the uncertainties on the total costs is illustrated in the summary tables in 10.3, where the costs total are calculated for three cases: most likely costs (no variation in prices), minimum costs (min variation/percentage for all prices), and maximum costs (max variation/percentage for all prices). Table 10.1 in section 10.3 provides a summary of the most likely, minimum, and maximum total costs (further subdivided in initial and additional costs) determined on that basis.
The initial cost include: acquisition of land, construction of additional facilities at the repository as well as the construction of the borehole and of the main repository. The additional costs include: the basic operation during 31 years, the additional operation during the first year of the borehole and of the main repository, the closure of the borehole and of the main repository and, finally, monitoring. It should be noted that all additional costs (including salaries) are based on 2011 net present values.

10.2.1 Acquisition of land

It is considered that the land acquired for the repository is located in a rural scarcely populated area. Depending on various conditions, such as land use, location and quality of land, etc., the square meter price will vary significantly.

Land prices may vary considerably depending on the location of the repository. Based on actual market prices, the costs for acquisition of the land may vary between 50 DKK/m^2 and 500 DKK/m^2. Costs for acquisition of the land may vary between 50 DKK/m^2 and 500 DKK/m^2.

A price of 200 DKK/m^2 is selected for the cost estimate and thus the variation is reflected by minimum and maximum costs corresponding to percentages of 25% and 250%, respectively.

10.2.2 Additional facilities at the repository

It is assumed that the additional facilities at the repository facility are established based on containers and lightweight steel structures or similar inexpensive solutions. The general layout of final storage plant is described in section 4.3, which includes a list of the additional facilities.

As opposed to the borehole and the main repository structures, the most likely costs for most of the additional facilities at the repository are not based on reference projects or design calculations, but on engineering estimates. However, the estimated lump sums for the packing and preconditioning facility and for the interim storage building are based on unit prices of similar structures that have recently been constructed by Danish Decommissioning. The overall uncertainty for the additional facilities at the repository is considered by using minimum and maximum percentages of the most likely costs of 75% and 150%, respectively.

10.2.3 Construction

Construction costs establish the major part of the total costs. Construction costs of the main repository have been determined based on bills of quantities (BoQs). The unit prices that have been used in the BoQs are for the major part based on actual bids on comparable items from reference projects, i.e. on extraction and evaluation of actual bid prices. Conservative values have been estimated for the very few items, where no information has been found available. The estimate of the most likely price of the borehole repository is based on a price quote on the actual requirements, dimensions, etc.

As the cost estimates for construction of the repository structures are based on bids for similar projects and on actual price quotes, the estimated prices are considered to represent realistic market prices. However, the actual market price may be influenced by other factors, in particular by the market situation, i.e. the capacity utilisation and thus the purpose of the bidding contractors.

The market situation may be considered by assuming a general uncertainty of 15% to 20% on all price estimates related to the repository structures. 17.5% are taken as plus/minus variation on the most likely costs.

10.2.4 Operation

It is considered that the initial filling period, where the bulk waste amount shall be placed in the repository, lasts for one year. Hereafter, it is assumed that the active operation continues for 30 years with a waste amount of 8 m^3 per year.

For the cost estimate it is assumed that the waste is supplied, packed and ready for deposit, i.e. the cost estimate excludes packing, transport, etc.

Due to the 31 years of operation, the total operational costs are very sensitive to assumptions concerning staffing and salaries. For the determination of the most likely price it has been assumed that the basic operation during 31 years is realised by permanent staff that is hired at certain, individual annual salaries (incl. social charges, etc.). For the initial filling period it is assumed that additional external personnel is hired from a contractor at much higher unit prices for the various (short-term) jobs. The operation costs during the initial filling period are subdivided in operation of the borehole and of the main repository. The overall uncertainty for the operation costs is considered by using minimum and maximum percentages of the most likely costs of 75% and 150%, respectively.

10.2.5 Closure

The costs for closure of the repository amount to a relatively small fraction of the total costs for most concepts, except for the above surface repository, where the closure actually is an integral part of the structure, cf. section 4.4. The procedures for closure are similar for the other repository concepts, but different quantities are required.

In correspondence with the construction costs, closure costs are determined by means of BoQs based on experience from recent comparable projects and actual price quotes. Thus, the general uncertainty of 15% to 20% on all most likely estimated costs also applies to the closure. 17.5% are taken as plus/minus variation on the most likely costs.

10.2.6 Monitoring

Monitoring during operation and monitoring after closure has been treated as one. The monitoring period taken into account for the cost estimates is 1+30 years. Monitoring will be required after the closure. The extent of the post-closure monitoring may actually be on the same level as the monitoring during the first 31 years.

The costs of monitoring during the first year of initial filling are considered to correspond to the costs for monitoring during the 30 years of active period. Moreover, monitoring costs are considered to largely agree between the different repository types (differences in the cost for the required devices are negligible), except the costs for establishing the monitoring wells that might vary with the required depth. However, the required depth also varies with the geological properties of the site and thus variation in the depth of the monitoring wells is not investigated, but considered to be covered by the selected minimum and maximum percentages of the most likely costs (see below).

One initial lump sum plus an annual lump sum are assumed and used for the estimate of the most likely monitoring costs for all repository types. The initial lump sum includes the establishment of the monitoring wells and the costs for other equipment, whereas the lump sum per year includes personal costs and costs for the analyses.

The overall uncertainty for the operation costs is considered by using minimum and maximum percentages of the most likely costs of 75% and 150%, respectively.

10.3 Outcome of cost estimates

In Appendix I, the outcome of the cost estimates is summarised in tabulated forms for:

- The most likely cost (no variation in prices)
- The minimum costs (min variation/percentage for all prices)
- The maximum costs (max variation/percentage for all prices)

The most likely, minimum, and maximum total costs (further subdivided in initial and additional costs) determined on that basis are summarised in Table 10.1. The initial costs include land acquisition, facilities, borehole and main repository. The additional costs include filling of the repository, operation, closure and monitoring.

Table 10.1 Most likely (ML), minimum, and maximum total costs with subdivision in initial and additional costs for all repository facility types

Repository facility type no	Main repository concept [1]	Cut-off structure [2]	Internal diameter	INITIAL costs in mio. DKK[54]			ADDITIONAL cost in mio. DKK[55]			TOTAL costs in mio. DKK		
				ML	Min	Max	ML	Min	Max	ML	Min	Max
1	ASR	–	–	47	35	66	255	192	382	302	227	447
2	NSR	Sheet P	–	70	54	94	274	207	405	344	261	498
3	MDR, GI [3]	DW	33.8 m	219	177	269	351	270	496	570	447	764
4		DW	26 m	184	148	227	307	234	444	490	381	670
5		DW	18 m	154	123	192	281	213	414	436	336	606
6		SP&SCL	33.8 m	216	174	264	351	270	496	566	444	760
7		SP&SCL	26 m	181	145	223	307	234	444	487	379	667
8		SP&SCL	18 m	152	122	190	281	213	414	434	335	604
9	MDR, GR [3]	DW	33.8 m	215	173	263	276	209	409	491	382	672
10		DW	26 m	264	214	321	271	204	402	534	417	723
11		DW	18 m	266	216	323	266	200	397	532	416	720
12		SP&SCL	33.8 m	211	170	259	276	209	409	487	379	668
13		SP&SCL	26 m	261	211	317	271	204	402	531	415	719
14		SP&SCL	18 m	264	214	321	266	200	397	530	414	718
15	MDR, IR [3], [4]	DW	33.8 m	297	241	359	311	234	460	607	475	820
16		SP&SCL	33.8 m	293	238	355	311	234	460	604	472	815
17	MDR, CA	SP&SCL	–	188	151	232	271	204	403	459	356	635
18		SP&RO	–	103	81	131	271	204	403	374	285	534

[1] ASR: above surface repository, NSR: near surface repository, MDR: medium depth repository, GI: operated from ground level, irreversible, GR: operated from ground level, reversible, IR: operated from inside, reversible, CA: cavern

[2] DW: diaphragm wall, SP: secant piles, SCL: sprayed concrete lining, RO: rock

[3] The construction costs of all shaft based MDRs are based on the assumption that the bottom slab is located at the deepest depth possible determined by the structural capacity with given internal diameter and thickness of external walls. In particular, the diameter 33,8 m MDRs could be placed less deep, which would lead to a decrease in the construction costs.

[4] The diameter 26 m and 18 m MDR, IR concepts interfere with the minimum volume requirements and have thus been excluded from the cost estimate.

[5] The costs of the main repository include costs of cut-off structures and excavation.

[54] The borehole costs make up: ML: 6.6 mio DKK, Min: 5.5 mio. DK, Max: 7.8 mio. DKK

[55] The borehole costs make up: ML: 3.2 mio DKK, Min: 2.4 mio. DK, Max: 5.0 mio. DKK

10.4 Fill and backfill material costs

Costs relating to conditioning is made up of cost for drums, containers etc. cost for fill materials and operational cost related to the conditioning process. An estimate is given in Table 10.2 for the costs of fill materials. Costs for the purchase of containers etc. cannot be given at present, since some of them are of a very specialised nature, and a decision with respect to extent of waste types to be placed in a borehole has not yet been taken. Operational costs can for the same reason not be set at present. These costs may include costs in relation to the characterisation process to be carried out.

Table 10.2 Estimated fill and backfill costs.

Fill / backfill material	Unit price, DKK/m^3	Amount, m^3	Total, mio. DKK
Cement-calcium	1,500 [4]	1,400 - 3,400	2.1 - 5.1
Bentonite [1]	1,000 [5]	1,400 - 3,400	1.4 - 3.4
Concrete [2]	650 [5]	1,400 - 3,400	0.9 - 2.2
Sand [3]	350 [5]	30 - 50	0.01 - 0.02

[1]: Bentonite granulate
[2]: "Rørfyld 0/4 mm"
[3]: 0 - 4 mm sand
[4]: Estimated price, since it is not a standard product.
[5]: Market price in Denmark

10.5 Externalities

The monetary risk related to accidents and long term consequences from the repositories cannot at this stage be estimated, since the final location of the repository is not known, and hence the population at risk is unknown. As a first step in the monetary assessment relevant unit values have been determined. Often in risk analysis the value of a statistical life is evaluated. However, in the present risk analysis this would not be sufficient, since the accidents are not immediately fatal, but may cause fatalities over longer term due to diseases caused by release of radionuclides. Hence, values for reduction of lifetime caused by disease are needed in addition to the value of a statistical life.

There are no officially approved Danish methodologies for setting the monetary value of a statistical life and lifetime reduction caused by disease. Based on experience from the EU, COWI has in other projects used the following values:

- Immediate fatalities
    - Value of a statistical life, used for immediate fatalities: 15 mio. DKK (2007 values), or alternatively
    - Reduction in lifetime: 900.000 DKK per person-year (2007 values)

- Fatalities caused by disease
    - Reduction in lifetime caused by disease: 450.000 DKK per person-year (2007 values)

In order to estimate the reduction in lifetime, it is necessary to estimate also the variation in age in the population considered and the expected reduction in lifetime for the population under consideration given different dose levels.

10.6 Costs related to the further process

The costs for the further process are detailed in Appendix J. They comprise costs for the activities listed in Table 11.2. The overall costs for this process are very dependent on the number of locations that will be subject to detailed field studies. If this will be carried out on 5to 6 locations[56], the costs are estimated to 27 to 39 mio. DKK.

[56] With detailed investigations on 2 to 3 locations.

11 Konklusioner og anbefalinger

Nærværende kapitel indeholder konklusioner og anbefalinger baseret på forstudiet. Efter ønske fra Dansk Dekommissionering er kapitlet skrevet på dansk. I den endelige udgaver af slutrapporten vil kapitlet være på engelsk i den engelske udgave af rapporten. Anbefalingerne er baseret på de udførte sikkerhedsanalyser med hensyn til forventelig langtidspåvirkning, risikoen for mulige uheld samt generelle anbefalinger til sikring af en lang levetid for depotet.

11.1 Generelle anbefalinger vedr. depotplacering m.m.

Ved valg af depotplacering er det vigtigt at sikre en god beskyttelse mod påvirkninger, som kan føre til tidligere eller øget påvirkning fra depotet, såsom oversvømmelse. Der gives derfor følgende generelle anbefalinger:

Et depot bør placeres i områder, der ikke er udsat for oversvømmelser fra havet eller stigende grundvand[57] samt erosion. Depotet bør endvidere placeres således, at vandet strømmer væk fra og ikke ind i depotområdet i tilfælde af voldsomme regnskyl. Endelig bør der ved placering af depotet tages højde for forventelige havstigninger som følge af f.eks. klimaændringer.

Et depot bør ikke placeres i områder med kendt risiko for jordskælv eller væsentlig risiko for sætninger.

Et depot bør placeres tilstrækkeligt langt fra større kommercielle lufthavne, således at risikoen for uheld som følge af flystyrt minimeres. Ved valg af en konkret lokalitet, bør der foretages en specifik vurdering af risiko ved flystart og landing gældende for både kort og langt sigt.

Hvis der i depotopbygningen anvendes armeret beton i de ydre dele af konstruktionen, skal det sikres, at klorid indholdet i det omgivende miljø ikke udgør en risiko med hensyn til korrosion. Tilsvarende må der heller ikke findes en væsentligt jævnstrømskilde i nærheden, f.eks. katodiske beskyttelsessystemer af ledninger eller underjordiske stålkonstruktioner.

[57] Væsentligt udover, hvad der er forudsat ved projekteringen af depotet.

11.2 Generelle anbefalinger vedr. konstruktion, indretning, fyldning og drift af depoterne

11.2.1 Visuel fremtoning af depotet

Det er foreslået, at depotets visuelle fremtoning bliver karakteristisk og af en art, som også i en fjern fremtid sikrer, at man vil være opmærksom på, at lokaliteten indeholder noget usædvanligt, og som vil styrke den kollektive hukommelse omkring anlæggets historie.

Det er foreslået, at depotanlægget designes med en yde afgrænsning, der kan forbinde anlægget med omgivelserne og samtidigt fastholde anlæggets indre geometriske form, således at det vil være muligt at genkende depotanlægget fra luften i lang tid fremover. Det er således foreslået, at anlæggets indre geometri opbygges, således at det ligner det internationale symbol for radioaktivitet. En uddybning af dette forslag findes i Kapitel 5 og i Appendiks L.

11.2.2 Reversibilitet

Det har været ønsket, at depoterne evt. skulle være reversible, således at relevante affaldstyper kunne udtages på et senere tidspunkt. Dette vil have betydning for dels udformningen af depotet, dels mulighederne for anvendelse af fyldmaterialer i depotet. I både skitseringen og den indledende sikkerhedsanalyse er dette vurderet. Nærværende afsnit indeholder de overordnede konklusioner.

Da beholdernes levetid generelt er mindre end f.eks. 300 år[58], må det forventes, at processen med udtagning vil være kompliceret og forbundet med større risiko end processen med placering af beholderne i depotet. Såfremt man ønsker at kunne udtage visse affaldstyper på et senere tidspunkt, og da det næppe vil dreje sig om alle affaldstyper, bør dette indgå i den overordnede organisering af affaldets placering i depotet.

Det er i skitseprojekteringen anbefalet, at der anvendes sand eller bentonit som fyld i depotet, såfremt det skal være reversibelt, og ellers sand eller beton[59], se nærmere detaljer under de enkelte depottyper i Kapitel 4. For flere affaldstyper er det dog som følge af sikkerhedsanalyserne anbefalet, at der, for at reducere risikoen for forhøjede doser ved uheld eller ved den langsigtede påvirkning, anvendes enten bentonit eller beton til fyld omkring containere eller tromler, såfremt de er placeret direkte i depotet[60]. Procedurerne for fjernelse af fyldmaterialer vil være vanskeligere, såfremt der anvendes bentonit eller beton, hvilket vil øge prisen samt risikoen for nærkontakt med aktivt materiale.

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[58] For de typisk anvendte beholdere mindre end 100 år.

[59] Anvendelse af et fyldmateriale i depotet, altså omkring containere og tromler vil være en nødvendig for at sikre stabiliteten af depotet, også når containere og tromler er korroderet.

[60] Se detaljer i Kapitel 9.

Nogle af de arbejdsmiljømæssige problemer relateret til reversibilitet samt risikoen for de omkringboende ved en senere udtagning af affaldet er behandlet nærmere i Kapitel 8. Det er en generel erfaring fra nukleare anlæg, at genhåndtering af affald skal udføres med stor forsigtighed.

Reversibilitet kan opretholdes i større eller mindre omfang, alt efter depotets øvrige udformning. I skitseprojekteringen er følgende muligheder for reversibilitet vurderet:

- Et depot placeret på overfladen vil i princippet altid være reversibelt, da afdækning og membraner samt fyld omkring affaldet (såfremt dette ikke er beton) vil kunne fjernes.

- Et depot placeret lige under jordoverfladen vil relativt enkelt kunne være reversibelt, så længe der ikke fyldes omkring containerne. Dette vil dog af hensyn til ulykker ikke være muligt for visse affaldstyper (se Kapitel 9) allerede fra depotets indfyldning. Hvis depotet skal genåbnes efter det har været fuldstændigt lukket, skal de afskærende væggges stabilitet først vurderes, hvorefter membran og de beskyttende beton dæk kan afmonteres.

- Et mellem-dybt depot kan etableres både reversibelt og irreversibelt, hvilket vil have betydning for selve konstruktionen (se Kapitel 4). Et reversibelt, mellem-dybt depot kan etableres med fyldning enten oppe fra eller indvendigt fra. Fordelen ved et depot med fyldning indefra er, at det i princippet vil være muligt at udtage bestemte affaldstyper efterfølgende uden at fjerne evt. overliggende affaldstyper. Ved et depot betjent oppefra vil senere udtagning af containerne endvidere afhænge af tilstanden af de løftekroge, der sidder på containerne. I øvrigt gælder de samme kriterier for de mellemdybe depoter som for et depot placeret nær overfladen.

- Et mellem dybt depot kan også etableres som et system af vandrette kaverner. Her er der principielt adgang til de enkelte "fingre", indtil disse lukket med en betonprop. Betonproppen kan også nedbrydes efter lukning i hvert fald i en periode op til 200 til 300 år. Adgangsskakten til fingrene kan holdes åben og om nødvendigt forstærkes på et senere tidspunkt.

Reversibilitet vil alt andet lige øge omkostningerne ved depotet med i størrelsesordenen 10 til 25 %, især afhængigt af depotets dybde. Heri indgår ikke udgifter til efterfølgende håndtering af affaldet samt afrensning af beholdere og containere for fyldmateriale og efterfølgende håndtering af dette fyldmateriale, som må forventes at indeholde et vist niveau af aktivitet afhængigt af, hvor længe der går, inden affaldet tages ud. Derudover vil en senere udtagning af affald fra depotet resultere i en øget risiko for påvirkning af omkringboende som følge af mulige uheld i forbindelse med udtagningen, ikke mindst fordi emballager m.m. kan risikere at være korroderede og skrøbelige.

11.2.3 Generelle anbefalinger vedrørende depotets konstruktion

Betonkonstruktioner skal udføres vandtæt i klasse 3 i henhold til DS/EN 1992-3:2006.

Design og udførelse af depotet skal udføres med henblik på at undgå sprækkedannelse i betonkonstruktionerne. Sprækker kan opstå som følge af dårligt kompaktering af betonen, forkert betonblanding, mangel på beskyttelse mod fordampning, dårligt udført støbning, termisk revnedannelse p.g.a. for store temperaturforskelle f.eks. mellem ny og tidligere støbt beton, for hurtig udtørring af betonen. Sådan sprækkedannelse skal minimeres ved omhyggelig udførelse.

Andre typer af revner er afhængige af design og kan minimeres ved bl.a. at sikre at betonen er forstærket korrekt, og at de mellemdybe depoter udføres som cirkulære konstruktioner.

Endvidere bør konstruktionerne være designet til at modstå forventelige jordskælv m.m.[61] indenfor den krævede levetid (min. 300 år).

De i Tabel 11.1 nævnte, primært kemisk relaterede, årsager kan have negativ effekt på betonkonstruktioner og bør modvirkes som beskrevet.

Tabel 11.1 Mulige skadelige påvirkninger på konstruktionerne samt anbefalede modvirkninger

Nedbrydningsmekanisme	Vurdering og tiltag
Sulfat angreb	Dette kan imødegås gennem valg af cementtype eller specifikke blandinger, som kan sikre sulfatresistens i det konkrete miljø.
Forsinket ettringit dannelse (Delayed Ettringite Formation, (DEF)	Risikoen for DEF modvirkes ved et krav om maks. 65 °C under størkning. Brug af cement med et flyveaske indhold på 65-70 % minimerer også risikoen for DEF.
Alkali-aggregat reaktioner og alkali-karbonat reaktioner	Risikoen for reaktioner skal undgås ved passende valg af ikke reaktive aggregater. Fine såvel som grove skal testes med standard metoder for at udelukke risiko for alkali reaktioner.
Sprækkedannelse i nystøbt beton	Tidlig sprækkedannelse skal holdes under kontrol f.eks. ved hjælp af afkølingsprocedurer baseret på temperatur/stress analyser.
Udsivning	Udsivning skal begrænses via et lavt vand/cement forhold i betonen. På sigt kan ind- og udsivning til og fra depotet dog ikke undgås helt.

[61] Svarende til en forventelig 1000 års hændelse.

Betonkonstruktioner skal forsynes med en passende membran på ydersiden. Den konkrete placering er angivet for de enkelte depottyper i Kapitel 4. Denne membran er ikke medtaget som en barriere i sikkerhedsanalysen, da levetiden af den kan variere betragteligt.

Såfremt depotet er placeret i jord- og grundvandslag med et lavt klorid indhold af hensyn til potentiel korrosion af armering, vil karbonat induceret korrosion udgøre den afgørende parameter i forhold til design af depotkonstruktionen.

Udover ovenstående er der i Kapitel 4 angivet en række specifikationer for materialer og udførelse for de enkelte depottyper, herunder krav til lertype og udførelse af lermembraner til depottypen placeret ovenpå jorden og til de anvendte plastmembraner. For denne depottype anbefales det endvidere, at der etableres et elektronisk lækage detektionssystem under plastmemebranen. Det er tillige væsentligt, at der er meget præcise krav til udførelse af membraner m.m. samt procedurer for rapportering og udbedring af skader på membraner og betonkonstruktioner under etableringen.

11.2.4 Generelle anbefalinger vedrørende depotets fyldning og drift

I forbindelse med opfyldningen bør der være etableret midlertidig afskærmning til reduktion af støvspredning ved evt. tab af tromler eller containere.

Affald, der er brændbart, bør straks efter placeringen i depotet afdækkes med passende fyld, se Kapitel 9, for at undgå, at en evt. brand spredes til dette affald.

Der bør sikres tilstedeværelse af brandslukningsmateriel både på anlægget som helhed og ved selve depotet for at reducere konsekvensen i forbindelse med uheld relateret til brand.

Endvidere bør elektriske installationer være udført af ikke-brandbare materialer, ligesom der ikke bør placeres transformere inde i depoterne (eller i nærheden af området for midlertidig opbevaring af affaldet inden endelig deponering).

Der bør straks udlægges fyldmateriale omkring grafitaffaldet efter placeringen i depotet for at undgå, at en evt. brand opvarmer evt. ikke udglødet grafit. Der bør ikke placeres brændbart materiale (sådanne affaldstyper, paller m.v.) i nærheden af grafit affaldet.

Der skal for de relevante typer af særligt affald, se kapitel 9, tages højde for kritikalitet ved affaldets placering i depotet.

De affaldstyper, det er mindst relevant evt. at genudtage[62] bør ud fra dette synspunkt placeres nederst i depoterne tillige med de mest støvende affaldstyper.

De affaldstyper, det kan være mest relevant at genudtage, kan dog samtidigt være de affaldstyper, der er mest kritiske i forhold til potentiel påvirkning af omgivelserne, hvorfor det kan være mest hensigtsmæssigt at placere dem dybest af sikkerhedsmæssige hensyn. Derfor bør reversibilitetsspørgsmålet overvejes nøje, inden organiseringen af affaldets placering i depotet.

Der skal opstilles et moniteringsprogram, der overvåger depotets potentielle påvirkning af omgivelserne (grundvand, overfladevand, gasudsivning, planteoptag m.m.). Grundvandsovervågningen vil typisk omfatte 1 opstrøms og 3 nedstrøms boringer. Denne endelige udformning af programmet vil afhænge af den konkrete lokalitet og den valgte depottype.

11.3 Anbefalinger baseret på den præliminære sikkerhedsanalyse

11.3.1 Depotkoncepter

Der er i sikkerhedsanalysen vurderet på 2 typer af terrænnære depoter: en type placeret ovenpå jorden og en type placeret lige under jordoverfladen.

For disse to typer er placering henholdsvis ovenpå og nede i ler, moræne, kalk (skrivekridt) og klippe undersøgt. For depoterne under overfladen er både undersøgt en placering lige under overfladen og en placering i en dybde på omkring 30 m under jorden. Ved de forskellige placeringer er der taget hensyn til sandsynlige tilgrænsende lag.

Der er i sikkerhedsanalysen vurderet depoter placeret mellemdybt for alle 4 geologier. For den højestliggende depottype (bund i ca. 50 m under terræn og den bredeste, se kapitel 4 for detaljer) er der beregnet for alle 4 geologier. For dette depot er der på basis af skitseprojekteringen set på både et depot betjent fra oven og et depot betjent indefra. For de dybere beliggende depoter er der kun regnet med betjening oppefra, idet betjening indefra vil optage en uhensigtsmæssig stor del af depotets volumen. Der er endvidere på basis af skitseprojekteringen også set på et mellemdybt depot opbygget som en kaverne i fed ler, kalk eller klippe. For den mellemste depottype er der beregnet for ler, kalk og klippe (bund i ca. 70 m under terræn) og for den dybest beliggende depottype er der beregnet for fed ler. Dette er valgt ud fra de sandsynlige dybder af de pågældende lag. Der er derudover set på, hvilke vandførende lag vil kunne forekomme i hvilke dybder i de pågældende formationer, således at der samtidigt kan foretages en vurdering af betydningen af dette.

[62] F.eks. hvor genoparbejdning ikke er relevant

Borehullet er i sikkerhedsanalysen foreslået placeret i kalk eller klippe i enten 100 til 150 meters dybde eller 250 til 300 meters dybde. Resultaterne vil også være gældende for depoter placeret i mellemliggende dybder.

Det vil være muligt med den opstillede model at vurdere placering i andre geologier efterfølgende.

Der er som tidligere nævnt foretaget indledende sikkerhedsanalyser af både potentielle uheldssituationer og af den mulige langsigtede påvirkning af en reference person. Sikkerhedsanalyserne er baseret på konservative, men realistiske forudsætninger.

Anbefalingerne er baseret på, at hændelser ikke bør give væsentlig risiko for en dosis per år hos referencepersonen på over 1 mSv, og at tillægsdosis fra den langsigtede eksponering ikke bør overstige 0,01 mSv per år.

De følgende anbefalinger søger at tage hensyn til den variation, der er for de relevante geologiske parametre, idet der jo er anvendt generiske geologier. Der er derfor inkluderet en vis sikkerhedsmargin i forhold til overholdelse af strålingskriteriet (i størrelsesordenen en faktor 1000 til 100.000, da usikkerheden på de resulterende dosisberegninger som følge af mulig variation i de afgørende parametre mindst er af denne størrelsesorden). Baseret på konkrete geologier og en stedsspecifik vurdering/undersøgelse vil variationen være mindre, hvorfor anbefalingerne evt. vil kunne ændres.

Endvidere er der skelet til, hvilke affaldstyper, der stadigt efter lang tid vil have et højt aktivitetsindhold, både absolut og per mængdeenhed, og som derfor i lang tid vil kunne give anledning til væsentlige doser, såfremt de ikke afskærmes.

Grundlaget for opgaven har jf. Beslutningsgrundlaget været, at de skitserede depoter skulle have en levetid på 300 år. De indledende sikkerhedsanalyser har som delresultat, at den samlede levetid af barrieresystemet for de anbefalede løsninger ligger i mellem 500 og 1000 år. Efter denne periode vil der stadigt være affaldstyper tilbage i depotet, som indeholder væsentlig aktivitet. Det drejer sig om følgende affaldstyper[63]:

- Type 1 (grafitaffald)
- Type 8 (sekundært affald fra dekontaminering)
- Type 12 (eksisterende affald fra Hot Cell)
- Type 13 (eksterne strålekilder)
- Type 14 (særlige strålekilder)
- Type 15 og 18 (bestrålet uran)
- Type 16 og 17 (bestrålet brændsel)
- Type 19 (ubestrålet uran)
- og i begrænset omfang type 21 (tailings).

[63] Se kapitel 9 for nærmere beskrivelse

Det er følgende nuklider, som giver anledning til aktiviteten:

C-14, Ra-226 samt uran og plutonium nukliderne samt disses døtre.

Det er således disse affaldstyper og nuklider, der er afgørende for sikkerhedsanalyserne relateret til den langsigtede udvikling.

Med de nedenfor beskrevne anbefalinger skønnes det årlige tillægsbidrag til dosis fra affaldsdepotet maksimalt at ligge på 0,00001 mSv for referencepersonen, når usikkerheden knyttet til beregningerne også tages i betragtning[64].

11.3.2 Generelle anbefalinger

Det er en generel konklusion, at placering af et depot, så vidt muligt ikke bør foretages tæt på væsentlige vandførende lag[65], da dette har større betydning end dybden af depotets placering.

Da sådanne er mere forekommende i morænelersformationer (og de typisk har højere hydraulisk ledningsevne), vil det generelt være nødvendigt at foretage større afskærmning af affaldet i disse formationer (i form af tykkere godstykkelse af beholdere og anvendelse af fyld både i og omkring beholdere og containere) for at opnå samme dosisniveau som for de øvrige geologier.

Med hensyn til mulige påvirkninger p.g.a. uheld vil der generelt være størst risiko ved depoter placeret nær overfladen, dels fordi de er mere udsat for f.eks. udslip forårsaget af boring og gravning, når depotet er "glemt" samt af flystyrt og meteoritter i hele depotets levetid, fordi spredning af støv vil være mere uhindret ved sådanne hændelser samt f.eks. tab af affald ved placering af dette og ved evt. udtagning af dette igen.

For alle depottyper undtagen et borehul[66], (der også som udgangspunkt betragtes som irreversibelt) vil der som følge af håndteringen af affaldet på lokaliteten, inden det fyldes i selve depotet[67], være risiko for at en referenceperson modtager en dosis støre end 1 mSv i forbindelse med, at depotet fyldes eller tømmes. Denne risiko vil være mindre end 1 x 10^{-6}, se også uddybningerne heraf i kapitel 8.5.

11.3.3 Terrænnære depoter

På grundlag af sikkerhedsanalyserne kan de følgende vurderinger gøres.

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[64] Dvs. når der er taget højde for den variabilitet og usikkerhed, der er knyttet til alle de parametre, der ligger til grund for den indledende sikkerhedsanalyse.

[65] I modellen er det regnet som direkte tilgrænsende lag.

[66] Hvor risikoen for at beholderne går i stykker er mindre på grund ag den meget kraftige godstykkelse.

[67] Håndtering i det temporære depot inden fyldning i selve depotet. Uheld kan skyldes tab af beholdere og brand.

Terrænnære depoter bør kombineres med et borehul, hvori følgende affaldstyper placeres: Type 12, 13, 14, 15, 16, 17 og 18[68]. Se den nærmere beskrivelse af disse affaldstyper i afsnit 11.3.1 samt i kapitel 9.

Såfremt der ønskes et depot placeret på overfladen, bør det placeres på moræneler. For uopsprækket klippe og fed ler vil den hydrauliske ledningsevne være så lille, at udsivning gennem siderne af depotet kan føre til utilsigtet overfladisk afstrømning. Skrivekridt kan være meget opsprækket i de øverste lag, og den foretagne generiske analyse giver ikke et tilstrækkeligt grundlag for at vurdere disse forhold.

Når containere og tromler er placeret I et depot på overfladen, skal der udlægges fyldmateriale mellem containerne og tromlerne og ovenpå disse. Materialet skal kompakteres med skråninger på maksimalt 1:3 for at undgå erosion af lag udlagt oven på topmembranen. Afhængigt af, om der ønskes en reversibel eller ikke reversibel løsning kan der anvendes bentonit eller beton som fyldmateriale.

Terrænnære depoter i moræneler (under overfladen) bør ikke etableres i geologier, hvor moræneleren direkte overlejrer bryozokalk eller anden vandførende formation, da dette vil give mulighed for hurtig påvirkning af drikkevand.

For alle terrænnære depoter viser sikkerhedsanalyserne, at der sal anvendes bentonit eller beton som fyldmaterialer mellem containere/tromler for at sikre en tilstrækkeligt lille dosis.

Der bør etableres et drænsystem omkring et terrænnært depot, således at regnvand ledes bort fra depotet. Der bør endvidere i moniteringsfasen etableres monitering af grundvandsstand omkring depotet, således at der kan foretages grundvandssænkning om nødvendigt.

I opfyldningsperioden bør der etableres en form for midlertidig overdækning af et depot placeret på overfladen, således at der ikke kommer vand til affaldet for tidligt. Endvidere bør denne overdækning udføres, således at den også giver beskyttelse mod støvspredning i tilfælde af tab af affaldsbeholdere. For et depot placeret på overfladen bør depotets sider eller omkransning udføres på en sådan måde, at ekstreme regnhændelser ikke forårsager vandfyldning af depotet og udvaskning fra dette i opfyldningsperioden.

I overvågningsperioden bør måling af radioaktivitet i beplantningen på og omkring depotet indgå i moniteringsprogrammet. I denne periode bør grundvandsstanden også moniteres omkring depotet, og der bør være etableret mulighed for grundvandssænkning om nødvendigt.

[68] Evt. kun dele af type 12 og 13 afhængigt af den endelige karakterisering

Uheldshændelser med risiko for forhøjet påvirkning af referencepersonen er (udover de generelle risici, se under Generelt) knyttet til flystyrt og meteornedfald samt til gravning og boring i depotet, når det er "glemt". De samlede sandsynligheder for, at referencepersonen opnår en dosis større end 1 mSv som følge af alle typer uheld[69], vil ligge i størrelsesordenen 1 x 10^{-5} for begge typer terrænnære depoter.

For at reducere risikoen for uoverlagt udgravning i depotet anbefales det at der etableres en form for synlig advarsel omkring og over depotet, som kan advare om tilstedeværelsen af depotet i en udgravningssituation og bidrage til en lang kollektiv hukommelse af baggrunden for denne lokalitet.

Der er generelt en større risiko for utilsigtet udslip af radioaktivitet i tilknytning til uheldshændelser m.m. knyttet til de terrænnære depoter end til de mellemdybe. Det skyldes især den forholdsvis store sandsynlighed for gravning eller boring i et sådant depot tillige med den lidt større risiko for spredning af støv til naboer i forbindelse med uheld ved depotets fyldning.

11.3.4 Mellemdybe depoter

For de mellem-dybe depottyper giver sikkerhedsanalysen følgende vurderinger.

Medmindre depoterne placeres nær vandførende lag (se ovenfor) er det forventelige dosisniveau af samme størrelsesorden for kalk og fed ler[70]. Placering i klippe giver som forventet meget lave doser. Det skal dog understreges, at der i den indledende sikkerhedsanalyse er forudsat, at klippen er uopsprækket. Vurdering af transport i opsprækket klippe vil afhænge meget af de stedsspecifikke forhold og dermed af hvilke recipienter, der kan forventes at blive påvirket af transport i sprækkerne.

Ved placering nær vandførende lag vil det være nødvendigt med fyld af bentonit eller beton for at opnå tilstrækkeligt lave langsigtede doser. Dette er uanset, om noget af affaldet placeres i et borehul eller ej. Dog vil placering af affaldtype 13 i borehullet generelt medføre lavere langsigtede doser.

Der er generelt lille forskel i dosis mellem de forskellige skitserede typer af mellemdybe depoter, fordi der er lille forskel på det gennemstrømmede tværsnitsareal. Forskellen mellem de reversible og irreversible depottyper vil bero på det anvendte fyldmateriales effekt (henholdsvis bentonit og beton), se afsnit 11.3.7.

Risici for referencepersonen for at få en forhøjet dosis relateret til uheld i mellemdybe depoter er (udover de generelle risici, se under Generelt) knyttet til mulig brand i materiel anvendt i depoterne, som fyldes (eller tømmes) indefra.

[69] Inkl. håndteringsuheld.

[70] Generelt dog lidt lavere for kalk (skrivekridt) end for fed ler-

Den samlede sandsynlighed for at en referenceperson modtager en dosis på mere end 1 mSv som følge af alle typer uheld, der kan forekomme ved et mellemdybt depot, er af størrelsesordenen 1·x 10^{-6}[71]. Sandsynligheden er lidt større for depoter betjent oppefra end for de andre depottyper, da muligheden for spredning af støv er størst for depoterne betjent oppefra. Dette kan dog reduceres ved de tidligere anbefalede tiltag.

Det fremgår af sikkerhedsanalysen, at kun uheld i forbindelse med fyldning og tømning af depotet er relevant for de mellemdybe depoter p.g.a. af den store afstand fra depotets top til jordoverfladen (< 30 m).

De mellemdybe depoter giver mindst risiko for spredning af forurenet støv til naboer i tilfælde af uheldshændelser[72]. Her er depottyperne, der betjenes indefra de mest sikre, da det er forudsat, at ffaldet sænkes ned via en elevator. For personalet er især skaktløsningerne dog mindst sikre, idet denne types konstruktion medfører lange flugtveje.

I tilfælde af brand vil især de depottyper, der betjenes indefra give større risiko for personalet end de terrænnære depoter. Det kan reduceres ved at etablere sikre flugttrapper og brandslukningssystemer. Sådanne løsninger skal selvfølgelig godkendes af arbejdsmiljø- og brandmyndigheder.

Det anbefales, at der opstilles klare procedurer for nedlukning af ventilation m.m. i tilfælde af brand for at reducere konsekvensen i forbindelse med uheld relateret til brand.

Der bør etableres filter på udluftningen af depotet, og afkastet bør som en del af moniteringsprogrammet undersøges løbende for aktivitet, ligesom der bør etableres mulighed for at lukke afkastet.

11.3.5 Borehuller

Sikkerhedsanalyserne viser, at den forventelige dosis ved placering af de nævnte affaldstyper i et borehule vil være meget lille. Der er ikke knyttet yderligere specifikke anbefalinger til borehuller.

11.3.6 Økonomi

På basis af skitseprojekteringen er følgende samlede omkostninger opstillet for de forskellige mulige depottyper, se tabel 10.1. Dette indeholder som nævnt ikke omkostninger til eventuel efterfølgende udtagning og håndtering af affald [73]fra et reversibelt depot. Overslaget indeholder omkostninger til placering af affaldet i depotet, monitering og drift af depotet i en 30-årig periode samt endelig lukning af depotet.

[71] Dog er maks. sandsynligheden lavere end for de terrænnære depoter

[72] Bortset fra borehullet.

[73] og evt. forurenede fyldmaterialer

Den store variation i totalomkostningerne for en kaverneløsning afhænger af, om kaverne etableres i klippe eller materialer, der vil kræve afskæring af vand.

Udover omkostningerne i Tabel 10.1 tilkommer omkostninger til fyldmaterialer og til dem videre proces frem til depotets etablering og fyldning. Størrelsesordenen er især afhængig af de nødvendige feltundersøgelser. De samlede omkostninger til dette udgør henholdsvis 1 til 5 mio. DKK til fyldmaterialer, afhængigt af depottype og valg af fyldmateriale, og 27 til 39 mio. DKK afhængigt af antallet af lokaliteter, der undersøges[74].

Derudover bør en endelig sikkerhedsanalyse inkl. samfundsmæssige omkostninger relateret til potentielle påvirkninger af naboer til den konkrete lokalitet. relevante enhedsomkostninger er anført i afsnit 10.5.

11.3.7 Konklusioner / anbefalinger vedr. konditionering

Det er en fordel, såfremt der kan anvendes et begrænset antal forskellige containere til konditionering af affaldet. Containere af samme størrelse og form er nemmere at håndtere og placere i depotet, end hvis de er af forskellig størrelse og form.

Generelt anbefales det, at efterfylde containere og hvor muligt tromler for at sikre mod sætninger af affaldet i depotet. Valg af fyld vil afhænge af ønsket om at kunne få adgang til evt. senere efterbehandling af affaldet eller ej, se Kapitel 11.2.2. De indledende sikkerhedsanalyser har ikke givet et tilstrækkeligt grundlag for at påvise væsentlige forskelle mellem fyldmaterialernes samlede effekt med hensyn til tilbageholdelse af nuklider på langt sigt[75]. Datagrundlaget for især vurdering af cement calcium granulatens egenskaber er endnu ikke er så stort[76]. Dette bør undersøges nærmere inden endeligt valg mellem fyldtyperne. Generelt afhænger tilbageholdelsesegenskaberne af en blanding af fyldets betydning for den samlede permeabilitet[77] og af de geokemiske forhold, som fyldet er med til at etablere, som dels har betydning for opløseligheden af nukliderne i affaldet og dels har betydning for fyldets evne til at binde nukliderne i en periode, hvorfor de er henfaldet yderligere, inden de kommer ud af depotet.

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[74] Det er som grundlag for overslaget forudsat, at der foretages indledende undersøgelser på 5 til 6 lokaliteter og mere detaljerede undersøgelser på 2 til 3 lokaliteter.

[75] Forskellen i effekt mellem beton og bentonit (som er de to fyldmaterialer, der er bedst belyst i litteraturen) vil også afhænge ag det enkelte nuklid, hvorfor en entydig afgørelse bliver vanskeligere.

[76] Den foreliggende begrænsede datamængde for granulaten tyder som forventeligt på gode egenskaber på grund af granulatens mulighed for at skabe et basisk miljø. Omvendt vil granulatens permeabilitet selv efter hærdning være større end for både bentonit og beton.

[77] Der bestemmer hvor hurtigt vandet kommer ind til beholderne og kan fremme deres korrosion, og hvor hurtigt vandet kommer ud af beholderne og depotet efter at have været i kontakt med affaldet.

Generelt viser sikkerhedsanalyserne at anvendelse af fyld mellem containere m.m. vil reducere den langsigtede påvirkning med 2 - 3 størrelsesordener. Såfremt der ikke er foreslået en specifik fyldtype omkring containerne eller i depotet som helhed, er det af hensyn til stabiliteten af depotet anbefalet, at der efterfyldes med sand.

I Kapitel 9 er foretaget en opsamling af anbefalinger vedr. konditionering baseret på sikkerhedsanalyserne af både uheldssituationer og langsigtet påvirkning.

De affaldstyper, der er mest kritiske i forhold til spredning i forbindelse med uheldshændelser m.m., er grafitaffaldet (type 1), det sekundære affald fra Hot Cell (type 8), affaldet fra spildevandsrensningen (type 9) og det blandede affald (type 10). Dette skyldes primært deres brændbarhed eller at de i væsentligt omfang består af eller indeholder store mængder af finkornet materiale set i sammenhæng med den samlede mængde af den pågældende affaldstype.

Der skal foretages en yderligere vurdering af den specifikke konditionering af affaldstyperne 15 til 18 (bestrålet affald og bestrålet brændsel). Derudover er det meget væsentligt at tage hensyn til kritikalitet, når disse affaldstyper placeres i en depottype.

Udover de nævnte anbefalinger, som primært tager udgangspunkt i hændelser i eller ved fyldning af depotet, bør der ved konditioneringen tages hensyn til den nødvendige sikkerhed ved selve håndteringen af affaldsbeholderne samt til krav, som kan blive stillet af hensyn til transporten fra Risø til depotet.

Derudover anbefales det, at der foretages en endelig karakterisering af affaldet i henhold til IAEAs og EU's bestemmelser for dette, inden endelig emballering og deponering. Man skal være opmærksom på, at en ændret klassificering (og dermed emballage) kan medføre ændringer i det nødvendige depotvolumen, hvorfor klassificeringen bør foretages, inden den egentlige projektering af depotetsåledes atd er i projekteringen sikres et tilstrækkeligt volumen til mulige ændringer.

Da der kan opstå fejl eller skader ved pakningen af affaldet, fysisk skade på beholderne under håndtering i forbindelse med oplagringen og evt. korrosion af emballager under opbevaringen, er det vigtigt, at der etableres en procedure for tjek af alle affaldspakker, før de transporteres fra Risø til depotet.

11.4 Anbefalinger vedr. fremtidige studier m.m.

11.4.1 Klassificering af affald

Den i forstudiet foretagne opgørelse og fordeling af nuklider på affaldstyper er præliminær og udført med henblik på forstudiets formål. Der bør, inden endelig konditionering og organisering af affaldets placering i depotet (herunder placering i et borehul) og fastlæggekse af nødvendigt depotvolumen foretages en mere detaljeret karakterisering af affaldet, som også tager hensyn til de krav, der knytter sig til den faktiske lokalitet. Denne karakterisering bør udføres i overensstemmelse med IAEA's og EU's standarder herfor og danne baggrund for en samlet detaljeret opgørelse af affaldet indeholdende beskrivelse af materialet, volumen vægt, aktivitet og fordeling på specifikke nuklider.

Det bør vurderes nærmere, om affaldstyperne eller dele heraf med et væsentligt metalindhold kan dekontamineres eller evt. reduceres i volumen ved smeltning eller for aluminiummets vedkommende ved komprimering.

Det anbefales, at kravene til klassificering og konditionering indgår som grundlag for den endnu ikke udførte dekommisionering.

Andre muligheder for håndtering af det metalliske uran bør undersøges nærmere.

11.4.2 Sikkerhedsanalyser

Generelt er det væsentligt, at de næste sikkerhedsanalyser baseres på parametre, der mere konkret er gældende for de vurderede lokaliteter. Dette gælder både parametre knyttet til de geologiske og hydrologiske forhold[78] og til forhold vedrørende de konkrete naboer, der potentielt kan udsættes for en påvirkning i forbindelse evt. uheldshændelser m.m. samt på langt sigt. Herunder bør mere realistiske parametre for andel af påvirkede fødevarer, udendørs ophold, relevante recipienter m.m. inddrages[79]. Endvidere bør dosisberegninger udføres for flere aldersgrupper som defineret af ICRP.

Ved den videre analyse af eventuelle klippelokaliteter er det væsentligt at få sprækkesystemet, herunder sprækkeretningen, detaljeret beskrevet.

I de næste faser i processen anbefales det, at modelleringen af transport i grundvandszonen som et led i sikkerhedsanalysen gøres mere detaljeret og omfatter yderligere relevante forhold, såsom sprækketransport og stokastisk håndtering af heterogenitet baseret på en mere specifik viden om den konkrete lokalitet.

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[78] En eventuel placering af et overflade depot på fed ler eller klippe bør således baseres på lokalspecifikke forhold og ikke på generiske betragtninger.

[79] Bl.a. er det hensigtsmæssigt konkret at vurdere, hvor stor en andel af tiden personer kan forventes at opholde sig udendørs på den konkrete lokalitet i forhold til de forudsatte 20 % af tiden.

Det anbefales endvidere, at modellen omfatter hensyntagen til både kortvarige og langvarige transiente forhold (f.eks. grundvandspumpning, vandstandsvariation og på langt sigt konsekvenser af klimaændringer). Parametre, som er afhængige af de geokemiske forhold, bør også bestemmes i større detalje baseret på den konkrete lokalitet.

11.4.3 Design

I den videre design af depotet bør der inkluderes plads til den (relativt lille) fremtidige mængde af affald fra f.eks. industrielle kilder og sundhedssektoren, der skal placeres i depotet.

Der bør ligeledes foretages mere detaljerede studier af de nødvendige krav til den anvendte beton[80] baseret på de værst tænkelige forhold med hensyn til aggressivitet, der kan forventes på den konkrete lokalitet. Tilsvarende studier bør udføres med hensyn til kvalitet af de forskellige relevante membrantyper. Der bør foretages en stokastisk modellering til beregning af de nødvendige krav set i forhold til den samlede ønskede levetid af konstruktionerne. Derudover bør der udføres test af den valgte beton med mere i forhold til dens modstandsdygtighed med hensyn til fugtgennemtrængning, karbonatisering m.m. Formålet er samlet set, i forhold til design, armering og materialevalg, at opnå en så tæt beton som muligt med det mindst mulige antal og størrelse af revner og bestående af kompatible materialer.

11.4.4 Andet

Der bør udføres en specifik vurdering af risikoen relateret til overlagt indtrængen i depotet eller terror med henblik på at få fastlagt evt. nødvendige anbefalinger med hensyn til design, systemer til beskyttelse mod indtrængen samt alarmsystemer. Dette vil til dels være afhængigt af den konkrete placering.

Der bør i forbindelse med de undersøgelser af planter og afdækningsjord, der foretages i moniteringsperioden opstilles en procedure for, hvorledes der skal reageres, hvis der konstateres radioaktivt udslip.

Som et led i udvælgelsen af den endelige lokalitet på baggrund af de lokaliteter, som udpeges i forstudiet, foreslås det at anvende Landskabskaraktermetoden til at vurdere de enkelte lokaliteters landskabelige robusthed.

Det er væsentligt, at der udarbejdes en vedligeholdelsesplan for depotanlægget, og at depotet inkl. denne vedligeholdelsesplan er omtalt i relevante optegnelser og databaser, således at kendskabet til depotets eksistens i så vidt omfang som muligt bevares.

[80] Herunder relevant tilsætningsstoffer, såsom flyveaske, micro silica m.m.

11.5 Den videre proces

Næste del af processen vedrørende etablering af et slutdepot for dansk radioaktivt affald vil omhandle en udpegning af potentielle lokaliteter for slutdepotet, herunder udførelse af detaljerede feltundersøgelser for disse lokaliteter, samt det forberedende arbejde til anlægslov, herunder projekteringslov, VVM-undersøgelse og skitseprojektering. Der er som et led i forstudiet udarbejdet en foreløbig aktivitets- og tidsplan med henblik på at kunne forudsige og planlægge det resterende forløb og den parallelle myndighedsbehandling m.m. mest hensigtsmæssigt.

Som basis er anvendt beslutningsgrundlaget og udbudsbetingelserne, input på møde med Dansk Dekommissionering den 29. november 2010 samt resultaterne af forstudiet.

11.5.1 Tidsplan for det resterende forløb

Der foreligger som resultat af nærværende forstudie en række anbefalinger med hensyn til valg af depottype, som giver mulighed for flere kombinationer, til dels afhængigt af geologierne på de sideløbende udpegede lokaliteter. Afhængigt af det samlede resultat fra alle forstudierne kan dette medføre en yderligere begrænsning i de samlede muligheder. Det er således bl.a. endnu ikke afklaret, hvorvidt der eventuelt bør etableres flere depottyper på samme lokalitet, eller om der vil blive tale om forskellige lokaliteter til forskellige depottyper. Da depoterne er indbyrdes afhængige, forudsættes det i det følgende, at myndighedsbehandlingen vil omfatte depotet/depoterne under et.

I Tabel 11.2 er givet et forslag til tidsramme for de enkelte delaktiviteter blandt andet baseret på Dansk Dekommissionerings viden om og forventninger til den videre proces. Tidsrammen er givet under forudsætning af, at der udvælges 5-6 lokaliteter til videre vurdering[81]. Hvis der indgår 20 lokaliteter, er det skønsmæssigt anslået, at undersøgelserne af disse at vare 1 - 2 år yderligere. Tidsrammen herfor er angivet i parentes.

[81] Det er ikke muligt uden nærmere kendskab til de konkrete lokaliteter at fastslå en tidsramme for op til 20 lokaliteter, da dette vil afhænge meget af, hvordan arbejdet kan planlægges, herunder tilgængeligheden af nødvendigt undersøgelsesudstyr og kvalificeret personale i de relevante perioder.

Tabel 11.2 Forslag til tidsramme for de fremtidige delaktiviteter

Aktivitet	**Tidsramme**
Planlægning af det videre forløb	Primo 2011 – ultimo 2011
Udvælgelse af 5 - 6 (20) lokaliteter, herunder detaljerede feltundersøgelse	Primo 2011 – medio 2012
Vedtagelse af projekteringslov	Medio 2012 – primo 2013
VVM-proces og skitseprojektering	Primo 2013 – ultimo 2015 (2017)
Forslag til og vedtagelse af anlægslov	2016 (2018)
Detailprojektering og udbud	2017 (2019)
Arealerhvervelse Udførelse	2018 (2019) 2018-2019 (2020-2021)
Ibrugtagning, drift og vedligeholdelse	2019 – kontinuert (2021 -)
Overvågning	Kontinuert

11.5.2 Aktivitetsplan for det videre forløb

Der er udarbejdet et forslag til aktivitetsplan for det videre forløb af processen vedrørende etablering af et slutdepot for dansk radioaktivt affald. Aktiviteterne vil omfatte:

- Planlægning af det videre forløb
- Udpegning af mulige lokaliteter, herunder detaljerede feltundersøgelser Projekteringslov
- VVM-proces og skitseprojektering
- Forslag til og vedtagelse af anlægslov
- Detailprojektering og udbud
- Arealerhvervelse
- Udførelse
- Drift og vedligeholdelse samt overvågning

Detaljerne for denne plan fremgår af Appendiks J tillige med et økonomisk overslag for aktiviteternes gennemførelse.

12 References

The reference list includes references for both the main report and all appendices.

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13 Glossary and abbreviations

Term	Abbreviation	Explanation
@risk		A Microsoft Excel based software program that can be used for e.g. calculating risk results based on probability parametric input
ADR		European agreement about transport on roads.
ALARA		As Low As Reasonable Achievable. A principle meaning that radiation doses to the personnel involved in the waste handling process as well as doses to the public in the long term perspective must be evaluated and minimised.
Alkali-carbonate reaction (ACR)		The alkali-carbonate reaction (ACR) is a process suspected for the degradation of concrete containing dolomite aggregate. Alkali from the cement reacts with the dolomite crystals present in the aggregate inducing the production of brucite (MgOH)2 and calcite (CaCO3). Brucite is responsible for the volumetric expansion after de-dolomotisation of the aggregate, due to absorption of water.
Alkali-silicate reaction (ASR)		The Alkali-Silica Reaction (ASR) is a reaction which occurs over time in concrete between the highly alkaline cement paste and reactive non-crystalline (amorphous) silica, which is found in many common aggregates. The ASR reaction is the same as the Pozzolanic reaction which is a simple acid-base reaction between calcium hydroxide, also known as Portlandite, or (Ca(OH)2), and silicic acid (H4SiO4 or Si(OH)4). This reaction causes the expansion of the altered aggregate by the formation of a swelling gel of Calcium Silicate Hydrate (CSH). This gel increases in volume with water and exerts an expansive pressure inside the material causing spalling and loss of strength of the concrete.
ASR		Above surface repository
Backfill		Material used to fill in the space in between containers to ensure the physical stability and avoid settlements or collapse within or around the repository. See also fill.

Term	Abbreviation	Explanation
Biosphere		Covers in our terminology the ecosystems involved in the risk assessment, i.e. soils, surface water etc.
BORE		Borehole
CCA		Coarse Control Arms were been used in DR3 for the control of the neutron flux. They consist of cadmium panels in stainless steel frames and have been highly activated.
Conditioning		Conditioning is the process of treating and packaging the waste for storage. Examples of treatment are: Solidifying liquids by mixing into concrete, mixing concentrate from water treatment into bitumen.
DD		Danish Decommissioning
Delayed ettringite formation (DEF)		Delayed ettringite formation (DEF) is a form of an internal sulfate attack in concrete. The associated reactions lead to ettringite formation, which is associated with expansion and thus cracking of the concrete. A number of factors such as concrete composition, curing conditions and exposure conditions influence the potential for DEF. The risk of DEF can be impeded through use of a temperature requirement of maximum 65°C during hardening.
DFN		Discrete fracture networks
Distribution coefficient	K_D	The fraction of solutes that are adsorbed related to solutes in water.
Early age cracking		Early-age cracking can be a significant problem in concrete. Volume changes in concrete will drive tensile stress development when they are restrained. Cracks can develop when the tensile stress exceeds the tensile strength, which is generally only 10% of the compressive strength. At early ages, this strength is still developing while stresses are generated by volume changes. Controlling the variables that affect volume change can minimize high stresses and cracking. The volume of concrete begins to change shortly after it is cast. The most important causes for volume change are: Chemical Shrinkage Autogenous Shrinkage Creep Swelling Thermal Expansion
Effective porosity	θ_e	A hydrogeological parameter that determines the effective transport part of the formation and thereby the transport velocity of dissolved species
EPM		Equivalent porous medium

Term	Abbreviation	Explanation
Fill		Material used to fill in cavities inside containers to ensure the physical stability and avoid settlements or collapse within or around the repository. Fill may also serve the purpose of shielding of high gamma radiation, which can be important to keep the doses to the personnel as low as possible during storage and handling of the waste packages and during operation of the repository. See also backfill.
Fixed head boundary condition		A so-called Dirichlet (or first-type) boundary condition that specifies the values a solution needs to take on the boundary of the domain.
FN diagram		A type of diagram often used to represent the societal consequences in terms of fatalities from accidents. The x-axis represents the number of fatalities (N) per accident. The y-axis shows the frequency of accidents with N or more fatalities.
Geosphere		Covers in our terminology the geological deposits below the soil zone including the groundwater system.
HDPE		High Density Poly Ethylene
Horizontal hydraulic conductivity	k_h	A hydrogeological parameter that determines the hydraulic gradient and flow/transport in the horizontal direction of the formation
Hydraulic conductivity	K	Hydraulic conductivity is a property of soil or rock that describes the ease with which water can move through pore spaces or fractures. It depends on the intrinsic permeability of the material and on the degree of saturation. The dimension for hydraulic conductivity is length per time, e.g. m/s. Given the value of hydraulic conductivity for a subsurface system (k), the permeability (κ) can be calculated as: $\kappa = k \cdot \frac{\mu}{\rho \cdot g}$ where: • μ is the viscosity of the fluid, kg/m/s • ρ is the density of the fluid, kg/m3 • g is the acceleration due to gravity, m/s2
In-flight phase		The phase of a flight where the aircraft has reached it's cruising height and has not started the descent
Isotropic		Uniformity in all directions.
Level indicator		An instrument in for example a tank containing liquid that indicates the amount of liquid in the tank

Term	Abbreviation	Explanation
Lost formwork		Lost formwork is based on customised precast concrete elements, which are used as formwork. The elements will not be removed after curing (as normal formwork), but remain an integral part of the final structure
MDR		Medium deep repository
Monte Carlo simulations		A stochastic modelling approach in which a large number of realisations e.g. groundwater flow simulations with equal probability are used to determine probability density functions of the results and thereby the relation between input parameter variability and output variability.
NSR		Near surface repository
Packaging		Packaging is the items and material used to pack the waste for transportation and storage.
Paroc panels		Paroc panels is the commercial name for panels manufactured in Finland and mainly distributed throughout Northern, Western and Eastern Europe. Paroc panels are steel-faced sandwich panels with a core of stone wool for facades, partitioning walls and ceilings. Paroc panel solutions provide fire safety with relatively short construction time and good overall economy.
Pasquill weather class		Used in gas and dust dispersion models to distinguish between different types of weather stability conditions. Class A is used for unstable (high turbulence) conditions and Class F for stable conditions.
Permeability	κ	Permeability is a measure of the ability of a porous material to transmit fluids. The dimension for permeability is area, e.g. m2.
Porosity	θ	Porosity is a measure of the void spaces in a material, and is a fraction of the volume of voids over the total volume (between 0–1). The effective porosity is most commonly considered to represent the porosity of a rock or sediment available to contribute to fluid flow through the rock or sediment. The total porosity is the total void space and as such includes isolated pores and the space occupied by e.g. clay-bound water.
Richter scale		Scale for measuring the intensity of an earthquake

Term	Abbreviation	Explanation
Sprayed concrete lining (SCL)		Sprayed concrete lining (SCL) is a stabilisation lining for rock faces consisting of a sprayed concrete layer, also known as shotcrete. Shotcrete is concrete (or sometimes mortar) conveyed through a hose and pneumatically projected at high velocity onto a surface, as a construction technique. Shotcrete undergoes placement and compaction at the same time due to the force with which it is projected from the nozzle. It can be impacted onto any type or shape of surface, including vertical or overhead areas. SCL can be reinforced by conventional steel rods, steel mesh and/or fibers.
Tension piles		Tension piles can be used to resist uplift forces at foundations. The resulting forces are transmitted to the soil along the embedded length of the pile. The resisting force can be increased in the case of bored piles by under-reaming.
Tremie concrete		In the Tremie Concrete method, concrete is placed below water level through a pipe, the lower end of which is kept immersed in fresh concrete so that the rising concrete from the bottom displaces the water without washing out the cement content.
TSP		Top shield plug from DR3.
TSR		Top shield ring from DR3.
Vertical hydraulic conductivity	k_v	A hydrogeological parameter that determines the hydraulic gradient and flow/transport in the vertical direction of the formation
Waste item		An individual waste object. Examples of waste items are: A source, a piece of metal, a plastic bag of dust, a piece of concrete.
Waste unit		A container or drum designed for transportation of waste to the repository and for storage. The waste unit will contain one or more waste items and possibly also fill.

Appendix A: Waste to be disposed, details

A.1 Distribution of nuclides considering waste type

The waste is grouped in accordance with the types indicated in Table 2.1 and Table 2.2 in the main report. In general this grouping is in accordance with the grouping used in Danish Decommissioning (2010a), however there are a few differences.

For each waste type the total activity indicated in Table 2.1 and Table 2.2 (valid for the year 2008) has been distributed on the nuclides found to be present in the specific waste type. The presence and fractions used are based on information received from Danish Decommissioning. For each waste type the specific origin and reference of the information has been registered in the background working reports. Further assumptions and assessments made in the present report are explained.

Danish Decommissioning (2010d) indicates the total activity inventory in the waste. Danish Decommissioning (2010a) indicates activities under the heading of parent radionuclides, i.e. the table does not indicate activity of specific daughters. The information in Danish Decommissioning (2010a) is used to estimate the fractions of the total activity levels indicated in Danish Decommissioning (2010d) to attribute to the indicated parent nuclides and daughters. However, Danish Decommissioning (2010b) specifies activity levels for both parent and individual daughters.

The distributions indicated below are for the use of the analyses in the present pre-feasibility study. In a future safety assessment for the actual repository, the data has to be reassessed.

For each waste type the radionuclides and the activity estimated for the year 2008, and the time development due to decay is illustrated including formation of daughters.

A.1.1 Decommissioning of DR1, DR2 and DR3

For the waste generated by decommissioning of DR1, DR2 and DR3 the nuclide distribution has been based on the following:

- Information in Danish Decommissioning (2009a) concerning waste from decommissioning of DR1; however, this reference does not contain much data on specific nuclides
- Information in Danish Decommissioning (2009b) and Ølgaard (2003b) describing the waste generated by decommissioning of DR2
- Information in Danish Decommissioning (2010a) about activity of specific nuclides in the waste.

The decommissioning waste consists of the following types:

Graphite (type 1)

The waste originates primarily from DR2 and DR3 and only a minor fraction originates from DR1.

The estimated 17 tons of graphite mainly consists of so-called 'stringers' (i.e. rectangular bars/items). The graphite has served to absorb neutrons. Graphite stringers in sections no longer than half metre from DR2 have been packed in steel containers. This is planned also to be the case for the graphite waste generated during decommissioning of DR3. The material from DR1 has not yet been packed.

It has been decided that graphite stringers from the central part of the thermal column shall be annealed before final packing to dissipate the Wigner-energy accumulated. Thus a potential sudden energy release and temperature rise at a later time is prevented.

Activity in 2008

The activity of the waste is:

Short lived β/γ-nuclides:	4,000 GBq
Long lived β/γ-nuclides:	120 GBq
α-nuclides:	-

Short lived β/γ-nuclides

The short lived β/γ -nuclides in this waste are considered to be:

- H-3, Co-60, Eu-152, Eu-154

Assessment of the type and distribution of the short lived β/γ-nuclides is based on the activity levels indicated in Ølgaard (2003b) and in Danish Decommissioning (2010a). Ølgaard (2003b) indicates that the dominating europium activities were unexpected, while only a limited amount of activity from Co-60 was observed. The active europium is considered to be due to impurities in the graphite.

The relative distributions and activity of the nuclides are considered to be:

	H-3	Co-60	Eu-152	Eu-154
Fraction	0,70	0.003	0.261	0.036
GBq	2,800	12	1,044	144

Long lived β/γ-nuclides

The long lived β/γ-nuclides in this waste are considered to be:

- C-14

	C-14
Fraction	1
GBq	120

Activity versus time

The activity of this waste versus time is presented in Figure A.1.

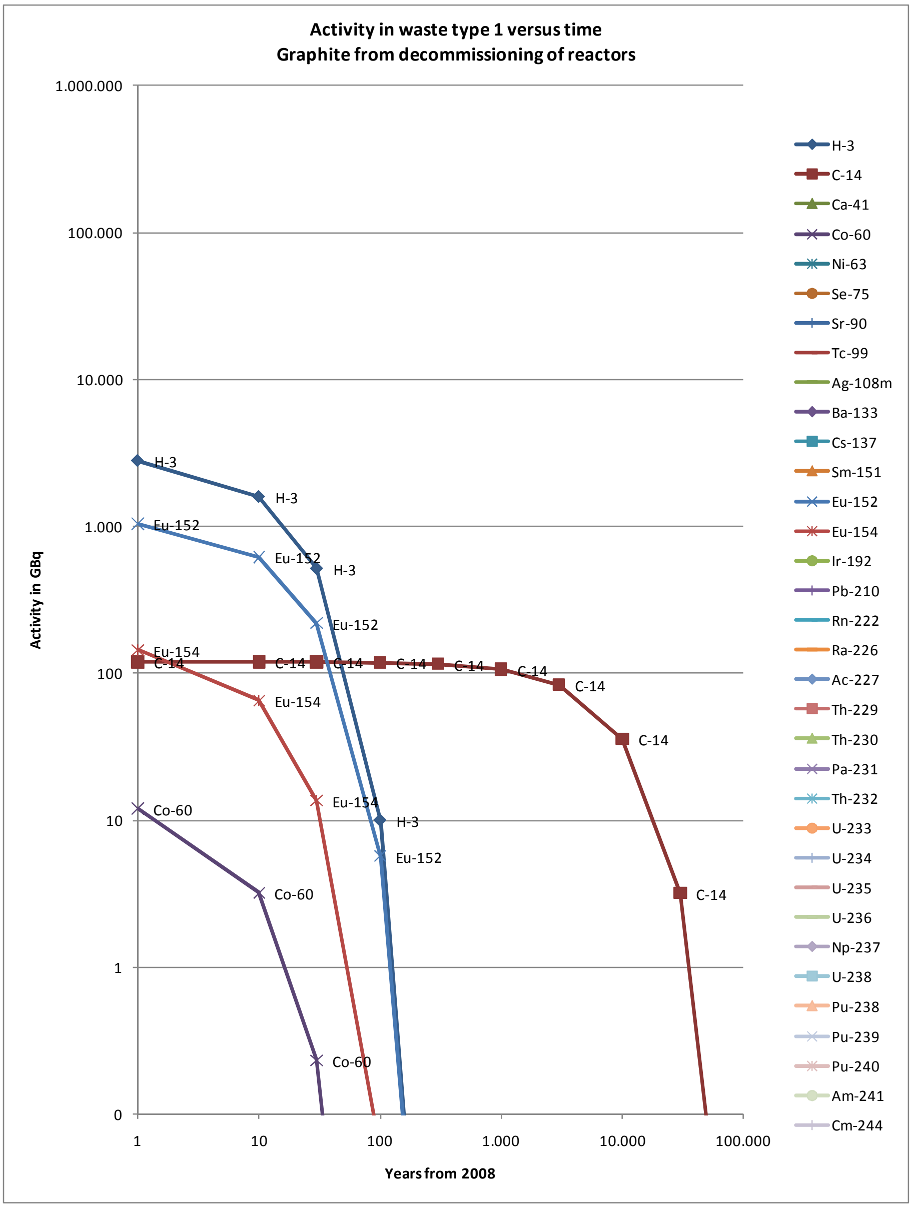


Figure A.1: Estimated activity in GBq versus time (from 2008) for graphite waste generated during decommissioning of DR1, DR2 and DR3

Aluminium (type 2)

The reactor vessels are made out of aluminium. The estimated 17 tons of aluminium thus consist of sheets and pipes in various diameters. Large-sized pipes (e.g. with diameter larger than 75-100 mm) have been cut open. Small-sized pipes have not been opened.

According to the inventory provided by Danish Decommissioning, some of the aluminium waste has already been packed into three ISO containers.

Activity in 2008

The activity of this waste is:

Short lived β/γ-nuclides: 20,400 GBq
Long lived β/γ-nuclides: -
α-nuclides: 0.7 GBq

Short lived β/γ-nuclides

The short lived β/γ-nuclides in this waste are considered to be:

- Co-60, Sr-90, Cs-137, Eu-152

The type and distribution of the short lived β/γ-nuclides is in line with the activity levels indicated in Ølgaard (2003c), table 8.4, "Aluminium sample" from the thermal column of DR2. The table indicates that Co-60 is the dominating nuclide. Table 4.23a in Danish Decommissioning (2010a) indicates a high and also dominating activity in the decommissioning waste from DR3 due to Co-60. Danish Decommissioning (2010a) does not split the waste types 1, 2, 3 and 4: however, based on Ølgaard (2003b) and the overall activity level of the aluminium waste, it is considered reasonable to assume, that a high activity due to Co-60 is associated with the aluminium waste.

The relative distributions and activity of the nuclides are considered to be:

	Co-60	Sr-90	Cs-137	Eu-152
Fraction	0.97	0.005	0.005	0,02
GBq	19,788	102	102	408

α-nuclides

The α-nuclides in this waste are considered to be:

- Pu-238, Pu-239, Pu-240, Am-241

The low activity due to α-nuclides is distributed based on the averages indicated in Danish Decommissioning (2010a) for waste from decommissioning of all three reactors.

The relative distributions and activity of the nuclides are considered to be:

	Pu-238	Pu-239	Pu-240	Am-241
Fraction	0.323	0.338	0.016	0.323
GBq	0.2	0.2	0.01	0.2

<u>Activity versus time</u>
The activity of this waste versus time is presented in Figure A.2.

Figure A.2 Estimated activity in GBq versus time (from 2008) for aluminium waste generated during decommissioning of DR1, DR2 and DR3

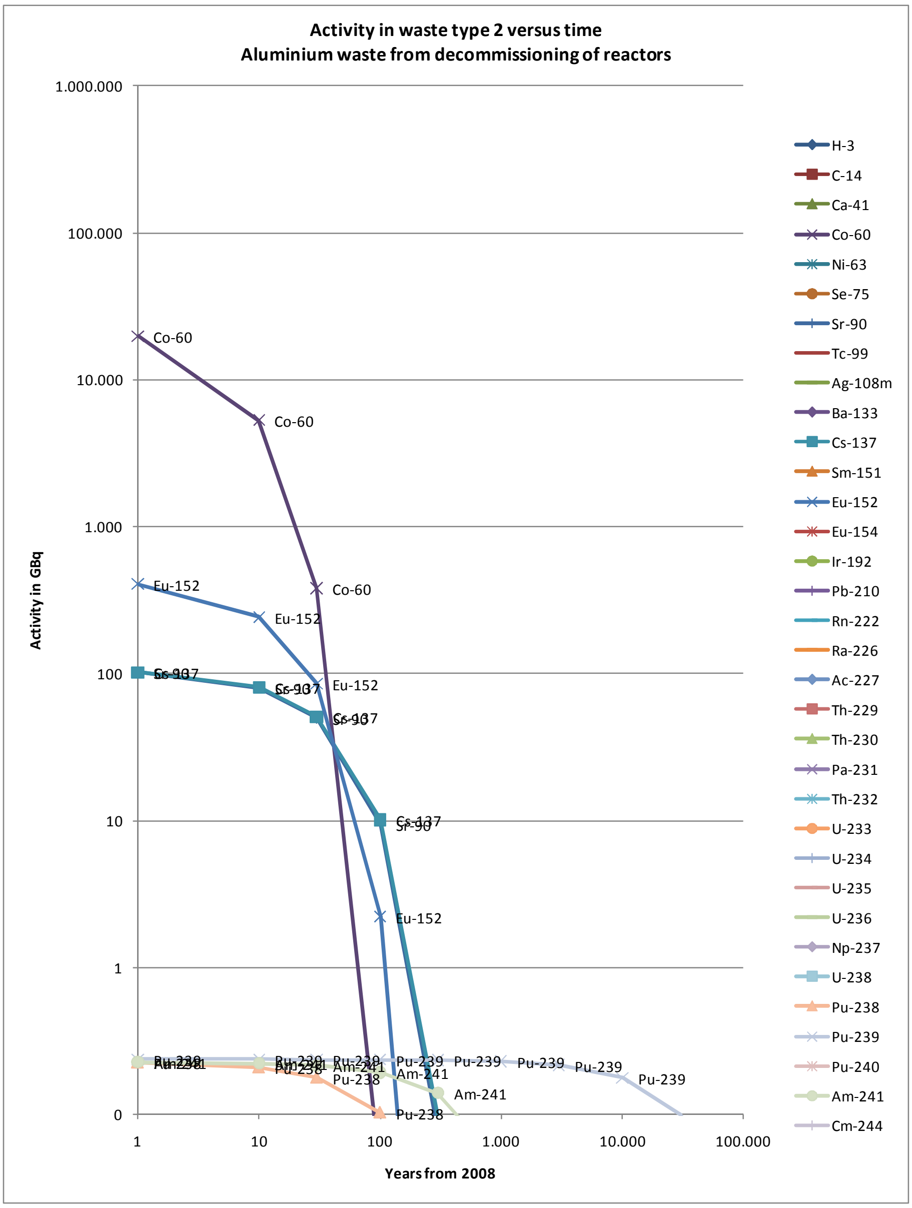


Figure A.2 Estimated activity in GBq versus time (from 2008) for aluminium waste generated during decommissioning of DR1, DR2 and DR3

Steel and lead (type 3) The waste is and will be sorted in two categories "lead" and "steel". In general, waste items in the category "steel" do not contain lead, i.e. they are likely to be different from the "lead" with regard to the nuclide content.

Approximately 50-70 of the estimated overall 345 tons of waste will consist of lead. The lead will primarily be in the form of bricks (approximately 12-15 kg each). At present the waste only includes minor amounts of lead primarily used for shielding (bricks) and the "lead nose" from DR2. This waste is mainly packed in steel containers. Some of the bricks are currently stored on pallets. Future waste generated during decommissioning of DR3 will include lead in larger amounts. Six ISO containers are estimated for this waste fraction.

The steel waste is mostly slabs and pipes, but items like vacuum cleaners (including dust) are also included. Smear samples from the decommissioning process indicate that the steel waste from vessels contains Co-60, Cs-137 and U-235, Danish Decommissioning (2008c). The presence of uranium is due to contamination during tests. Some steel pipes contain the nuclides typical for activated barite concrete (see type 4), i.e. Ba-133, Eu-152 and smaller amounts of Co-60.

The steel (approximately 728 m^3) consist of various items that will be cut into suitable sizes and then packed as tight as possible in containers. The choice of container depends on: The dose rate, the overall activity, activity due to long lived nuclides, and the contents of very mobile nuclides. Approximately 95 (ISO) containers are needed for this waste fraction.

<u>Activity in 2008</u>

The overall activity is:

Short lived β/γ-nuclides:	66,600 GBq
Long lived β/γ-nuclides:	18,000 GBq
α-nuclides:	1.5 GBq

<u>Short lived β/γ-nuclides</u>
The short lived β/γ-nuclides in this waste are considered to be:

- H-3, Co-60, Sr-90, Ba-133, Cs-137, Eu-152

Information on the short lived β/γ-nuclides is retrieved from Ølgaard (2003b) and from Danish Decommissioning (2010a). Danish Decommissioning (2010a) indicates a high activity in the decommissioning waste from DR3 due to H-3 and Co-60. Danish Decommissioning (2010a) does not split the waste types 1, 2, 3 and 4; however, based on Ølgaard (2003b) and the overall activity level of the steel waste, it was considered reasonable to assume, that the high activity due to H-3 and Co-60 is associated with the steel waste. Danish Decommissioning (2008c) supports this information, since section 4 indicates that H-3 was identified in scrape off from the inside of the cooling system of DR2.

The relative distributions and activities of the nuclides are estimated to:

	H-3	Co-60	Sr-90	Ba-133	Cs-137	Eu-152
Fraction	0.54	0.4	0.01	0.03	0.01	0.01
GBq	35,964	26,640	666	1,998	666	666

Long lived β/γ-nuclides
The long lived β/γ-nuclides in this waste are considered to be:

- Ni-63, Ag-108m

Information on the long lived β/γ-nuclides is based on Danish Decommissioning (2010a), waste from decommissioning of DR1, DR2 and DR3. This reference indicates a large activity due to Ni-63 and a small amount of Ag-108m in waste from DR2 and DR3.

The relative distributions and activities of the nuclides are estimated to:

	Ni-63	Ag-108m
Fraction	0.999	0.001
GBq	17,982	18

α-nuclides
The α-nuclides in this waste are considered to be:

- Pu-238, Pu-239. Pu-240, Am-241

The low activity due to the α-nuclides is distributed based on the averages indicated in Danish Decommissioning (2010a) for waste from decommissioning of all three reactors.

The relative distributions and activities of the nuclides are estimated to:

	Pu-238	Pu-239	Pu-240	Am-241
Fraction	0.323	0.338	0.016	0.323
GBq	0.5	0.50	0.02	0.5

Activity versus time
The activity of the waste versus time is presented in Figure A.3.

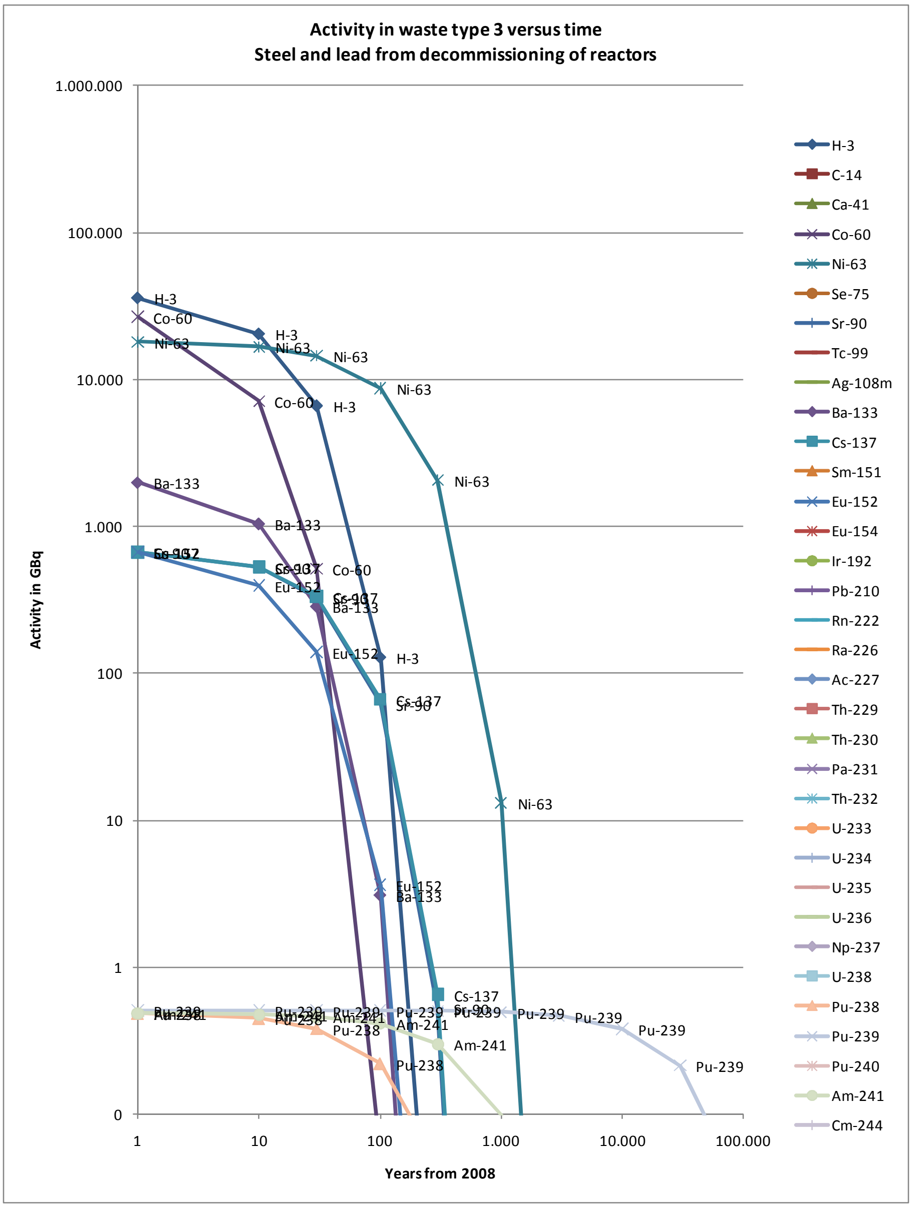


Figure A.3: Estimated activity in GBq versus time (from 2008) for steel and lead waste generated during decommissioning of DR1, DR2 and DR3

Concrete (type 4)

The concrete consist partly of ordinary concrete and partly of heavy concrete (with aggregate of barite or steel and an approximate density of 3.9 t/m^3). The DR3 concrete includes 9 m^3 of "steel ball concrete".

Some heavy concrete waste from decommissioning of DR2 contains aggregated barium sulphate (barite). The barite concrete includes nuclides typical for activated barite concrete, i.e. Ba-133, Eu-152 and smaller amounts of Co-60.

The heavy concrete from decommissioning of DR1 and DR3 contains iron. "Steel" contains nickel. Ni-63 is produced by neutron irradiation of Ni-62, one of the most abundant stable isotopes of nickel.

During decommissioning, minor amounts of concrete dust from cutting etc will be generated. This will first be packed in plastic bags. It is estimated that a total of 230 ISO containers are needed for the concrete with the lowest activity and for the concrete gravel/dust.

Since most of the concrete is on the outside of the reactor tank at DR3, the innermost material has the highest activity, whereas the outermost material can be released. Therefore, the total amount is yet uncertain. However, the current estimate of 1,129 m^3 for disposal is likely to be in the lower end, for which reason a span of 50% is added to be on the safe side.

<u>Activity in 2008</u>

The activity of the waste is:

Short lived β/γ-nuclides: 570 GBq
Long lived β/γ-nuclides: 38,000 GBq
α-nuclides: 108 GBq

<u>Short lived β/γ-nuclides</u>
The short lived β/γ-nuclides in this waste are considered to be:

- H-3, Co-60, Ba-133, Eu-152, Eu-154

Information on the presence and distribution of short lived β/γ-nuclides is retrieved from Ølgaard (2003b). Ølgaard (2003b) indicates that "there may also be some tritium", however the document does not include information on activity level. Danish Decommissioning (2001a) indicates a noticeable activity in the decommissioning waste from DR3 due to Ba-133 and Eu-152. Danish Decommissioning (2010a) does not split the waste types 1, 2, 3 and 4; however, based on Ølgaard (2003b) and the overall activity level of the concrete waste, it was considered reasonable to assume, that the activity due to Ba-133 and Eu-152 is associated with the concrete waste. Other nuclides would be present in smaller amounts (and indeed small amounts compared to the activity of these nuclides in the other 3 types of decommissioning waste).

The relative distributions and activities of the nuclides are estimated to:

	H-3	Co-60	Ba-133	Eu-152	Eu-154
Fraction	0.15	0.02	0.66	0.16	0.01
GBq	86	11	376	91	6

<u>Long lived β/γ-nuclides</u>
Information on the presence of long lived β/γ-nuclides is based on Danish Decommissioning (2010a). The short lived β/γ-nuclides in this waste are considered to be:

- Ni-63, Ca-41

Danish Decommissioning (2010a) indicates a large activity due to Ni-63 in waste from DR2 and DR3. This is assumed to originate from neutron activation of steel balls in the concrete.

The reference also indicates the presence of C-14, Cl-36, Ni-59, Mo-93, Nb-94, and Ag-108m, however in very small amounts.

	Ni-63	Ca-41
Fraction	0.9995	0.0005
GBq	37,981	19

<u>α-nuclides</u>
The α-nuclides in this waste are considered to be:

- Pu-238, Pu-239. Pu-240, Am-241

The activity due to α-nuclides is distributed based on the averages indicated in Danish Decommissioning (2010a) for waste from decommissioning of all three reactors. The relative distributions and activities of the nuclides are estimated to:

	Pu-238	Pu-239	Pu-240	Am-241
Fraction	0.323	0.338	0.016	0.323
GBq	35	37	2	35

<u>Activity versus time</u>
The activity of this waste versus time is presented in Figure A.4.

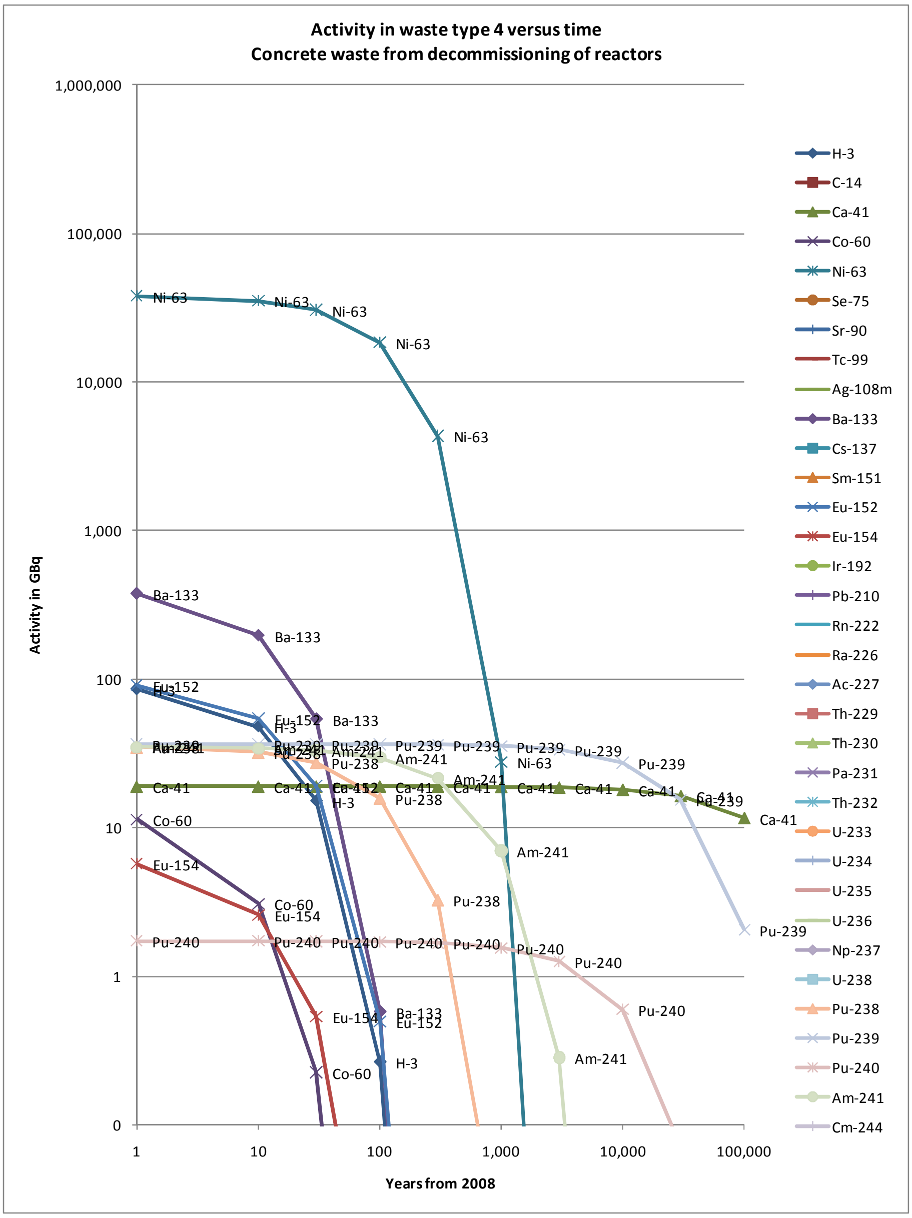


Figure A.4: Estimated activity in GBq versus time (from 2008) for concrete waste generated during decommissioning of DR1, DR2 and DR3

A.1.2 Decommissioning of HotCell

Stainless steel and lead (type 5)

It is assumed that 15 tons of lead from the Hot Cell will be decontaminated and released. Therefore, the previous estimate of three tons made by Danish Decommissioning is considered too low. The waste includes a few approximately 2 x 2 m steel shutter doors that will require separate handling due to their size if they are to be disposed together with the other waste. However, it is assumed that the shutter doors will be decontaminated and are therefore not to be included in the waste for disposal.

The activity is not indicated in Table 2.1 as it is not known at present. Danish Decommissioning assumes that a successful decontamination will render the activity insignificant.

Decontamination of waste items using sandblasting will cause the activity to be transferred to the blasting material. This material is included in waste type 8.

Concrete (type 6)

The concrete waste from decommissioning of the HotCell facility is packed in steel containers.

The activity is not indicated in Table 2.1, as it is not known at present. Danish Decommissioning judges that the volumes and activity will be insignificant.

Waste from attic (type 7)

The exact contents of the waste from the attic above the HotCell facility are not known. The waste will include: filters and piping from parts of the ventilation system as well as various slightly contaminated equipment that was placed there when the facility was shut down. The waste that cannot be decontaminated is to be disposed of in the repository. It is expected, however, that most items can be decontaminated (resulting in the activity being transferred to the cleaning agent).

The activity is not indicated in Table 2.1, as it is not known at present. Danish Decommissioning finds that the activity will be insignificant compared to other types of waste from the facility.

Secondary waste (type 8)

This type of waste includes materials generated during the decontamination of the Hot Cell, e.g. sand and paint dust from sandblasting of the interior surfaces of the cells, and it contains PCB-paint. The existing waste has been put into 210 l steel drums with concrete lining. It is estimated that the total amount of waste will amount to 100 - 120 full drums. Danish Decommissioning plans to load the drums in steel containers (with six drums in each container). 20 steel containers are needed.

The material will be contaminated by predominantly Cs-137 and possibly a few Co-60 pellets.

<u>Activity in 2008</u>
The estimated activity is:

Short lived β/γ-nuclides:	3,000 GBq
Long lived β/γ-nuclides:	1 GBq
α-nuclides:	160 GBq

The relative distributions of the nuclides and the activity are based on information in Danish Decommissioning (2001a), since the total activity levels here are in good agreement with the levels indicated above.

Short lived β/γ-nuclides
The short lived β/γ-nuclides in this waste are considered to be:

- Co-60, Sr-90, Cs-137, Eu-152, Eu-154

	Co-60	Sr-90	Cs-137	Eu-152	Eu-154
Fraction	0.02	0.4	0.57	0.01	0
GBq	60	1,200	1,710	30	0

Long lived β/γ-nuclides
The long lived β/γ-nuclides in this waste are considered to be:

- Tc-99

	Tc-99
Fraction	1
GBq	1

α-nuclides
The α-nuclides in this waste are considered to be:

- Pu-238, Pu-239, Pu-240, Am-241, Cm-244

	Pu-238	Pu-239	Pu-240	Am-241	Cm-244
Fraction	0.37	0.06	0.10	0.40	0.07
GBq	59	10	16	64	11

Activity versus time
The activity of the waste versus time is presented in figure A.5.

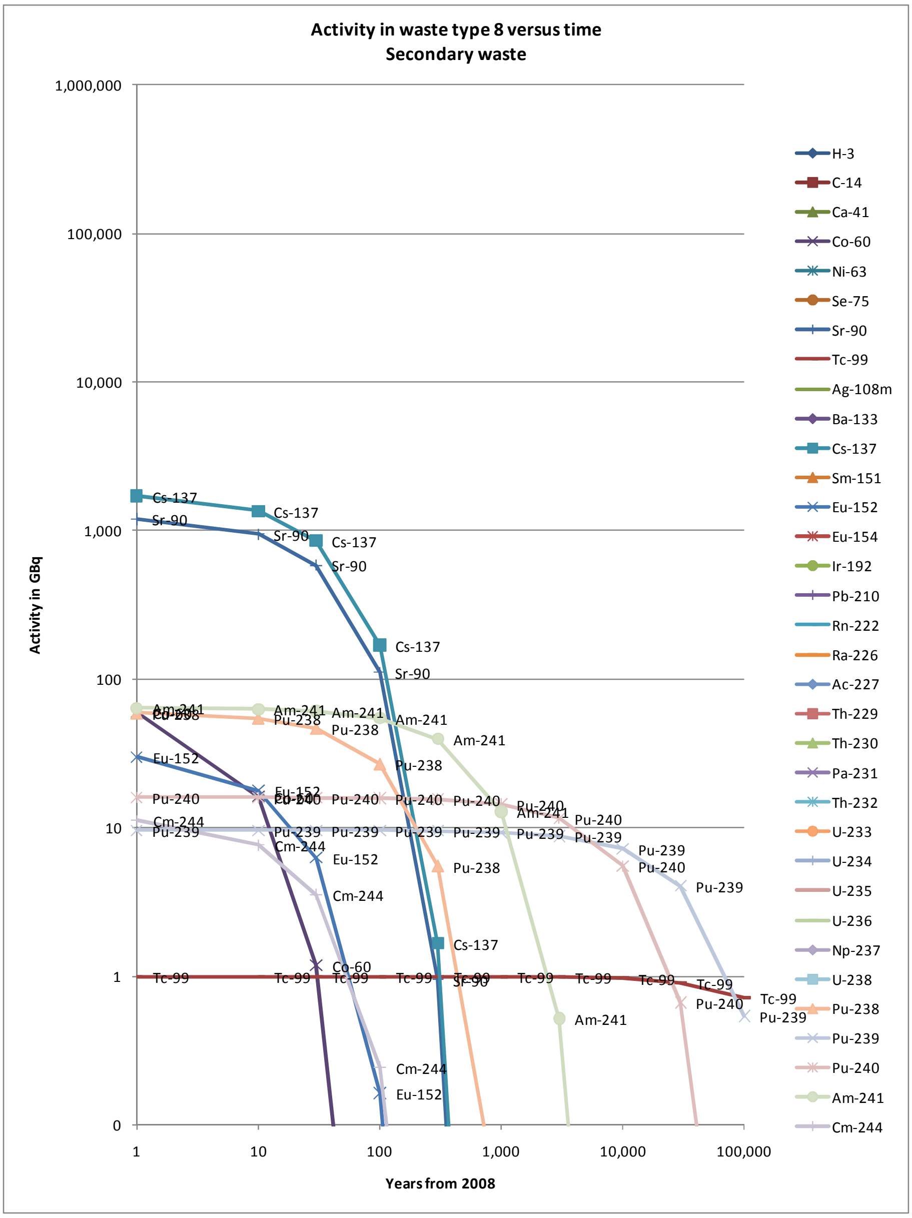


Figure A.5: *Estimated activity in GBq versus time (from 2008) for secondary waste generated during decommissioning of HotCell*

A.1.3 Existing waste with low activity

Waste from water treatment plant (type 9)

Evaporator residues (salts and sludge) from the waste water evaporator plant are packed in 210 l concrete lined steel drums. Older drums (398 items) are damaged by corrosion. Newer drums (about 700 items) have been filled with bituminized evaporator residues. This introduces a problem, since the bitumen product is hydroscopic and swells causing the drums to deform and open. Damaged drums are to be repacked and backfilled.

<u>Activity in 2008</u>
The activity is:

Short lived β/γ-nuclides: 1,800 GBq
Long lived β/γ-nuclides: 0.5 GBq
α-nuclides: 130 GBq

The total activity indicated in Danish Decommissioning (2001a) for the entries "Inddamp konc.", "Bit næser", "Bituminiseret" and "20 Cs højeste" is in agreement with the activity levels indicated above. Thus it is assumed that the distribution of nuclides given in Danish Decommissioning (2010a) is representative. However, in accordance with Danish Decommissioning (2010d) the waste containing Sm-151 is included in waste type 11.

The information given indicates that some of the drums contain small amounts of long lived nuclides. Danish Decommissioning estimates that 10 % of the drums may contain up to 10 g of active uranium plus 2 g of transuranic elements. This corresponds to a total activity of these elements of about 0.2 GBq. This is of the same order of magnitude as the activity of uranium indicated in Danish Decommissioning (2010a).

The nuclides in the waste are considered to be:

<u>Short lived β/γ-nuclides</u>
The short lived β-nuclides in this waste are considered to be:

- Co-60, Sr-90, Cs-137, Eu-154

The relative distributions and activity of the nuclides are considered to be:

	Co-60	Sr-90	Cs-137	Eu-154
Fraction	0.009	0.39	0.562	0.039
GBq	16	702	1,012	70

<u>Long lived β/γ-nuclides</u>
The long lived β/γ-nuclides in this waste are considered to be:

- Tc-99

The relative distributions and activity of the nuclides are considered to be:

	Tc-99
Fraction	1
GBq	0.5

<u>α-nuclides</u>
The α-nuclides in this waste are considered to be:

- Pu-238, Pu-239, Pu-240, Am-241, Cm-244 and very small amounts of U-234, U-235, U-238

The relative distributions and activity of the nuclides are considered to be:

	Pu-238	Pu-239	Pu-240	Am-241	Cm-244
Fraction	0.35	0.04	0.06	0.42	0.13
GBq	45	5	8	54	17

	U-234	U-235	U-238
Fraction	0.003	0.0006	0.003
GBq	0.4	0.09	0.4

<u>Activity versus time</u>
The activity of the waste versus time is presented in Figure A.6.

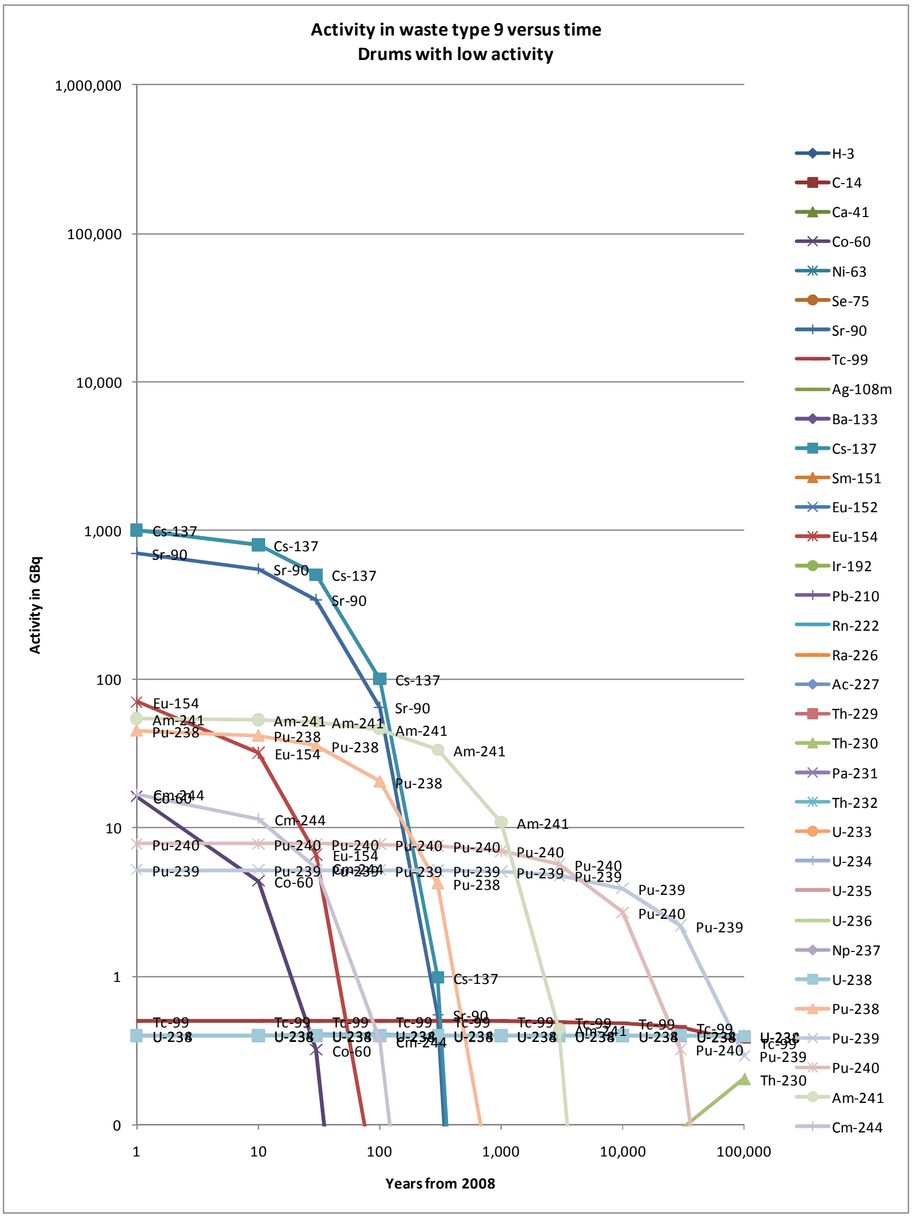


Figure A.6: Low activity waste. Estimated activity in GBq versus time (from 2008) for waste from water treatment plant (type 9) in drums

Compressed waste and soil (type 10)

388 of approximately 4,400 drums are filled with contaminated soil (soil contaminated with Sr-90 and traces of Cs-137 according to Danish Decommissioning (2001a). strontium etc.). Approximately 65 % of these drums have been repacked into 280 l drums.

The rest of the drums contain various kinds of waste that has been filled into 210 l concrete-lined drums and compacted before the drums were closed. This part of the waste may contain any type of contaminated equipment or materials, i.e. aluminium, steel, glass plastic, rubber, rags, paper. About 300 of these drums contain long-lived nuclides most likely originating in smoke detectors, i.e. small Am-241 sources. Items are likely to be damaged as the waste has been compressed to reduce the volume.

The overall level of activity is in good agreement with the level indicated in Danish Decommissioning (2010a) under the headings "Presset lavakt." and "Sr jord" and the nuclide distribution is assumed to be as indicated in this reference. However, in accordance with Danish Decommissioning (2010) the waste containing Sm-151 is included in waste type 3.

Activity in 2008
The total activity in the waste is:

Short lived β/γ-nuclides: 2,600 GBq
Long lived β/γ-nuclides: 0.6 GBq
α-nuclides: 170 GBq

Short lived β/γ-nuclides
The short lived β/γ-nuclides in this waste are considered to be:

- Co-60, Sr-90, Cs-137, Eu-154

The relative distributions and activity of the nuclides in 2008 was considered to be:

	Co-60	Sr-90	Cs-137	Eu-154
Fraction	0.24	0.31	0.42	0.03
GBq	624	806	1092	78

Long lived β/γ-nuclides
The long lived β/γ-nuclides in this waste are considered to be:

- Tc-99

The relative distributions and activity of the nuclides in 2008 was considered to be:

	Tc-99
Fraction	1
GBq	0.6

α-nuclides
The α-nuclides in this waste are considered to be:

- Pu-238, Pu-239, Pu-240, Am-241, Cm-244

The relative distributions and activity of the nuclides in 2008 was considered to be:

	Pu-238	Pu-239	Pu-240	Am-241	Cm-244
Fraction	0.35	0.04	0.06	0.42	0.13
GBq	60	7	10	72	22

<u>Activity versus time</u>
The activity of the waste versus time is presented in Figure A.7.

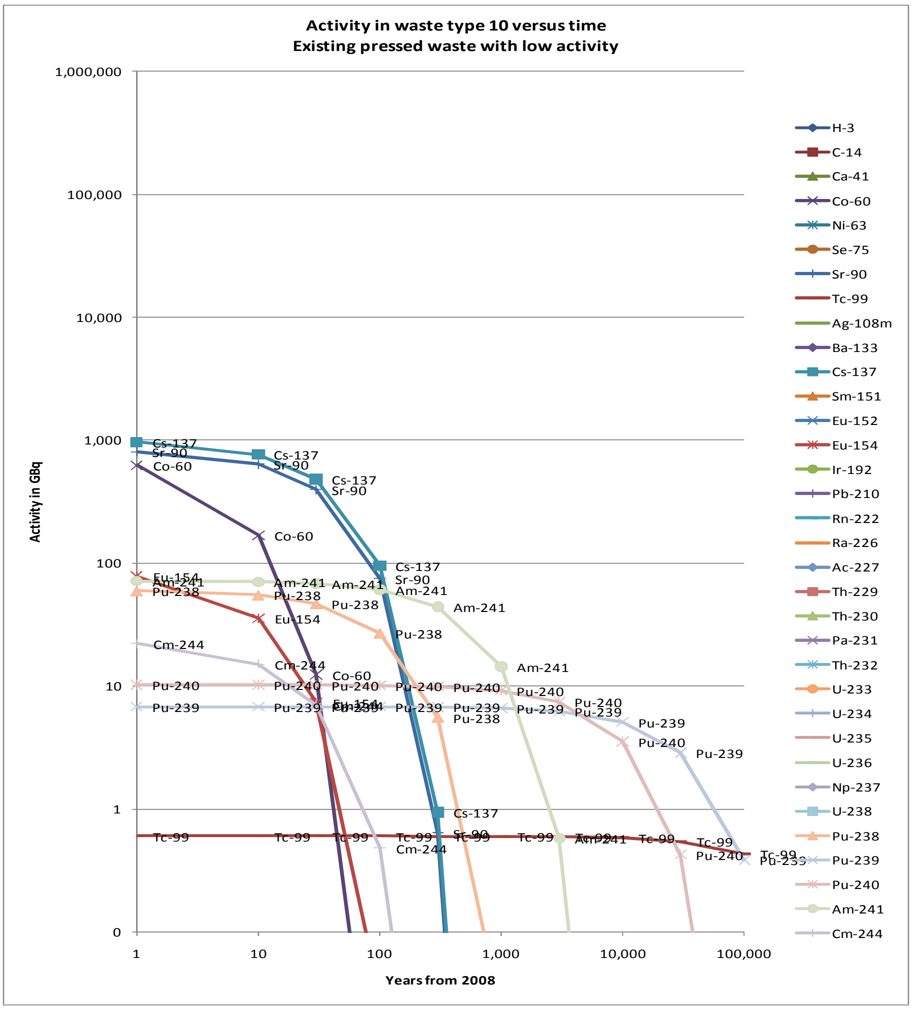


Figure A.7: Low activity waste. Estimated activity in GBq versus time (from 2008) for pressed waste (type 10) in drums

A.1.4 Existing medium activity waste

Waste from DR3 (type 11)

This waste consists of 17 stainless steel containers for CCAs, 40 units of 210 l galvanized steel drums with concrete lining, and of two large waste items (TSP unit and TSR unit).

The stainless steel containers contain CCAs (the 2.1 m control arms from DR3 and the 0.8 m underplugs carrying the arms). The arms represent 680 kg of activated stainless steel and 150 kg of activated cadmium. Both materials are active and contain mainly the long-lived isotope Ni-63, which does not cause any gamma radiation. However, they do also contain a significant amount of Co-60 (almost 150 GBq in year 2010) and some Cl-36. The underplugs are made from stainless steel and cast with heavy concrete.

At present the level of activity is rather high in particular due to β/γ-emission from Co-60. Thus the waste is difficult to handle and is likely to require further shielding before it may be moved for final disposal.

The medium activity waste also includes 40 galvanized drums holding waste with a somewhat lower activity level.

Two waste items have been identified to be too large to be packed in containers. The two items are the top shield plug (TSP) and the top shield ring (TSR) from DR3. They must both be segmented and packed in ISO containers or steel containers, or they must be packed in a special packing, designed individually. The dimensions of the items are as follows:

- TSP unit: ø2500 mm, height: 1800 mm, weight: 55 tons.
- TSR unit: ø2940 mm, height: 1700 mm, weight: 40-65 tons.

It is estimated that eight ISO containers are needed for each of the two items.

Activity in 2008

The total activity is:

Short lived β/γ-nuclides:	5,400 GBq
Long lived β/γ-nuclides:	18,000 GBq
α-nuclides:	-

It is estimated in Danish Decommissioning (2010b) that the activity of the arms in year 2000 was: 2,400 GBq Ni-63; 22 GBq Ni-59; 560 GBq Co-60 and 90 GBq Cd-113. Further, 0.2 GBq C-14; 0.1 GBq Cl-36 and 0.02 GBq Nb-94. Doc. [12] also estimates that the activity in year 2000 of an old set of underplugs was 3,200 GBq Co-60; 14,000 GBq Ni-63 and 130 GBq Ni-59. A new set of underplugs to be removed from DR3 is estimated to include only 1/3 of the activity of the long lived nuclides.

The nuclide distribution given above is in good agreement with the activity indicated for the waste described as "C+T lager fra DR3" in Danish Decommissioning (2010a). Thus the overall activity of the waste in Danish Decommissioning (2010a) is distributed on the nuclides as indicated below. The nuclides not identified in Danish Decommissioning (2010b) are attributed to nuclides in the 40 drums.

In addition to this, Danish Decommissioning (2010d) indicates that the 15 GBq + 20 GBq Sm-151 indicated in Danish Decommissioning (2010a) for low activity waste is also included in the decommissioning waste from the reactors.

Short lived β/γ-nuclides
The short lived β/γ-nuclides in this waste are considered to be:

- Co-60, Sr-90, Cs-137

The relative distributions of the nuclides and their activity are considered to be:

	Co-60	Sr-90	Cs-137
Fraction	0.995	0.002	0.003
GBq	5,373	11	16

Long lived β/γ-nuclides
The long lived β/γ-nuclides in this waste are considered to be:

- Ni-63, Sm-151

The relative distributions of the nuclides and their activity are considered to be:

	Ni-63	Sm-151
Fraction	0.998	0.002
GBq	17,964	36

Activity versus time
The activity of this waste versus time is presented in Figure A.8.

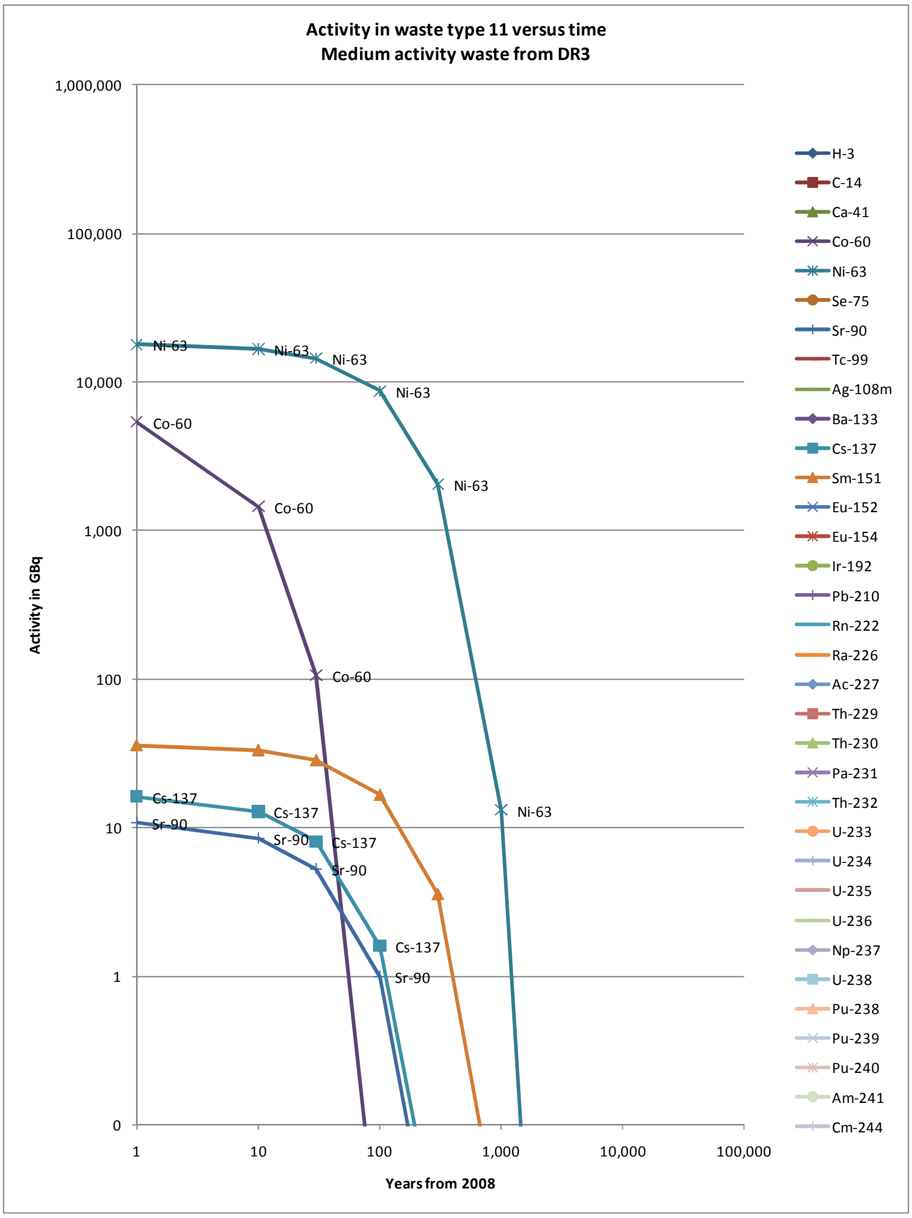


Figure A.8: Medium activity waste. Estimated activity in GBq versus time (from 2008) for waste from DR3 (type 11)

Waste from HotCell (type 12)

This waste consists of 180 drums with various waste items from the Hot Cell, 40 A-bins, plus a number of items too long to be placed in drums.

The 180 drums are 210 l concrete-lined galvanized drums containing various waste from the Hot Cell operation.

A-bins are cylinder-shaped stainless steel containers (wall thickness of 1.25 mm). The inner diameter is 22 cm, the height is 87 cm. The volume of an A-bin is approximately 30 l. "A bøtter alm" includes 40 A-tubs holding spent fuel. Moving these A-tubs will require remote operated equipment. The A-tubs also have to be repacked before disposal considering the type of repository chosen.

The various items are items too large to be put into drums. The actual, maximum measures are not yet known, but some of them must be reduced in size to fit into the disposal packages.

Due to the general uncertainty about the 'various items', the exact number of containers cannot be determined at this pre-feasibility stage. However, currently it is assumed to be 70 containers.

Activity in 2008

The activity is:

Short lived β/γ-nuclides: 33,000 GBq
Long lived β/γ-nuclides: 147 GBq
α-nuclides: 1,300 GBq

The waste is considered to be the waste described in Danish Decommissioning (2010a) as "C+T lager fra HotCell", "C-Gruber mest HotCell" and "A-bøtter alm". The overall level of activity is in good agreement with the activity indicated above.

Short lived β/γ-nuclides

The short lived β/γ-nuclides in this waste are considered to be:

- Co-60, Sr-90, Cs-137, Eu-154

The relative distributions of the nuclides and their activity are considered to be:

	Co-60	Sr-90	Cs-137	Eu-154
Fraction	0.65	0.14	0.2	0.01
GBq	21,450	4,620	6,600	330

Long lived β/γ-nuclides

The long lived β/γ-nuclides in this waste are considered to be:

- Tc-99, Sm-151

The relative distributions of the nuclides and their activity are considered to be:

	Tc-99	Sm-151
Fraction	0.02	0.98
GBq	3	144

α-nuclides

The α-nuclides in this waste are considered to be:

- Pu-238, Pu-239, Pu-240, Am-241, Cm-244 and very small amounts of U-234, U-238

The relative distributions of the nuclides and their activity are considered to be:

	Pu-238	Pu-239	Pu-240	Am-241	Cm-244
Fraction	0.31	0.04	0.06	0.47	0.12
GBq	403	52	78	611	156

Activity versus time

The activity of the waste versus time is presented in Figure A.9.

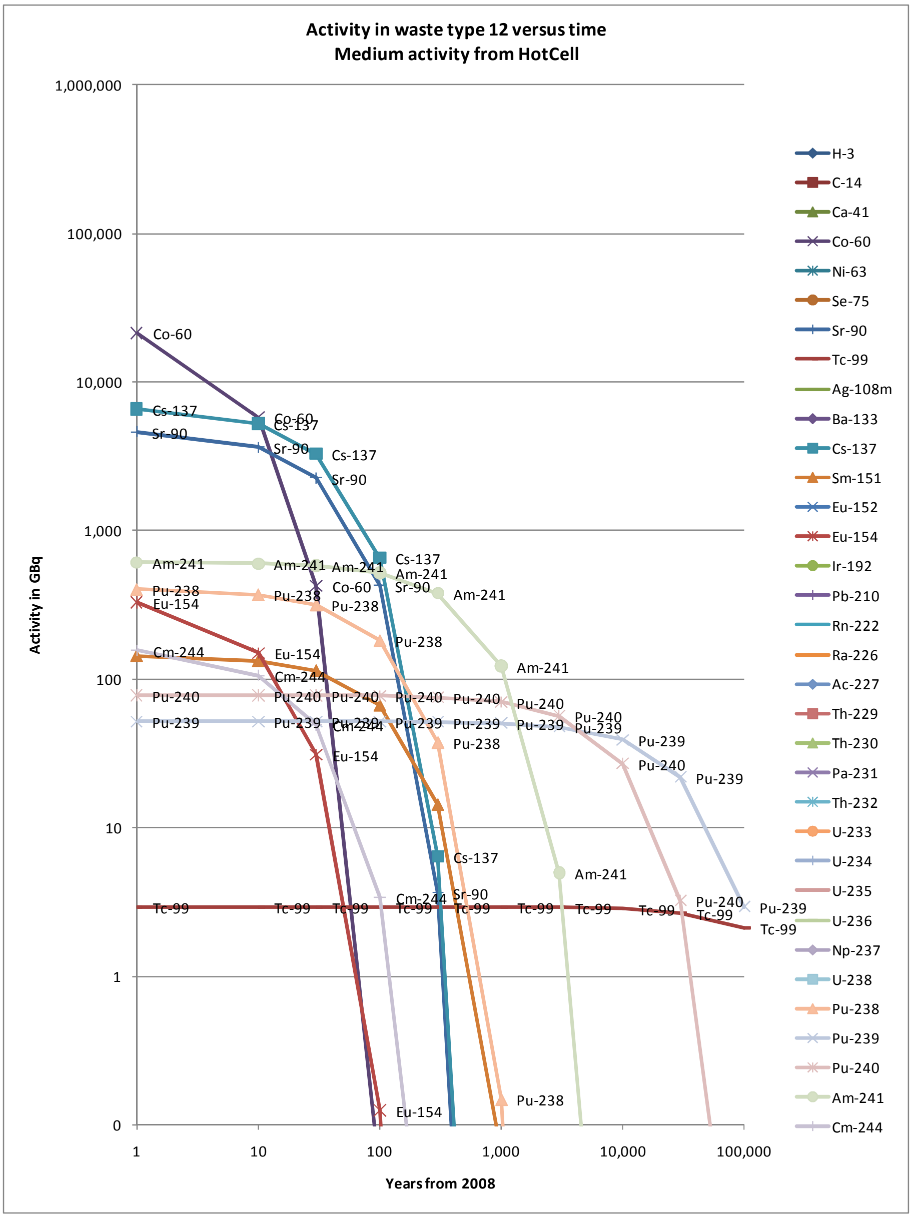


Figure A.9: Medium activity waste. Estimated activity in GBq versus time (from 2008) for waste generated at Hot Cell (type 12)

Medium active waste, external origin (type 13)

This waste is for a large part of external origin, including both sources and contaminated waste of a more general nature. The sources are mainly delivered from external suppliers (industry and hospitals). Some of them have been put into 18 units of 210 l concrete-lined galvanized drums.

In addition to the 18 drums, waste type 13 also includes some other unspecified waste. Among others, this additional waste includes two large sources: one from Skejby Hospital, containing cobalt, and one from Rigshospitalet – the Copenhagen University Hospital – containing caesium.

The two sources contain Co-60 with an activity of 33,000 GBq and Cs-137 with an activity of 30,000 GBq from medical use. It is not foreseen that Danish Decommissioning will receive such powerful sources in the future. Also a number of sources originating from SIS and Risø's activities are included in this waste.

The waste is at present packed in galvanized concrete lined steel 210 l drums. If it is decided, that specific sources are to be deposited in a bore hole, the drums presently holding these items have to be repacked.

Activity in 2008
The activity inventory for this group of medium activity waste is per June 1, 2008:

Short lived β/γ-nuclides:	370,000 GBq
Long lived β/γ-nuclides:	300 GBq
α-nuclides:	1,500 GBq

This type of waste is an important contributor to the overall activity of the waste with respect to the short lived nuclides. The indicated activity corresponds to 30% of the total short lived activity of the waste.

Detailed information about the nuclide contents of the majority of the waste from external sources is available in Danish Decommissioning (2009c). It is known that some of this waste has been packed with the low activity waste (type 10). However for the purpose of this study it was assumed that this external waste is only included in the type 13 waste. Considering the comparably much lower total activity of the type 10 waste this is a reasonable simplification.

In accordance with Danish Decommissioning (2009c)the external waste includes a long list of nuclides. These are indicated below together with the activity of the waste per year 2008. However, considering the great number of different nuclides, only nuclides with an activity level above 1 GBq have been included.

Short lived β/γ-nuclides:

	H-3	P-32	P-33	S-35	Ar-41	Fe-55	Co-57	Co-60
GBq	3,789	4	1	4	1	2	22	38,371
Fraction	0.048	0	0	0	0	0	0	0.486

	Se-75	Kr-85	Sr-90	Cd-109	Sn-113	I-125	I-131	Ba-133
GBq	1,730	197	42	3	2	18	1	1
Fraction	0.022	0	0.001	0	0	0	0	0

	Cs-137	Pm-147	Eu-152	Gd-153	Ir-192	Pb-210
GBq	31,771	13	10	28	3,197	7
Fraction	0.403	0	0	0	0.041	0

Long lived β/γ-nuclides:

	C-14	Ni-63	Kr-81
GBq	420	39	29
Fraction	0.915	0.085	0

α-nuclides:

	Ra-226	U-238	Pu-238	Am-241	Am-243
GBq	246	4	3	2,955	2
Fraction	0.077	0.001	0.001	0.921	0.001

The total activity of this external waste estimated for 2008 based on the information in Danish Decommissioning (2009c) is:

Short lived β/γ-nuclides: 79,203 GBq
Long lived β/γ-nuclides: 488 GBq
α-nuclides: 3,210 GBq

<u>Nuclides in waste of external origin in years to come</u>
To assess the nuclide inventory of the external waste to be sent to DD for final disposal until the year 2040, the information in Danish Decommissioning (2009c) was extrapolated adding every year the average additional amount of each nuclide for the period 2000 - 2009, however not including the two powerful sources and not considering any new larger sources. Decay of the nuclides was considered as some nuclides have a very short half life.

<u>The rest of the medium activity waste</u>
Subtracting the estimate for the waste of external origin for 2008 from the activity indicated in Table 2.2 in the main report indicates that a large amount of short lived β-nuclides is not accounted for.

Short lived β/γ-nuclides: 290,000 GBq
Long lived β/γ-nuclides: Small amounts
α-nuclides: Small amounts

It is assumed that the origin of this is the sources described in Danish Decommissioning (2010a) as "Kilder p.t. i brug i DK ifølge SIS og Risø oplysninger". Thus the sources are partly of external origin and partly sources used at Risø.

The activity of this waste is however, in the order of 800,000 GBq (considering decay until 2008) due to the short lived β-nuclides and 12,000 GBq due to α-nuclides.

For the purpose of this study it is assumed that all sources indicated except a number of the Co-60, Cs-137 and the Am-241 sources are included in the type 13 waste.

The nuclides indicated below are considered to account for the remaining activity in the remaining part of the medium activity waste:

Short lived β/γ-nuclides: H-3, Co-60, Sr-90, Cs-137

The relative distributions and activities of these nuclides in 2008 were estimated to:

	H-3	Co-60	Sr-90	Cs-137
GBq	800	144,200	800	144,200
Fraction	0.003	0.497	0.003	0.497

α-nuclides: Ra-226, U-234, U-238

The relative distributions and activities of these nuclides in 2008 were estimated to:

α-nuclides:

	Ra-226	U-234	U-238
GBq	39	12	24
Fraction	0.52	0.16	0.32

<u>Activity versus time</u>
The activity of the waste versus time is presented in Figures A.10a and A.10b.

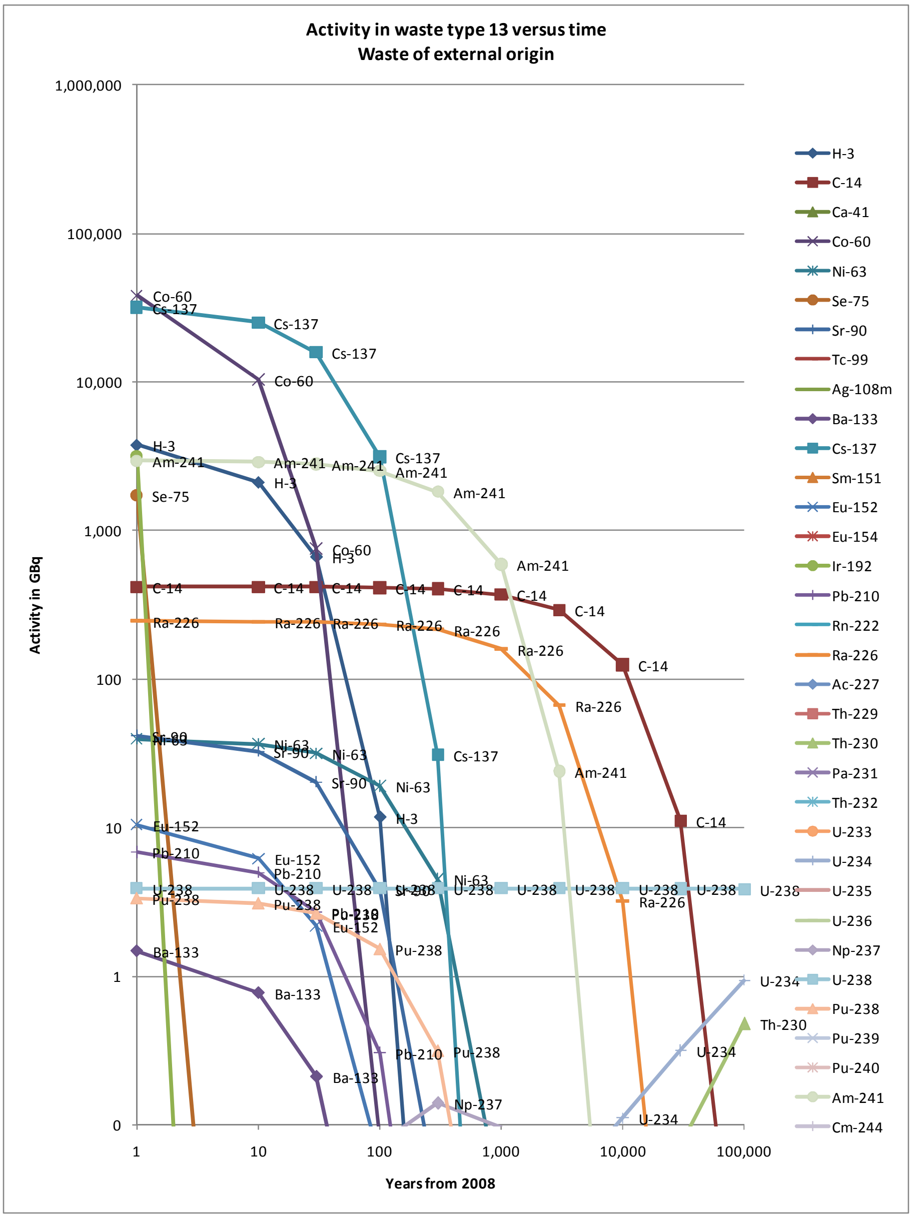


Figure A.10a: Medium activity waste of external origin. Estimated activity in GBq versus time (from 2008) (type 13 part a)

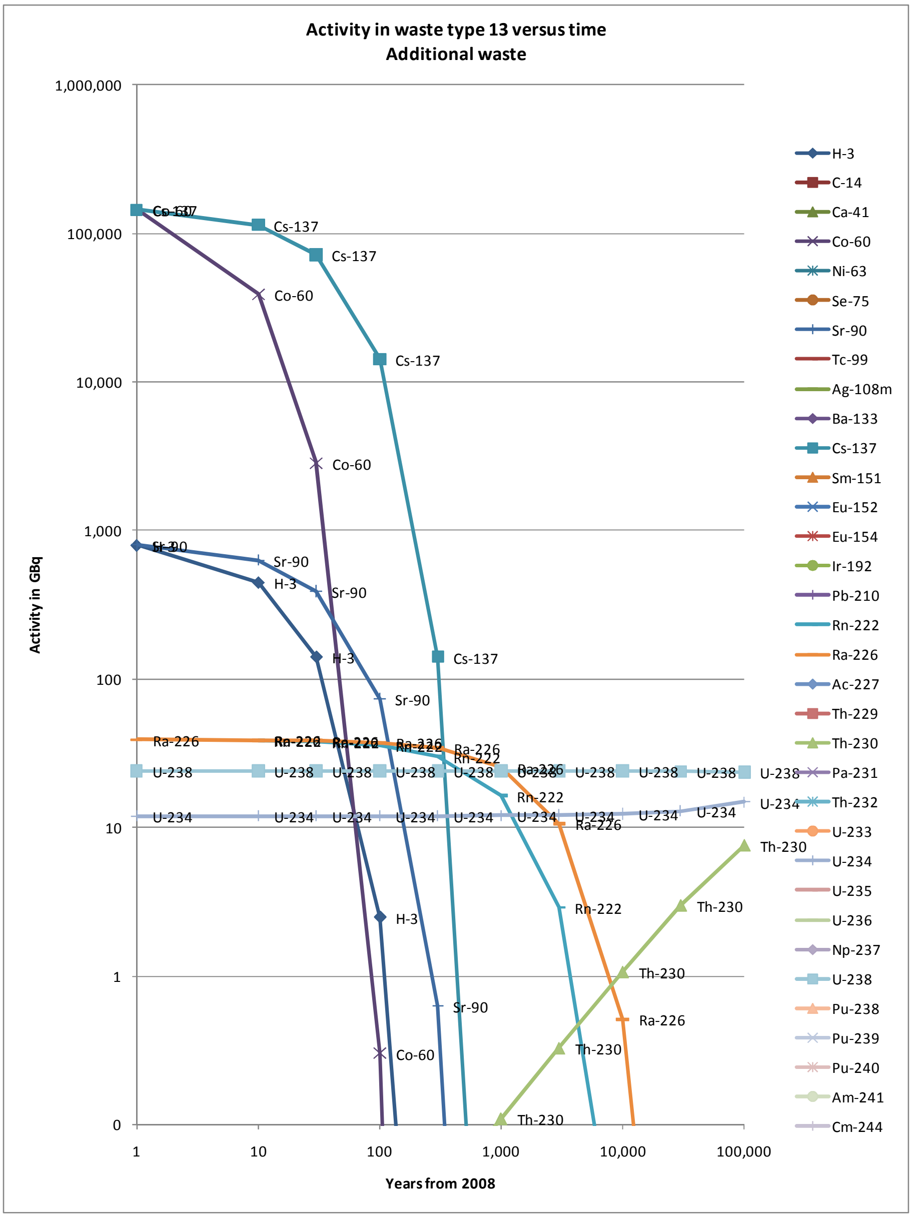


Figure A.10b: Additional medium activity waste. Estimated activity in GBq versus time (from 2008) (type 13 part b)

A.1.5 Special waste

The special waste may be packed in steel containers and deposited among the rest of the waste, or the special waste - or part of this - may be repacked in slim containers and stored in a deep shaft.

The volume of the packed waste indicated in Table 2.1 and Table 2.2 in the main report refers to the volume of the waste packed in steel containers. When packed for storage in a shaft, the volume of the conditioned waste will be considerably smaller.

Due to high activity levels, repacking will require a Hot Cell facility for remote handling of the waste.

For this waste the criticality of the waste has to be considered, i.e. the active waste has to be separated by distance to reduce the likelihood of a self sustaining nuclear chain reaction.

About 20 powerful sources (type 14)

Although it has been tried to dispose of all major sources, some must still be sent to the repository. It is assumed, that the waste will include about 20 larger sources.

The sources, though powerful, are physically small. They generally consist of a small lead chamber covered in a layer of paraffin. The chamber has a window which is vulnerable allowing water to pass in and out if damaged. All nuclides are solid compounds.

Activity in 2008
The activity is, see table 3.3:

Short lived β/γ-nuclides: -
Long lived β/γ-nuclides: -
α-nuclides: 1,000 GBq

α-nuclides

Danish Decommissioning (2008b) informs that the majority of the sources are long lived Ra-226 and Am-241 α-sources and Ra-Be and Am-Be α/neutron sources. The reference indicates the presence of at least 16 Ra-Be sources and 28 Am-Be sources.

Danish Decommissioning (2010a) indicates under the heading "Store kilder" activity due to both β/γ-nuclides and α-nuclides.

For the purpose of the present analysis it is assumed that these β/γ-nuclide sources are either reused for other purposes or included in waste type 13. Among the α-nuclides Danish Decommissioning (2010a) only indicates a 185 GBq source of Pu-239. In Danish Decommissioning (2010a) under the heading: "Dekom. DR1" a 4 GBq Ra-226 source is listed. These are assumed to be included in the type 14 waste. The remaining α-activity is considered to be due to an equal amount of Ra and Am sources.

The α-nuclides in this waste are considered to be: Ra-226, Pu-239, Am-241.

The relative distributions and activity of the nuclides are considered to be:

	Ra-226	Pu-239	Am-241
Fraction	0.41	0.185	0.406
GBq	410	185	406

Activity versus time
The activity of the waste versus time is presented in figure A.11.

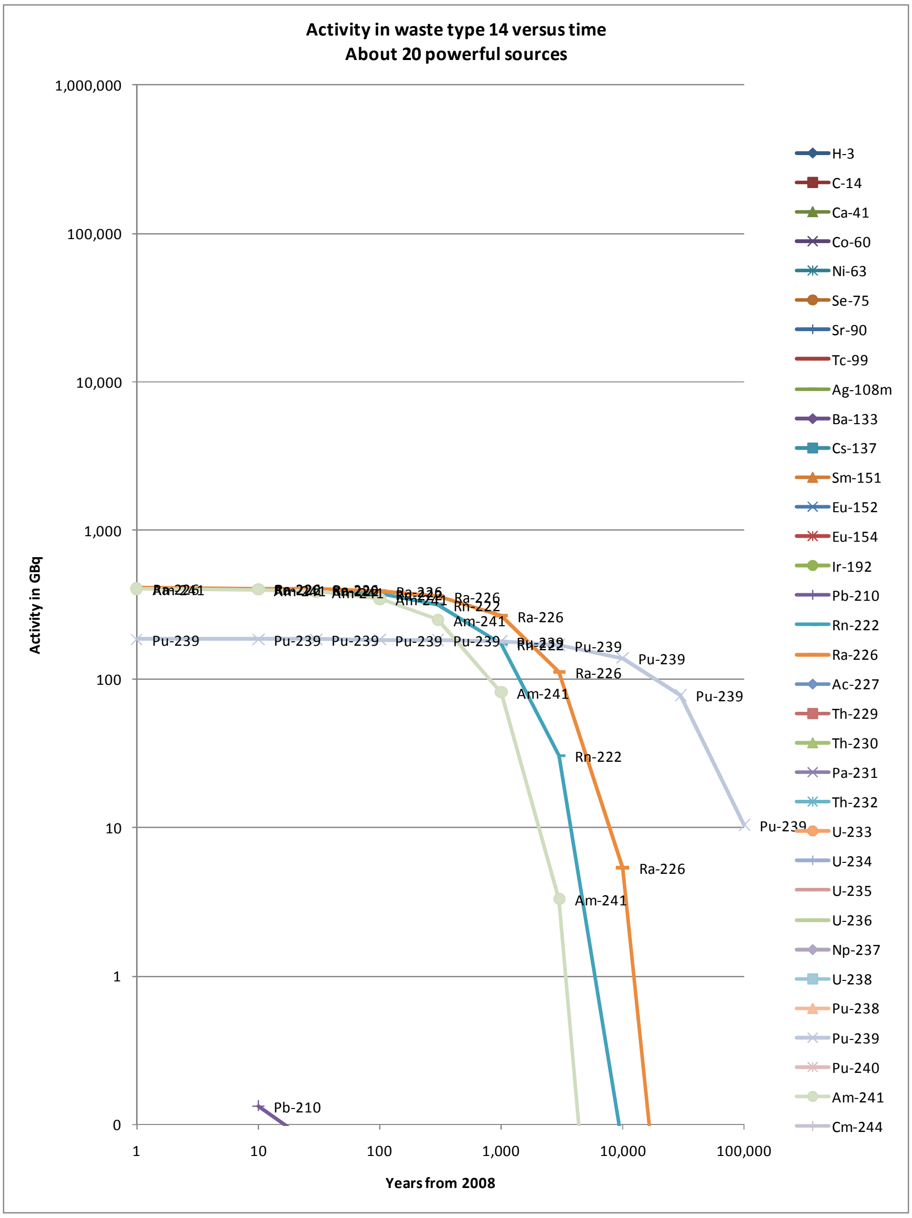


Figure A.11: Estimated activity in GBq versus time (from 2008) for 20 larger sources (type 14)

1.2 kg irradiated uranium in solution (type 15)

The waste consists of 1.2 kg of irradiated uranium. The waste has to be conditioned before being sent to the repository. There are two options. 1) Repacking with the aim of storage in a deep shaft. This may be costly, as the waste must be handled by remote operated equipment. 2) Repacking into steel containers.

Activity in 2008
The activity is, see table 3.3:

Short lived β/γ-nuclides: 4,000 GBq
Long lived β/γ-nuclides: 9 GBq

α-nuclides: 400 GBq

The nuclide distribution is considered to be as indicated in Danish Decommissioning (2010a) for the waste described as "8 kg in 7 A-bøtter". The relative amount of activity considering the types of nuclide is in good agreement.

<u>Short lived β/γ-nuclides</u>
The short lived β/γ-nuclides in this waste are considered to be:

- Sr-90, Cs-137, Eu-154

The relative distribution and the activities of the nuclides in 2008 are estimated to:

	Sr-90	Cs-137	Eu-154
Fraction	0.40	0.57	0.03
GBq	1,600	2,280	120

<u>Long lived β/γ-nuclides</u>
The long lived β/γ-nuclides in this waste are considered to be:

- Tc-99, Sm151

	Tc-99	Sm-151
Fraction	0.02	0.98
GBq	0.1	5.9

<u>α-nuclides</u>
The α-nuclides in this waste are considered to be:

- Pu-238, Pu-239, Pu-240, Am-241, Cm-244

The relative distribution and the activities of the nuclides in 2008 are estimated to:

	Pu-238	Pu-239	Pu-240	Am-241	Cm-244
Fraction	0.30	0.04	0.06	0.46	0.14
GBq	120	16	24	184	56

<u>Activity versus time</u>
The activity of this waste versus time is presented in Figure A.12.

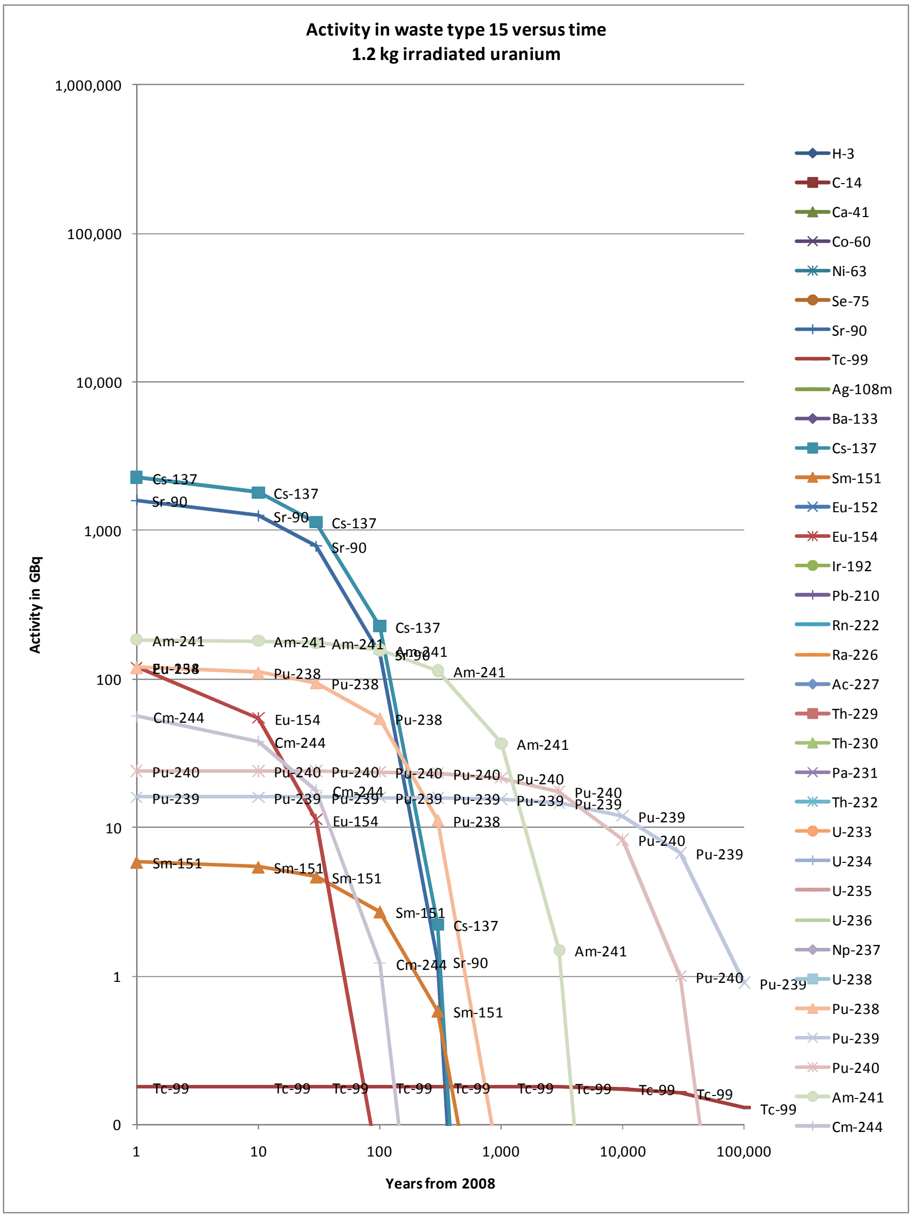


Figure A.12: Estimated activity in GBq versus time (from 2008) for 1.2 kg irradiated uranium (type 15)

12 kg irradiated fuel (type 16)

A volume of 20 m^3 as indicated in Table 2.2 in the main report is the volume of the waste packed in steel containers. In case the waste is to be stored in a deep shaft, it shall be packed in "canisters" and the volume would be much smaller.

Activity in 2008

The activity is:

Short lived β/γ-nuclides: 23,000 GBq
Long lived β/γ-nuclides: 55 GBq
α-nuclides: 1,500 GBq

The same distribution of nuclides as assumed for waste of type 15 has been used.

Short lived β/γ-nuclides
The short lived β/γ-nuclides in this waste are considered to be:

- Sr-90, Cs-137, Eu-154

The relative distribution of the nuclides in 2008 is estimated to:

	Sr-90	Cs-137	Eu-154
Fraction	0.40	0.57	0.03
GBq	9,200	13,110	690

Long lived β/γ-nuclides
The long lived β/γ-nuclides in this waste are considered to be:

- Tc-99, Sm-151

	Tc-99	Sm-151
Fraction	0.02	0.98
GBq	1	57

α-nuclides
The α-nuclides in this waste are considered to be:

- Pu-238, Pu-239, Pu-240, Am-241, Cm-244

The relative distribution and activity of the nuclides in 2008 is estimated to:

	Pu-238	Pu-239	Pu-240	Am-241	Cm-244
Fraction	0.30	0.04	0.06	0.46	0.14
GBq	450	60	90	690	210

Activity versus time
The activity of this waste versus time is presented in Figure A.13.

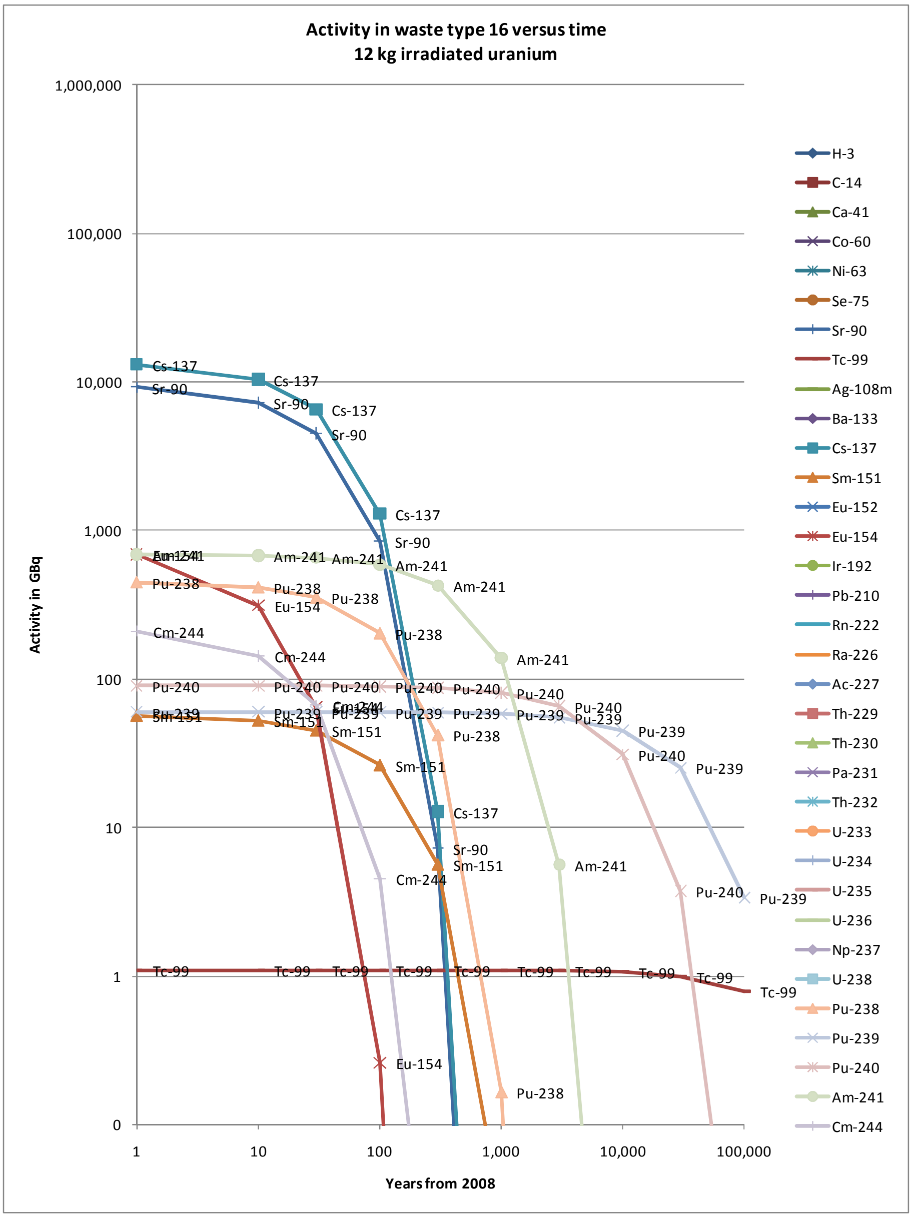


Figure A.13: Estimated activity in GBq versus time (from 2008) for 12 kg irradiated fuel (type 16)

222 kg irradiated fuel (type 17)

<u>Activity in 2008</u>

The activity is:

Short lived β/γ-nuclides: 730,000 GBq
Long lived β/γ-nuclides: 5200 GBq
α-nuclides: 31,000 GBq

This waste includes the largest fraction of short lived activity, nearly 60 % of all waste, and the largest fraction of α-activity, about 83 %. For this waste, criticality is a specific difficulty when considering packing for disposal.

The activity levels of this waste type are in good agreement with the waste described in Danish Decommissioning (2010a) as "225 kg i 18 A-bøtter". Thus, the same nuclides and nuclide distribution as indicated for this waste is assumed here.

Short lived β/γ-nuclides
The short lived β/γ-nuclides in this waste are considered to be:

- Sr-90, Cs-137, Eu-154

The relative distribution and activity of the nuclides in 2008 is estimated to:

	Sr-90	Cs-137	Eu-154
Fraction	0.39	0.57	0.04
GBq	284,700	416,100	29,200

Long lived β/γ-nuclides
The long lived β/γ-nuclides in this waste are considered to be:

- Tc-99, Sm-151

	Tc-99	Sm-151
Fraction	0.02	0.98
GBq	100	5100

α-nuclides
α-nuclides in this waste are considered to be:

- Pu-238, Pu-239, Pu-240, Am-241, Cm-244

The relative distribution and activity of the nuclides in 2008 is estimated to:

	Pu-238	Pu-239	Pu-240	Am-241	Cm-244
Fraction	0.35	0.04	0.06	0.45	0.10
GBq	10,850	1,240	1,860	13,950	3,100

Activity versus time
The activity of the waste versus time is presented in Figure A.14.

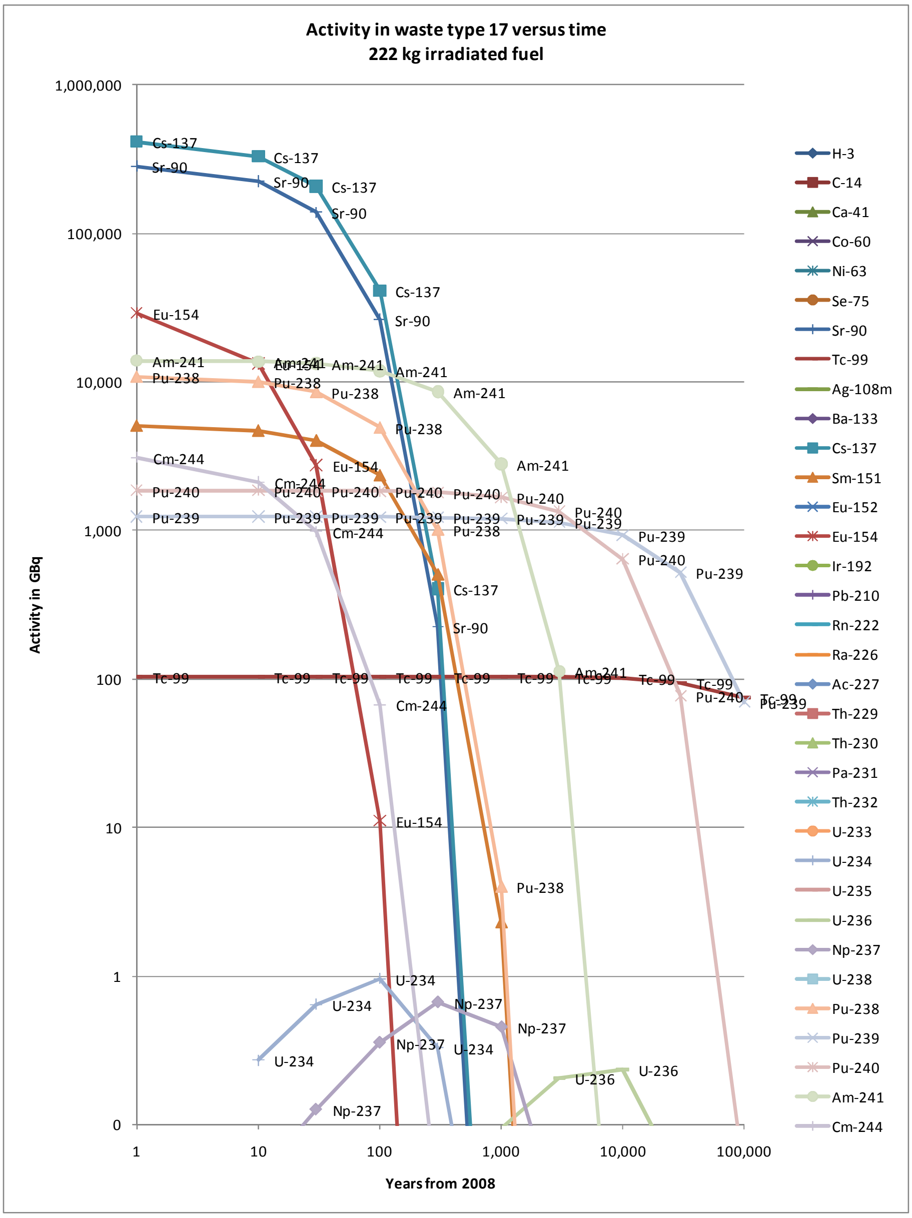


Figure A.14: Estimated activity in GBq versus time (from 2008) for 222 kg irradiated fuel (type 17)

DR1 core solution (type 18)

The acidic U-235 core solution must be solidified in a shielded facility before final disposal. A way to do this would be to make the core solution alkaline by mixing it with concrete. An option is to pack the solidified core in 4 "A-bøtter", which are again packed in a steel container.

<u>Activity in 2008</u>
The activity is, see table 3.3:

Short lived β/γ-nuclides: 120 GBq
Long lived β/γ-nuclides: 1 GBq
α-nuclides: 4 GBq

The nuclides in this waste are assumed to be as indicated in Danish Decommissioning (2010a) for the waste type described as "DR1 core".

Short lived β/γ-nuclides
The short lived β/γ-nuclides in this waste are considered to be:

- Sr-90, Cs-137, Eu-154

The relative distribution and activity of the nuclides in 2008 is estimated to:

	Sr-90	Cs-137	Eu-154
Fraction	0.40	0.56	0.04
GBq	48	67.2	4.8

Long lived β/γ-nuclides
The long lived β/γ-nuclides in this waste are considered to be:

- Sm-151

The relative distribution and activity of the nuclides in 2008 is estimated to:

	Sm-151
Fraction	1
GBq	1

α-nuclides
The α-nuclides in this waste are considered to be:

- U-234, U-235, U-236, Pu-238, Pu-239, Pu-240, Am-241

The relative distribution and activity of the nuclides in 2008 is estimated to:

	U-234	U-235	U-238	Pu-238	Pu-239	Pu-240	Am-241
Fraction	0.049	0.034	0.012	0.416	0.025	0.024	0.44
GBq	0.2	0.1	0.1	1.7	0.1	0.1	1.8

Activity versus time
The activity of this waste versus time is presented in Figure A.15.

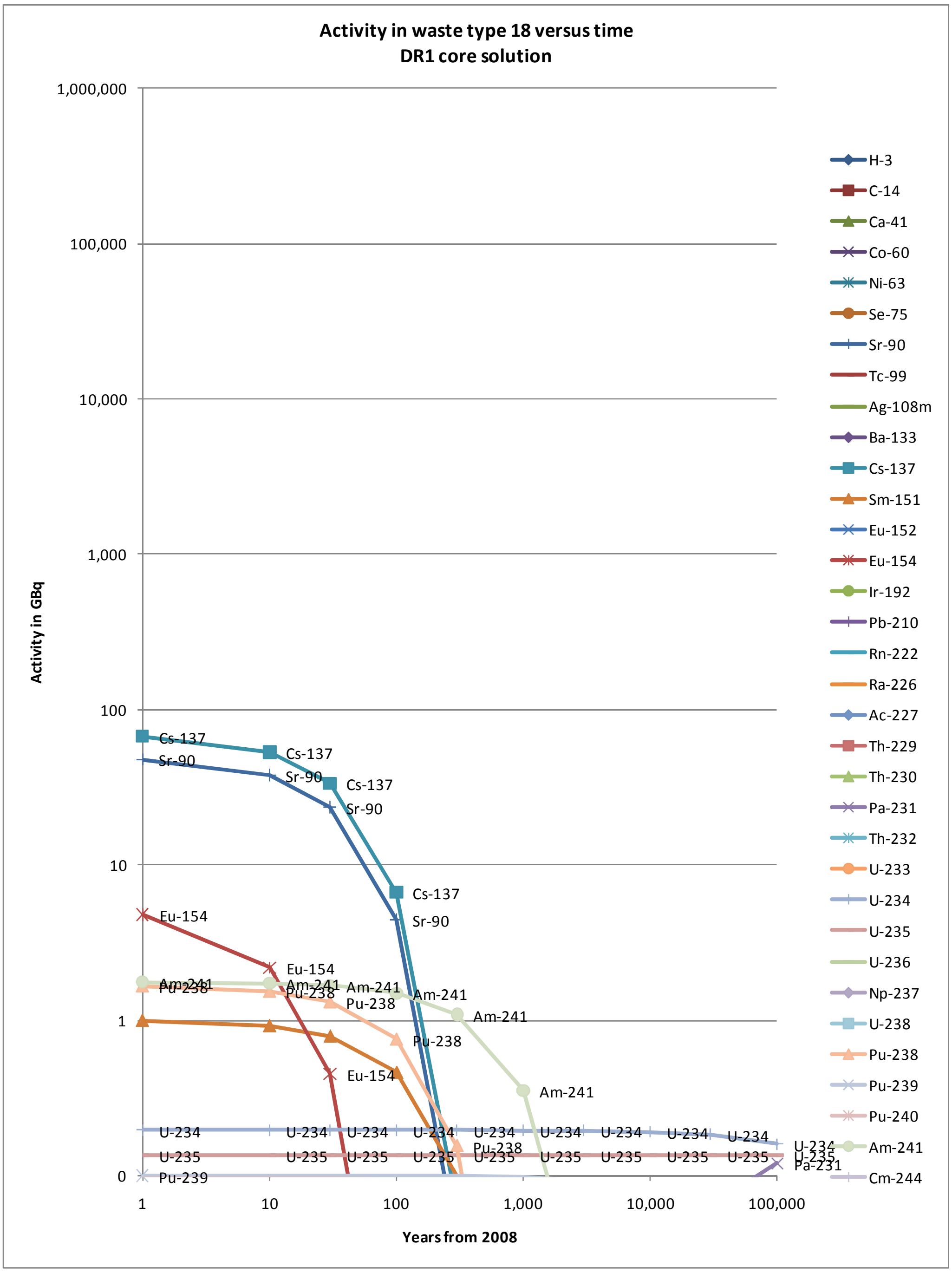


Figure A.18: Estimated activity in GBq versus time (from 2008) for the core solution from DR1 (type 18)

Non-irradiated uranium (type 19)

The nuclide distribution is as indicated in Danish Decommissioning (2010a) for the waste group "U+Th".

Danish Decommissioning is still investigating, if the waste may be reused for other purposed and not be sent to the repository. However, for the present it is considered that this waste must be sent to the repository.

Activity in 2008
The activity is:

Short lived β/γ-nuclides: -
Long lived β/γ-nuclides: -
α-nuclides: 50 GBq

α-nuclides
The α-nuclides in this waste are considered to be:

- U-234, U-235, U-238 and traces of Th-232

The distribution of the nuclides in 2008 is estimated to:

	U-234	U-235	U-238
Fraction	0.41	0.09	0.50
GBq	20.5	4.5	25

Activity versus time
The activity of this waste versus time is presented in figure A.16.

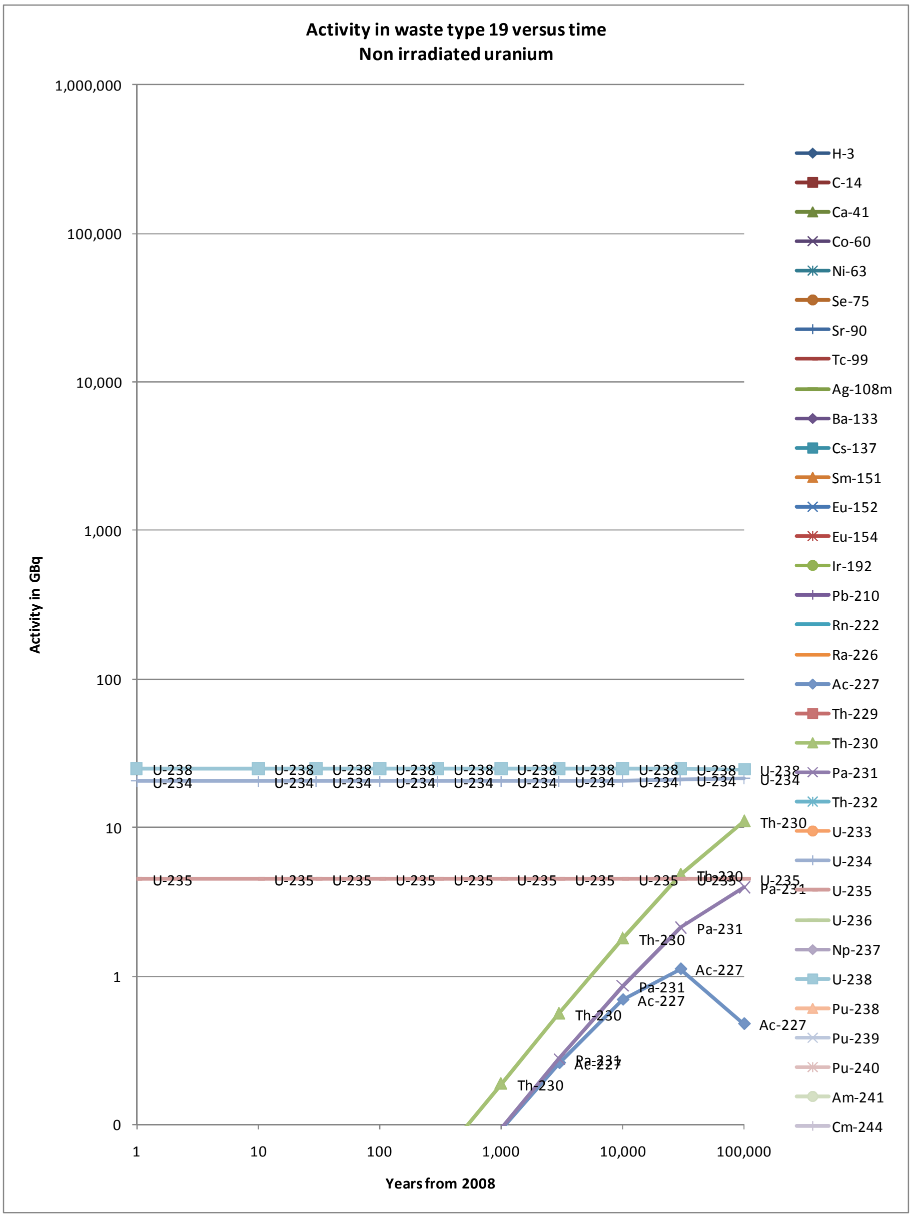


Figure A.16: Estimated activity in GBq versus time (from 2008) for non-irradiated uranium (type 19)

Heavy water (type 20)

Danish Decommissioning has informed that this waste is no longer at Risø and thus need not to be considered further in the pre-feasibility study.

A.1.6 Tailings and concrete

Tailings (type 21)

The amount of tailings is according to Ministry of Interior and health (2008) 1,130 t. The approximate content of contaminants is as follows (Danish Decommissioning (2010c):

- F: 3500 ppm
- Zn: 200 ppm
- Pb: 300 ppm

- Th: 1260 ppm
- U: 350 ppm.

Eighty ISO containers are needed for the final disposal of the tailings. Seventy ISO containers are needed for the final disposal of the concrete waste.

Activity in 2008

In contrast to the activities indicated for the other waste types the activity of both parent and daughters are indicated:

Radium series:	4.8 GBq of U-238 and the same amount for each daughter down to U-234
Radium series:	5.9 GBq of Th-230 and the same amount for each daughter
Thorium series:	5.8 GBq of Th-232 and the same amount for each daughter
Actinium series:	0.22 GBq of U-235 and Th-231
Actinium series:	0.27 GBq of Pa-231 and the same amount for each daughter.

This information is included in the subsequent modelling as indicated below.

α-nuclides

The α-nuclides in this waste are considered to be:

- Ra-226, Th-230, Th-232, U-234, U-235, and U-238

The distribution of the nuclides in 2008 is considered to be:

	Ra-226	Th-230	Th-232	U-234	U-235	U-238
Fraction	0.22	0.22	0.21	0.17	0.01	0.17
GBq	5.9	5.9	5.8	4.8	0.2	4.8

A.2 Overall distribution of nuclides on waste types

The overall distribution of nuclides on the different waste types is shown in

Table A.13.1,

Table A.13.2 and Table A.13.3.

Table A.13.1 Total estimated activity in GBq versus time (from 2008) for the short lived β/γ-nuclides considered in the pre-feasibility study. The activity levels are distributed on waste type

Nuclide		**H-3**	**Co-60**	**Se-75**	**Sr-90**	**Ba-133**	**Cs-137**	**Eu-152**	**Eu-154**	**Ir-192**	**Rn-222**
Waste type	1	2,800	12	0	0	0	0	1,044	144	0	0
Waste type	2	0	19,788	0	102	0	102	408	0	0	0
Waste type	3	35,964	26,640	0	666	1,998	666	666	0	0	0
Waste type	4	86	11	0	0	376	0	91	6	0	0
Waste type	5	0	0	0	0	0	0	0	0	0	0
Waste type	6	0	0	0	0	0	0	0	0	0	0
Waste type	7	0	0	0	0	0	0	0	0	0	0
Waste type	8	0	60	0	1,200	0	1,710	30	0	0	0
Waste type	9	0	16	0	702	0	1,012	0	70	0	0
Waste type	10	0	624	0	806	0	962	0	78	0	0
Waste type	11	0	5,373	0	11	0	16	0	0	0	0
Waste type	12	0	21,450	0	4,620	0	6,600	0	330	0	0
Waste type	13	4,589	182,571	1,730	842	1	175,971	10	0	3,197	0
Waste type	14	0	0	0	0	0	0	0	0	0	0
Waste type	15	0	0	0	1,600	0	2,280	0	120	0	0
Waste type	16	0	0	0	9,200	0	13,110	0	690	0	0
Waste type	17	0	0	0	284,700	0	416,100	0	29,200	0	0
Waste type	18	0	0	0	48	0	67	0	5	0	0
Waste type	19	0	0	0	0	0	0	0	0	0	0
Waste type	21	0	0	0	0	0	0	0	0	0	6
Sum GBq		43,439	256,546	1,730	304,496	2,376	618,596	2,250	30,643	3,197	6

Table A.13.2 Total estimated activity in GBq versus time (from 2008) for the long lived β/γ-nuclides considered in the pre-feasibility study. The activity levels are distributed on waste type.

Nuclide		**C-14**	**Ca-41**	**Ni-63**	**Tc-99**	**Ag-108m**	**Sm-151**
Waste type	1	120	0	0	0	0	0
Waste type	2	0	0	0	0	0	0
Waste type	3	0	0	17,946	0	18	0
Waste type	4	0	19	37,981	0	0	0
Waste type	5	0	0	0	0	0	0
Waste type	6	0	0	0	0	0	0
Waste type	7	0	0	0	0	0	0
Waste type	8	0	0	0	1	0	0
Waste type	9	0	0	0	1	0	0
Waste type	10	0	0	0	1	0	0
Waste type	11	0	0	18,000	0	0	36
Waste type	12	0	0	0	3	0	144
Waste type	13	420	0	39	0	0	0
Waste type	14	0	0	0	0	0	0
Waste type	15	0	0	0	0	0	6
Waste type	16	0	0	0	1	0	57
Waste type	17	0	0	0	104	0	5,096
Waste type	18	0	0	0	0	0	1
Waste type	19	0	0	0	0	0	0
Waste type	21	0	0	0	0	0	0
Sum GBq		540	19	73,966	110	18	5,340

Table A.13.3 Total estimated activity in GBq versus time (from 2008) for the α-nuclides considered in the pre-feasibility study. The activity levels are distributed on waste type.

Waste type	Ra-226	Th-230	Th-232	U-234	U-235	U-238	Pu-238	Pu-239	Pu-240	Am-241	Cm-244
1	0	0	0	0	0	0	0	0	0	0	0
2	0	0	0	0	0	0	0	0	0	0	0
3	0	0	0	0	0	0	0	1	0	0	0
4	0	0	0	0	0	0	35	37	2	35	0
5	0	0	0	0	0	0	0	0	0	0	0
6	0	0	0	0	0	0	0	0	0	0	0
7	0	0	0	0	0	0	0	0	0	0	0
8	0	0	0	0	0	0	59	10	16	64	11
9	0	0	0	0	0	0	45	5	8	54	17
10	0	0	0	0	0	0	60	7	10	71	22
11	0	0	0	0	0	0	0	0	0	0	0
12	0	0	0	0	0	0	403	52	78	611	156
13	285	0	0	12	0	28	3	0	0	2,955	0
14	410	0	0	0	0	0	0	185	0	406	0
15	0	0	0	0	0	0	120	16	24	184	56
16	0	0	0	0	0	0	450	60	90	690	210
17	0	0	0	0	0	0	10,850	1,240	1,860	13,950	3,100
18	0	0	0	0	0	0	2	0	0	2	0
19	0	0	0	21	5	25	0	0	0	0	0
21	6	6	6	5	0	5	0	0	0	0	0
Total	701	6	6	38	5	58	12,027	1,612	2,088	19,023	3,572

Appendix B: Possible packaging of waste, details

B.1 Details on fill and backfill material

Within the pre-feasibility study, the following types of fill and backfill materials have been investigated:

- Cement-calcium granulate;
- Concrete;
- Bentonite; and
- Sand and gravel.

Details on the properties of these materials are provided below.

Cement-calcium granulate

Danish Decommissioning has carried out test series with a specially manufactured cement-calcium granulate. The tested cement-calcium granulates is composed of 2/3 calcium granulate and 1/3 standard Portland cement which was mixed in a standard concrete mixer. The mixed cement-calcium granulates has a density of 1.3 - 1.6 ton/m^3 at the natural water content after manufacture. The final density after absorption of water as well as the porosity and hydraulic conductivity has not yet been tested.

The tested granulate showed the following grain distribution: 10 % < 0.5 mm, 36 % < 1.0 mm, 58 % < 1.6 mm; 75 % < 3.2 mm. The granulate was originally designed to be used as backfill in between drums in a steel container. Tests have shown that the dry-mixed granulate can easily be filled in between the drums due to a low friction angle. Furthermore, it is possible to use the granulate inside drums, in connection with packing or re-packing of drums. It is considered suitable for this purpose due to a combination of a low friction angle, a high pH, and a porosity, which allows for expansion of corroded waste items.

The final stability of the cement-calcium granulates after absorption of water has not been tested, but it is assumed that it will resemble the texture of semi-cured concrete. The mixed granulate will retain a high pH level and should therefore be relatively protective toward corrosion of the steel waste containers.

Concrete

Concrete is a well-known and tested material for packaging between different types of hazardous waste materials including nuclear waste.

Normal concrete can be manufactured at densities ranging from 2.0 to 2.5 tons/m^3 by applying different aggregates to the concrete mixture. In any case, the weight of the concrete will significantly reduce the waste storage capacity of the containers. Properly specified and prepared concrete is very impermeable, while cracked concrete can become very permeable. The effective porosity of intact concrete is very low, while the total porosity will depend on the specification and preparation of the concrete (Neville (1995), Mindess & Darwin (2003)).

Concrete can be poured in between waste components in a container. Concrete will also be well-suited as backfill in between containers, and it may be used below the groundwater level. Furthermore, concrete will be very stable after installation, and concrete will have a high pH level and will not be corrosive to the steel containers. If concrete is used as backfill in between containers, the repository will not be reversible. If concrete is poured inside the containers, the disposal will be reversible in the sense that it will be possible to remove containers from the repository in future.

Concrete will constitute a quite low permeable barrier against migration of liquids as long as the concrete remains intact. Cracked concrete will allow migration of liquids and gases, but will maintain an ability to shield for radiation. In addition, concrete will have a certain ability to retain nuclides.

Bentonite

Bentonite can be used as pure fill or backfill material or mixed in various ratios with different types of material. The Swedish Nuclear Fuel and Waste Management Co have carried out a number of laboratory tests with different mixtures of bentonite and crushed rock (SKB, 2006). These tests have described the relation between the swelling pressure and dry density at different NaCl concentrations, and similarly the relation between hydraulic conductivity and dry density. Given the tested variation in density, the swelling pressure may vary from 4.5 to 13 MPa.

There are different types of bentonite products on the market. Bentonite can be purchased in bulk as loose powder, as pellets or as a liner or a carpet, where it is fixed in between two layers of textile. Powdered bentonite can be compacted to relatively high dry densities (up to approximately 2.0 ton/m^3), while bentonite pellets will normally be installed at dry densities of some 1.1 ton/m^3. After installation of the bentonite, water must be added for the bentonite to begin the swelling process. After full expansion, the bentonite minerals will occupy previous voids both between the bentonite and the surroundings. The residual swelling pressure after initial expansion will be high for compacted and dense bentonite, while the residual swelling pressure will be relatively low for bentonite pellets, because a lot of the swelling potential is used to fill gaps between the pellets themselves.

Bentonite pellets will easily fill in gaps between the waste containers and inside these if used as fill.

The effective porosity of bentonite will be very low after the swelling has completed. The hydraulic conductivity will to some extent depend on the final density of bentonite, but will in any case be at least $K < 10^{-10}$ m/s.

The expanded bentonite will retain its physical stability and strength against subsidence, even if the moisture is removed from the bentonite. However, in this case, shrinkage cracks in the bentonite will be formed, and unless the reintroduction of water allows the bentonite to swell again, the bentonite will loose its tightness.

Bentonite has good retention properties for contaminants, since there are many adsorption sites related to the clay particles.

Sand and gravel

Specially selected and prepared sand and gravel can in principle be used as fill inside and backfill in between waste containers. The selected material should be sieved to provide a uniform grain size, as this will secure a low friction angle, and the material will then easily fill in all cavities between waste materials and/or waste containers.

For uniform sand, a dry density of 1.7 - 1.8 ton/m^3 can be assumed, and porosity will be at least 30 %, the hydraulic conductivity will be K > 10^{-4} m/s even with relatively small grains.

The selected sand should be composed of pure silicate minerals and must at least satisfy the criteria for aggregates used in the concrete industry to ensure minimum aggressiveness toward corrosion of the containers.

Due to its higher permeability, sand is generally not considered feasible as a fill material inside containers.

Sand and gravel will remain loose and will not form a rigid body in the same way as for example concrete. When the waste containers corrode, the sand will gradually fill in the voids, while the total height/volume of the waste containers will subside. Sand/gravel does not react with aluminium and may therefore be used in containers with aluminium waste in order to prevent the development of gas.

B.1.1 Retention properties

The aim of the use of fill and backfill materials is to delay and reduce the release of radionuclides from the repository due to the properties of the materials. These properties can be divided into three groups:

- Properties altering the flow of water through the waste items and repository.
- Properties altering the solubility of the radionuclides
- Properties leading to retardation of the radionuclides compared to the water flow.

The first group of properties are related to the porosity of the material and thus the hydraulic conductivity. Materials with low hydraulic conductivity will reduce the amount of water both coming into the repository (and thus initiating the degradation of the concrete and steel elements making up the construction of repository and the containers holding the waste) and leaching from the repository. In this way the time of degradation of the physical barriers will be delayed and the total amount of radionuclides leaving the repository will be reduced.

The hydraulic conductivities used in the calculation of long term leaching from the repository are given in Table B.13.4. The parameters are based on a broad search of data both in the literature related to assessment of potential impact from radionuclide waste repositories and geotechnical literature in general. In Table B.13.4 only the best estimate value is given. In the modelling of the uncertainty related to the overall release of nuclides from the repository, a reasonable interval for the variation of the hydraulic conductivity is also used. As can be seen from Table B.13.4, the difference between the hydraulic conductivities of the possible fill and backfill materials runs in orders of magnitude.

Table B.13.4 Hydraulic conductivity of fill and backfill materials

Soil	Hydraulic conductivity (m/s) most likely
Calcium cement granulate	$1.00E^{-08}$
Bentonite	$1.00E^{-10}$
Cement	$1.00E^{-09}$
Sand	$1.00E^{-04}$

The solubility of the radionuclides in the repository will depend on the solid geochemical form of the waste in question, the mix of the radionuclides and the geochemical environment created by the fill materials surrounding the waste.

In order to assess the release of radionuclides from the repository properly, taking the presence of fill materials into account, geochemical calculations should be performed for a specific combination of the above. This requires that all relevant properties are known and that the relevant parameters for the geochemical calculations are available. Quite an amount of research has been carried out over the last years, e.g. in relation to EU projects aiming to combine the knowledge of a number of research institutes working with radionuclides and the questions related to disposal of waste from radioactive activities. But a lot of specific information is still not available[82].

Coelho, et al (2009) has reported for the PAMINA project that a comparison of modelling of radionuclide migration in the near field of a repository has shown that the difference for the results between modelling the release based on geochemical modelling and a combined modelling based on solubilities and retardation coefficients, K_D, are not great. Since the prefeasibility study by default is based on generalised assumptions[83], it has been chosen not to use geochemical modelling, but an approach based on solubilities and retardation factors.

[82] See for instance Coelho, et al (2009)

[83] E.g. no specific information is available on about relevant solid phases

Information on solubilities has been collected from a number of published articles and report, which have taken into account both fill materials and waste mixes. Based on this, solubilities[84] relevant for different fill material environments have been suggested, see Table B.13.5. Data does not exist for all radionuclides and fill material combinations, and in these cases estimations have been made based on the available information and general knowledge about the radionuclides and the fill materials. In the modelling of the release from the waste items, this has been taken into account and reasonable variations have been incorporated to evaluate the uncertainty connected with the possible variations in solubility. As can be seen from Table B.13.5, the solubilities can differ up to two orders of magnitude dependent on the fill material. Where no difference is indicated, this is either due to the reported data showing no difference between the fill materials or due to lack of information. When a lower solubility is used for a specific fill material this will obviously lead to a lower overall release of that radionuclide into the environment.

The different fill materials also have different properties with respect to retain the radionuclides relatively to the water flow. This is either due to an enhanced ability to adsorb the radionuclides through incorporation into e.g. the clay mineral lattice (bentonite) or due to the chemical environment present in the fill material leading to precipitation of the radionuclides (cement-calcium granulate). Often it is not possible to completely distinguish between these types of mechanisms and in general all relevant mechanisms are "pooled" together and described by way of a retardation coefficient, K_D, when used for modelling of the delay of a release.

[84] That is, orders of magnitude for the solubilities

Table B.13.5 Suggested solubilities for different for the relevant radionuclides and fill materials, M. * As Ra;** As Co

	Bentonite fill	Cement - calcium calcium fill	No fill, Above & Near surface	No fill, MD and borehole
H	1 E^{2}	1 E^{2}	1 E^{2}	1E^{2}
C	1 E^{-3}	1 E^{-3}	1 E^{-3}	1 E^{-3}
Ca	1 E^{-2}	1 E^{-2}	1 E^{-2}	1 E^{-2}
Co	1 E^{-5}	1 E^{-5}	1 E^{-5}	1 E^{-5}
Ni	1 E^{-5}	1 E^{-5}	1 E^{-3}	1E^{-5}
Se	1 E^{-9}	1 E^{-10}	1 E^{-10}	1 E^{-10}
Sr	1 E^{-5}	1 E^{-5}	1E^{-4}	1 E^{-5}
Tc	1 E^{-9}	1 E^{-8}	1 E^{-9}	1 E^{-8}
Ag	1 E^{-9}	1 E^{-9}	1 E^{-5}	1 E^{-5}
Ba*	1 E^{-9}	1 E^{-9}	1E^{-7}	1 E^{-8}
Cs	1 E^{-1}	1 E^{-1}	1 E^{-1}	1 E^{-1}
Sm	1 E^{-9}	1 E^{-9}	1 E^{-7}	1 E^{-7}
Eu	1 E^{-6}	1 E^{-6}	1 E^{-6}	1 E^{-6}
Ir**	1 E^{-5}	1 E^{-5}	1 E^{-5}	1 E^{-5}
Pb	1 E^{-6}	1 E^{-6}	1 E^{-5}	1 E^{-6}
Ra	1 E^{-9}	1 E^{-9}	1 E^{-7}	1 E^{-8}
Ac	1 E^{-6}	1 E^{-6}	1 E^{-6}	1 E^{-6}
Th	1 E^{-9}	1 E^{-9}	1 E^{-8}	1 E^{-9}
Pa	1 E^{-8}	1 E^{-7}	1 E^{-7}	1 E^{-7}
U	1 E^{-9}	1 E^{-9}	1 E^{-9}	1 E^{-9}
Np	1 E^{-10}	1 E^{-10}	1 E^{-9}	1 E^{-9}
Pu	1 E^{-8}	1 E^{-8}	1 E^{-6}	1 E^{-8}
Am	1E^{-6}	1 E^{-7}	1 E^{-7}	1 E^{-7}
Cm	1E^{-6}	1 E^{-7}	1 E^{-7}	1 E^{-7}

Similarly as for the solubilities used in the calculations of release, K_D values have been collected for a number of articles and reports related to evaluation of retardation of radionuclides by different material (Atkinson, et al (1988); Berner (1992); Bradbury & Van Loon (1998); Herbert & Iden (2010); IAEA (2004); Nasif & Neyama (2003); Nykyri et al. (2008); Pulkkanen & Nordman (2010); Vieno & Nordman (1999)). Based on this, suggestions have been made for the values to be used in the prefeasibility study.

As for the solubilities, data is not available for all combinations of radionuclides and fill materials, and estimates have had to be made based on the general knowledge about the properties of the radionuclides and the fill materials. Table B.13.6 gives the suggested K_D values. Where two values are given, the first value is for near and above surface repositories, and the second value is for medium deep repositories and boreholes. The difference is due to the influence of the redox conditions on the K_D value. Where little information is available, no variation is suggested together with the use of relatively precautionary values.

When relevant K_D values are known for a fill material, this can be used to calculate the relative velocity of the radionuclide compared to the velocity of the water passing through the material, and if the thickness of the fill layer is also known, the retardation time of the fill material for a specific nuclide can be calculated[85]. The retardation times can then be used to estimate the overall time of release of radionuclides from the repository, e.g. in relation to their decay time. One of the assumptions behind these calculations is that the release from the fill material is reversible, once the concentration in the water drops beneath the saturated concentration of the radionuclide. This is a common assumption used in modelling of retardation.

[85] Also taking the overall possible retention into account.

Table B.13.6: Suggested K_D values for different fill/backfill materials, m3/kg.

	Bentonite	Sand	Concrete	Granulate
H	0	0	0	0
C	0	0	0	0
Ca	0.01	0.01	0.01	0.01
Co	0.03	0.01	0.01/0.1	0.01/0.1
Ni	0.03	0.01	0.01/0.1	0.01/0.1
Se	0.001	0	0	0
Sr	0.001	0.0003	0.001	0.001
Tc	0.1/1	0.1/1	0.1/1	0.1/1
Ag	0.001	0.001	0.001	0.001
Ba	0.001	0.001	0.001	0.001
Cs	0.02	0.002	0.002/0.02	0.02
Sm	1	0.3	0.3	0.3
Eu	0.1	0.1	0.1	0.1
Ir	0.005	0.005	0.005	0.005
Pb	0.5	0.05	0.05/0.5	0.05/0.5
Ra	0.005	0.005	0.005	0.005
Ac	0.5	0.5	0.5	0.5
Th	2	2	2	2
Pa	0.2	0.2	0.2/2	0.2/2
U	0.5	0.2	0.5/5	0.5/5
Np	4	1	1/5	1/5
Pu	4	1	1/5	1/5
Am	5	1	0.2/1	0.2/1
Cm	5	1	0.2/1	0.2/1

B.2 Current types of containers used at the Risø area

Only certain types of containers have been used for the temporary storage of radioactive waste at the Risø area. The different types of containers are described in SIS (2009). According to Danish Decommissioning, the current waste for final disposal at the repository is kept in the types of containers, described in the subsections below.

Steel containers

Special, thick walled steel containers have been designed for the decommissioning waste. There are two kinds of steel containers:

Type 1:
- Length x width x height (outer measures): 205 cm x 140 cm x 115 cm
- Material: 10 mm steel (front, roof, bottom, side panels)
- Opening: top lid. Steel plate tightened to the container by means of clamps
- Reinforcement profiles in the sides and bottom enables the piling of four containers each with the maximum gross weight
- Net volume: 3.2 m^3
- Maximum gross weight: 13000 kg

Type 2:
- Length x width x height (outer measures): 212 cm x 147 cm x 139 cm
- Material: 10 mm steel (front, roof, bottom, side panels)
- Opening: top lid. Steel plate tightened to the container by means of clamps
- Reinforcement profiles in the sides and bottom enables the piling of four containers each with the maximum gross weight.
- Net volume: 4.2 m^3
- Maximum gross weight: 13000 kg

ISO containers

Standard open top industrial package type 2 freight containers.

10-feet ISO containers in half height:
- Outer measures: Length x width x height: 299 cm x 244 cm x 130 cm
- Material: steel (front, roof, bottom, side panels)
- Opening: top lid. Equipped with rubber packing to keep the lid tight
- Designed for piling of four containers each with the maximum gross weight.
- Net volume: 7.6 m^3
- Maximum gross weight: 21000 kg

Steel drums

There are five types of steel drums:

A1: Concrete-lined, galvanized 210 l steel drum:
- Outer diameter x height: 59 cm x 88 cm
- Material: galvanized steel drum lined with minimum 5 cm concrete inner lining.

A1: Concrete-lined, painted 210 l steel drum:

- Outer diameter x height: 59 cm x 88 cm
- Material: painted steel drum lined with minimum 5 cm concrete inner lining.

B: Concrete-lined, painted 280 l steel drum for re-packing of A1 or A2 drums:

- Outer diameter x height: 63 cm x 93 cm
- Material: painted steel drum lined with 2 cm concrete inner lining

C: Un-lined, painted 210 l steel drum:

- Outer diameter x height: 59 cm x 88 cm
- Material: painted steel drum without lining

C1: Un-lined, painted 280 l steel drum:

- Outer diameter x height: 63 cm x 93 cm
- Material: painted steel drum without lining

Stainless steel containers for Coarse Control Arms

So-called Coarse Control Arms (CCA) were used in DR3 for the control of the neutron flux. They consist of cadmium panels in stainless steel frames and have been highly activated. The CCAs are packed in 130 l stainless steel containers (eight in each container) with the following characteristics:

- Outer diameter x height: 32 cm x 210 cm
- Material: stainless steel.
- Minimum 19 cm steel shielding in the lid.
- Arrangement for emptying liquids.

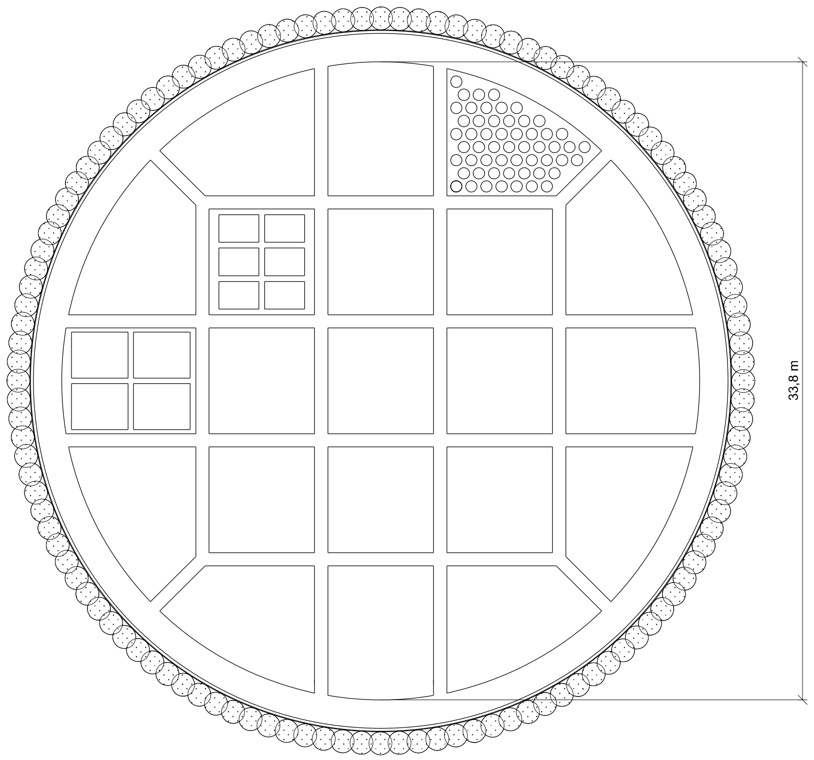
A-bins

A-bins are cylinder-shaped stainless steel containers (wall thickness of 1.25 mm). The inner diameter is 22 cm, the height is 87 cm. The volume of an A-bin is approximately 30 l (for further details on A-bins, see Danish Decommissioning (2009).

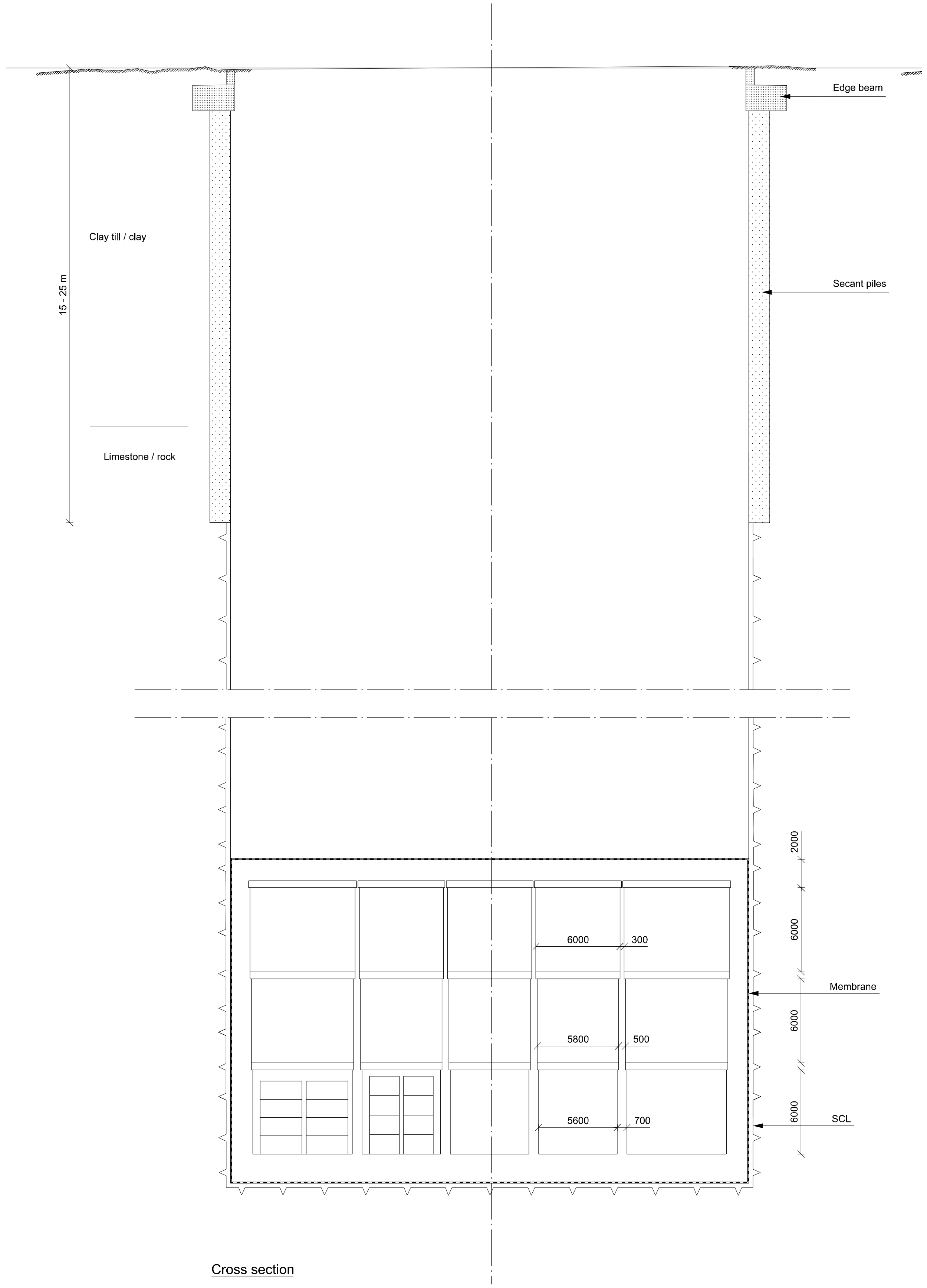
Flasks

A specific type of flask is used for holding the nuclear solution from DR1.

Appendix C: Repository design, details



Ground plan



Cross section

Note :

Alle measurements are in mm, unless otherwise noted.

Rev.	Date	Description	Designed	Checked	Approved

Dansk Dekommissionering

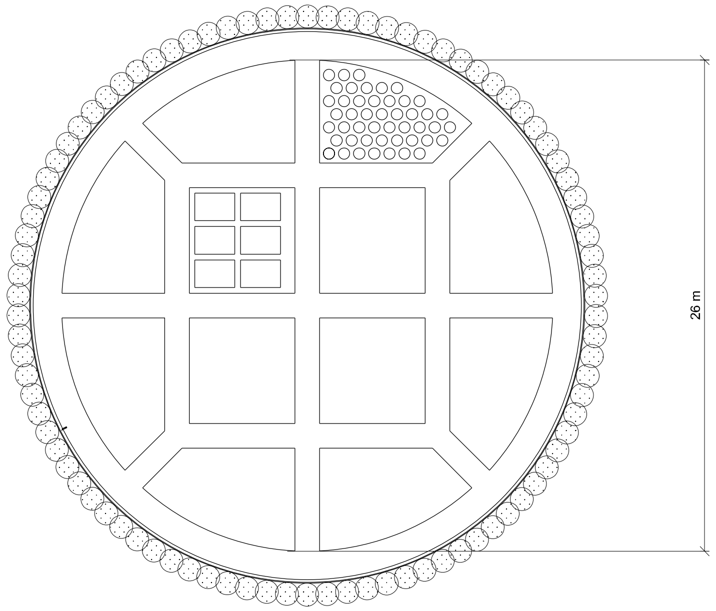
Pre-feasibility study for final disposal of radioactive waste

Medium depth repository, shaft operated from ground level

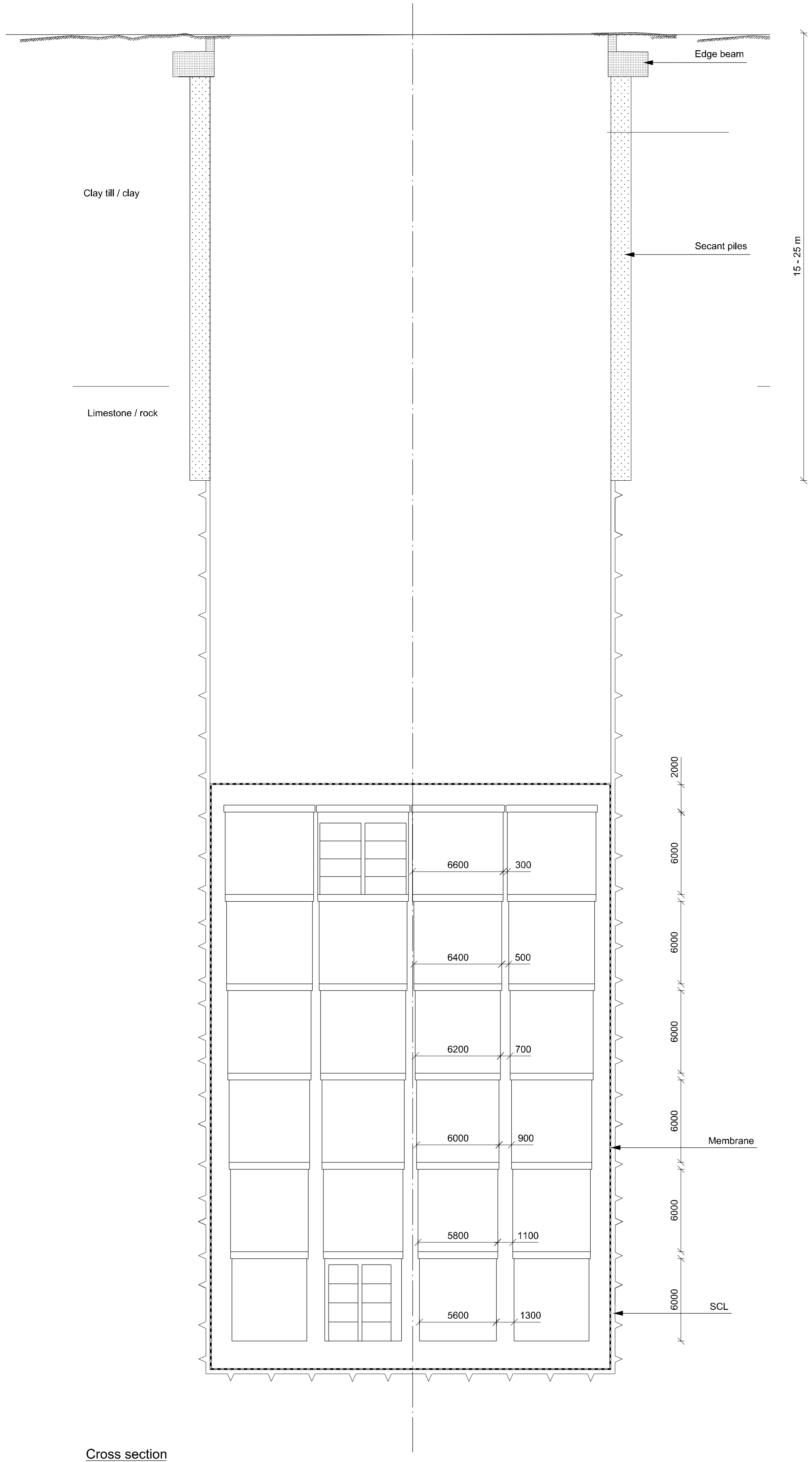
Reversible, 33.8 m, secant piles and SCL

Project no.	P-72024
Designed	FLP / VRSA
Checked	ANKU
Approved	CASK
Scale	-
Date	28.09.2010
Drawing no.	1-06
Rev.	0

COWI



Ground plan



Cross section

Note :

All measurement are in mm, unless otherwise noted.

Rev.	Date	Description	Designed	Checked	Approved

Dansk Dekommissionering

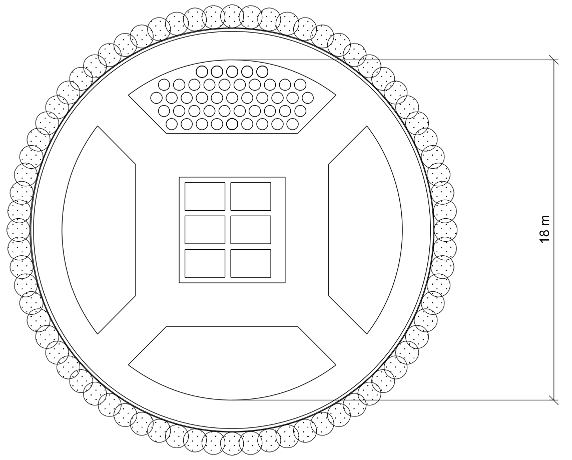
Pre-feasibility study for final disposal of radioactive waste

Medium depth repository, shaft operated from ground level

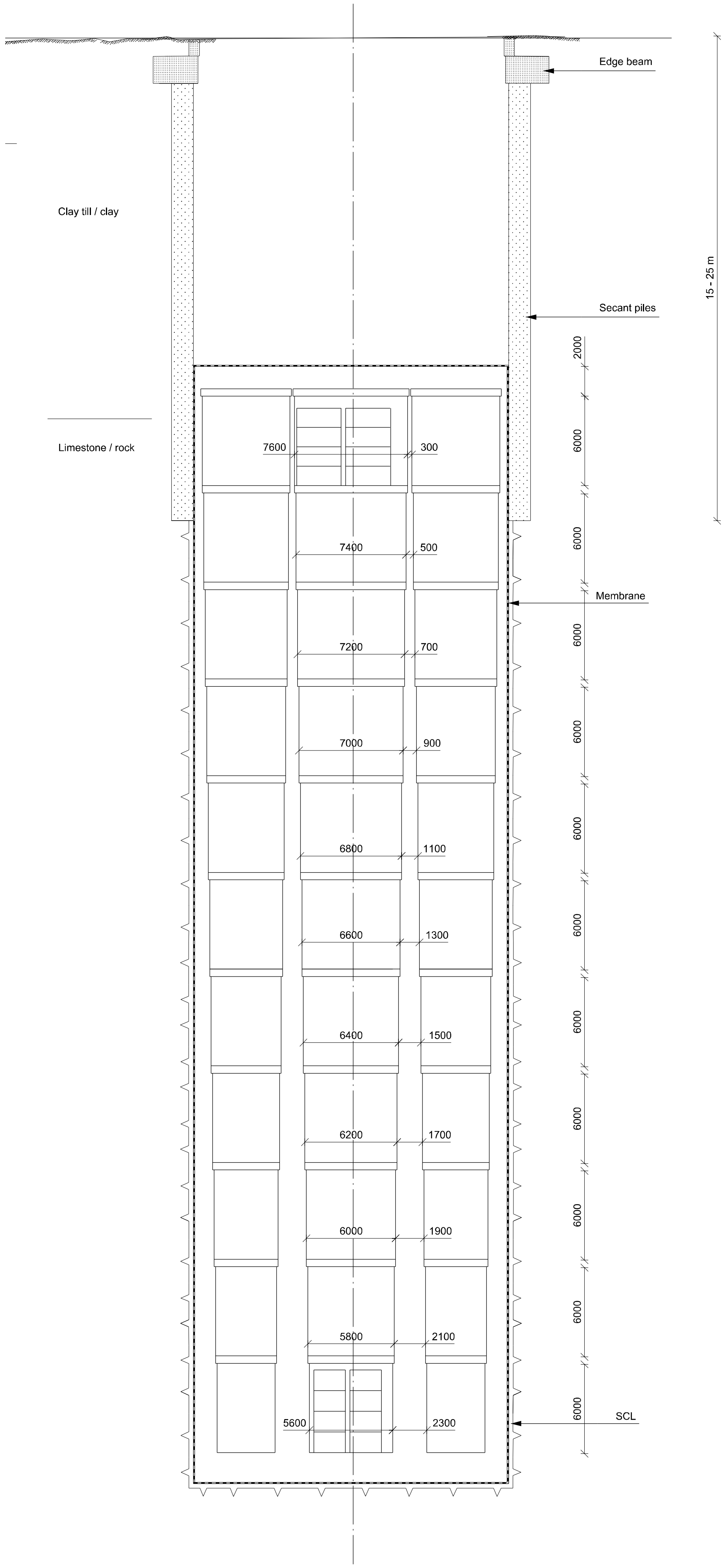
Reversible, diameter 26 m, secant piles and SCL

Project no.	P-72024
Designed	FLP / VRSA
Checked	ANKU
Approved	CASK
Scale	-
Date	28.09.2010
Drawing no.	1-07
Rev.	0

COWI



Ground plan



Cross section

Note :

All measurements are in mm, unless otherwise noted.

Rev.	Date	Description	Designed	Checked	Approved

Dansk Dekommissionering

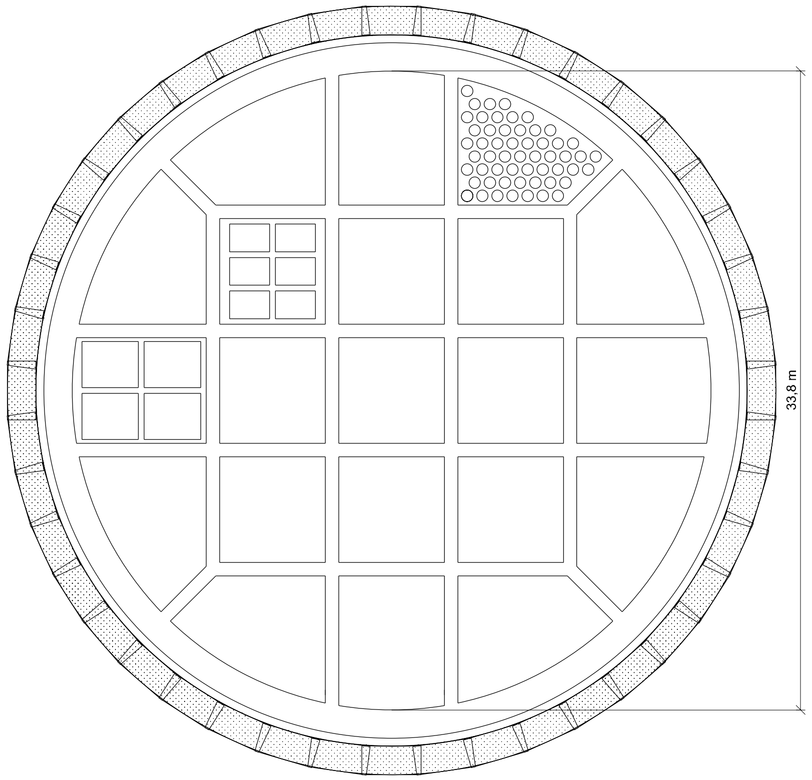
Pre-feasibility study for final disposal of radioactive waste

Medium depth repository, shaft operated from ground level

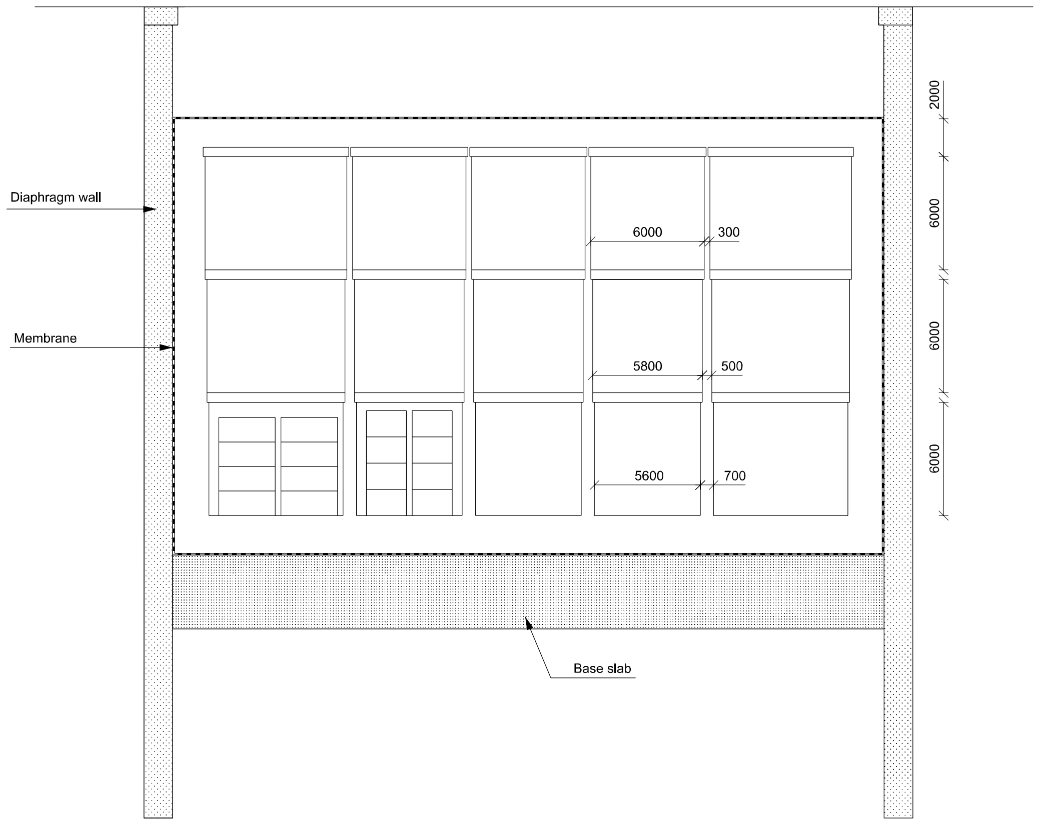
Reversible, diameter 18 m, secant piles and SCL

Project no.	P-72024
Designed	FLP / VRSA
Checked	ANKU
Approved	CASK
Scale	-
Date	28.09.2010
Drawing no.	1-08
Rev.	0

COWI



Ground plan



Cross section

Note :

All measurements are in mm, unless otherwise noted.

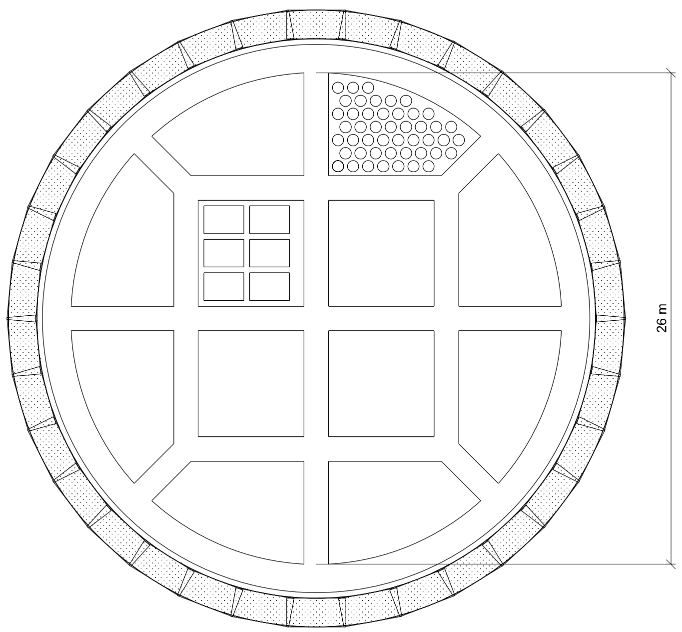
Rev.	Date	Description	Designed	Checked	Approved

Dansk Dekommissionering
Pre-feasibility study for final disposal of radioactive waste

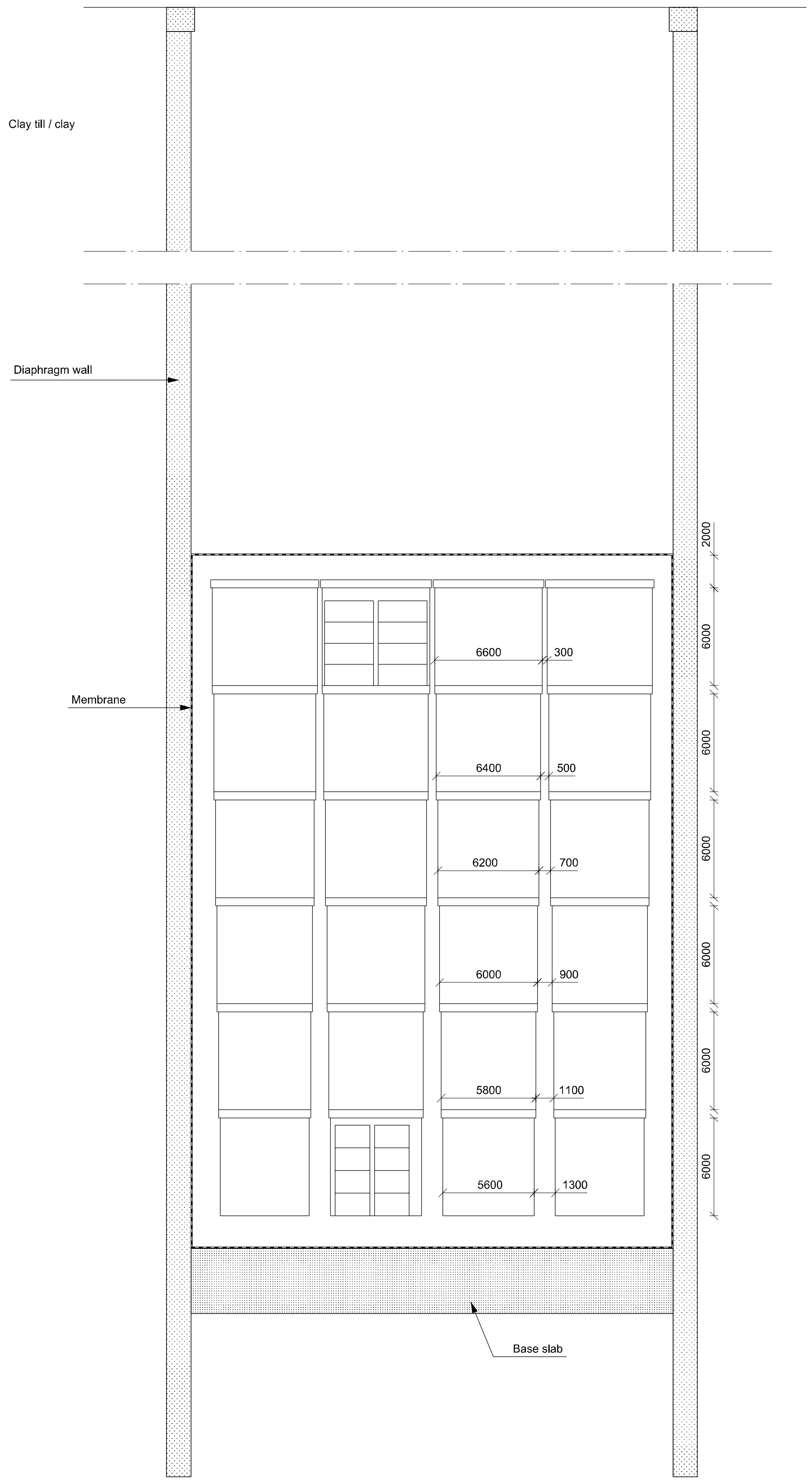
Medium depth repository, shaft operated from ground level
Reversible, diameter 33.8 m, diaphragm wall

Project no.	P-72024
Designed	FLP / VRSA
Checked	ANKU
Approved	CASK
Scale	-
Date	28.09.2010
Drawing no.	1-09
Rev.	0

COWI



Ground plan



Cross section

Note :

All measurements are in mm, unless otherwise noted.

Rev.	Date	Description	Designed	Checked	Approved

Dansk Dekommissionering

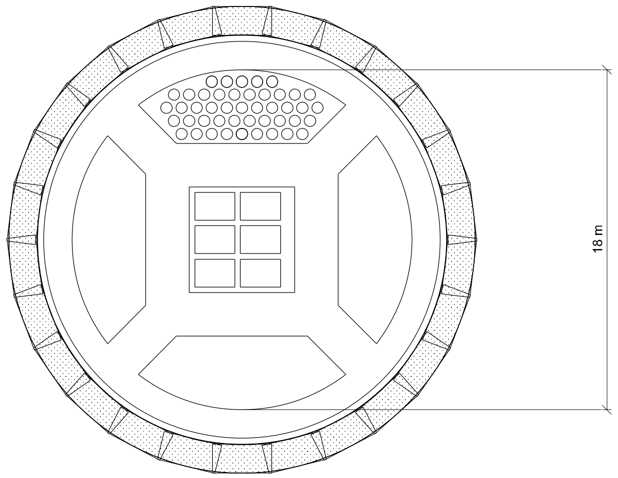
Pre-feasibility study for final disposal of radioactive waste

Medium depth repository, shaft operated from ground level

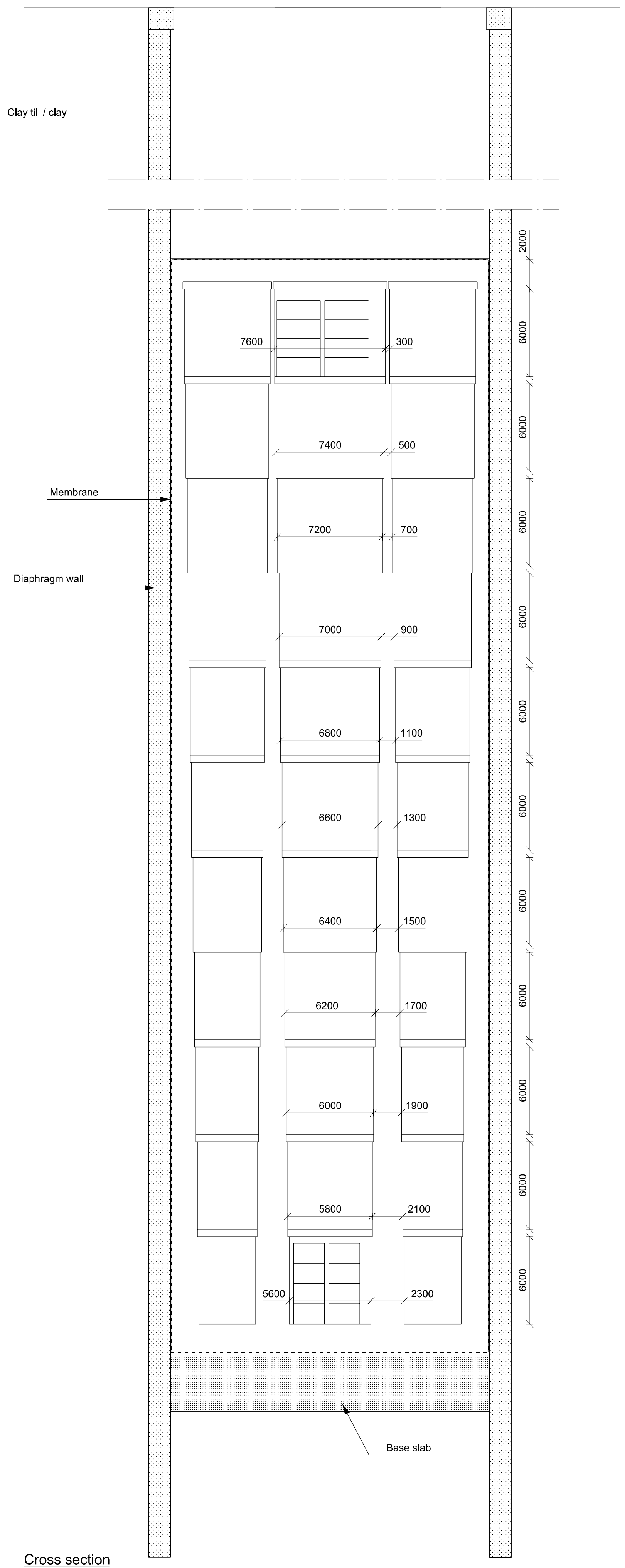
Reversible, diameter 26 m, diaphragm wall

Project no.	P-72024
Designed	FLP / VRSA
Checked	ANKU
Approved	CASK
Scale	-
Date	28.09.2010
Drawing no.	1-10
Rev.	0

COWI



Ground plan



Cross section

Note :

All measurements are in mm, unless otherwise noted.

Rev.	Date	Description	Designed	Checked	Approved

Dansk Dekommissionering

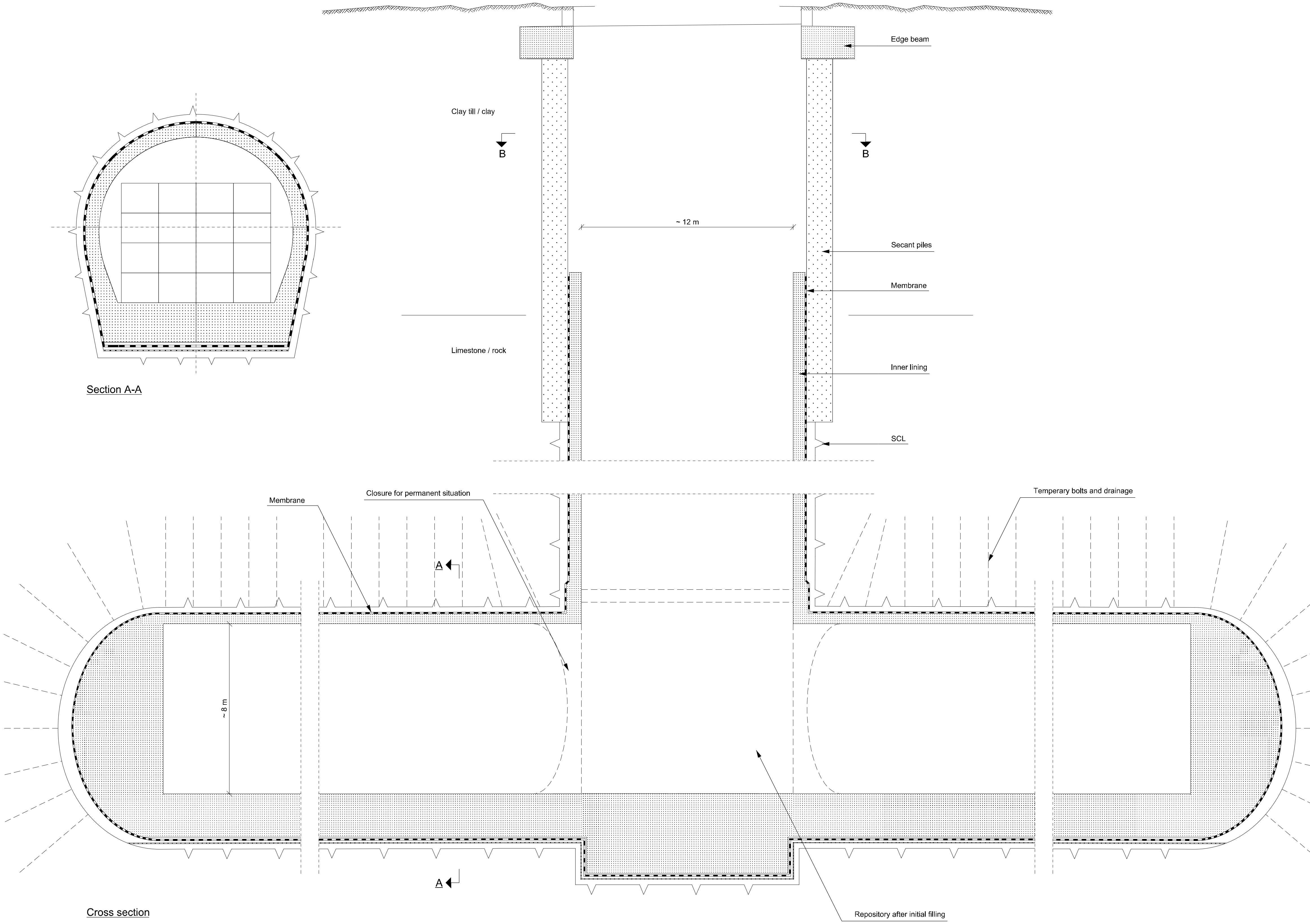
Pre-feasibility study for final disposal of radioactive waste

Medium depth repository, shaft operated from ground level

Reversible, diameter 18 m, diaphragm wall

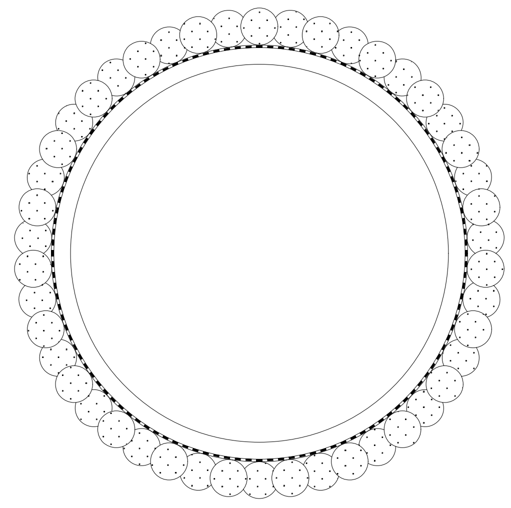
Project no.	P-72024
Designed	FLP / VRSA
Checked	ANKU
Approved	CASK
Scale	-
Date	28.09.2010
Drawing no.	1-11
Rev.	0

COWI



Section A-A

Cross section



Section B-B

Note :

All measurement are in mm, unless otherwise noted.

Rev.	Date	Description	Designed	Checked	Approved

Dansk Dekommissionering
Pre-feasibility study for final disposal of radioactive waste

Medium depth repository, cavern operated inside repository
Secant piles and SCL

Project no.	P-72024
Designed	FLP / VRSA
Checked	ANKU
Approved	CASK
Scale	-
Date	28.09.2010
Drawing no.	1-12
Rev.	0

COWI

Appendix D: Repository model, details

The repository groundwater model shall provide a source term for the subsequent groundwater and dose models. The source term shall represent the amount released per year of specific radionuclides of interest. Due to simplifications in the subsequent models the source term shall be constant in time. However, a "0" contribution is permitted from the start, modelling a delay in start of release and a "0" contribution is permitted when the amount of nuclide present has been exhausted due to wash out.

D.1.1 Simplifications

The following simplifications regarding the estimate of the source term have been made:

- The general decay of the radionuclides is considered in the final dose calculations only, because it has to be considered for how long time a particular radionuclide has travelled before it reaches the individual recipients. Further some of the radionuclides decay into daughters with quite different transport characteristics (this is all managed in the dose model).

- In general a constant concentration of the nuclides in the source term may be assumed. A requirement is though, that the concentration must be the maximum of any concentration experienced. This is conservative.

- The model estimates the time until the release of specific nuclides from specific types of waste units begin. Until the first waste package opens the concentration of the nuclide in the source term is set to "0". This is conservative.

- The model keeps an account of the total amount of each nuclide in each waste package. When the amount present in all waste packages has been washed out the concentration in the source term is set to "0". This is conservative.

Within the time and resources set aside for the task, a very detailed model taking into account all variations in individual waste units is not feasible. Also information supporting such a model is not available for the present. Thus simplifications were made.

- The source term is estimated considering a number of variable parameters. Among these are 11 types of repository, 4 geologies, 3 types of backfill, 5 types of fill and 3 scenarios for distribution of the waste units between a borehole and an upper repository. 21 types of waste have been considered.
- When the repository is closed and the groundwater level no longer controlled the repository will become water filled. The time to water filling is modelled based on the groundwater flux and the depth of the repository.

- The presence of water inside the repository will cause the steel of waste units to corrode and open the waste units. When a waste unit is found to be open to water, washout of nuclides may start. All waste units of the same type are considered to open at the same time.
- The water flow rate through repository and waste units is determined by the hydraulic conductivity of soil and packaging materials. A simple model considering the parameters for soil and materials is set up. Degradation of structures and materials is modelled by an increase in the hydraulic conductivity of the structure or material.
- The water volume in the repository corresponds to the volume of voids in the repository, voids in the waste units and the porosity of materials and fill.
- The amount of water flowing through the repository and waste units is determined by their cross section.
- All elements have to be dissolved to be transported by the groundwater
- Water flowing through the waste units is assumed to be saturated in the elements of interest.
- The retention time in fill and backfill materials is estimated by a simple model depending on the K_d value for the specific element and its near field.
- The concentration of elements washed out of the repository does not consider decay, i.e. inside the repository the fraction of radionuclides versus other nuclides of the same element remains the same in time.
- The model identifies a maximum value. The model is discrete in time not continuous. The model considers the moments 10; 30; 100; 300; 1,000; 3,000 and 10,000 years. To ensure that all tops of short duration falling within intervals are identified the model keeps account of the time for start and end of release of each nuclide in each waste type. Tops are ascribed to the moment before its arrival.

D.2 Results

The repository groundwater model provides a source term for the subsequent groundwater and dose models. The source term represents the amount released per year of specific radionuclides of interest. Due to simplifications in the subsequent models the source term is required to be constant in time and decay is not yet considered. The source term represents the maximum concentration observed. This is conservative. However, a "0" contribution is permitted from the start and a "0" contribution is permitted when the amount of nuclide present has been exhausted.

D.2.1 Source term

Figure and Figure D.13.1 presents the estimated intermediate results for nuclide concentrations. The subsequent models require this to be expressed in the unit GBq/m^3. The concentrations have to be multiplied with the water flux to derive the amount. Thus the source term is expressed in the unit GBq/year - although decay is not yet considered.

The source term is based on the maximum of the concentrations estimated for the individual nuclides. The figures indicate the level of conservatism. The very high values need not appear at an early time as the model considers the time until deterioration of individual waste units.

Figure D.1 presents results for a repository with no backfill and no fill in waste units. Thus no retention is considered due to sorption onto these materials. Due to the high porosity of the repository the water flux is high, nearly 8 m^3 per year.

Figure D.2 presents a repository backfilled with bentonite. Most nuclides are now retained within the repository for a period longer than the 10,000 years of interest. Only elements like H, C and Sr are washed out. The porosity of the repository is now much lover and the water flux is but 0.07 m^3 per year.

It is noted that the concentration of some elements is lower when the water flux is high. This is because the model considerers the total amount in each waste unit and do not wash out more than present.

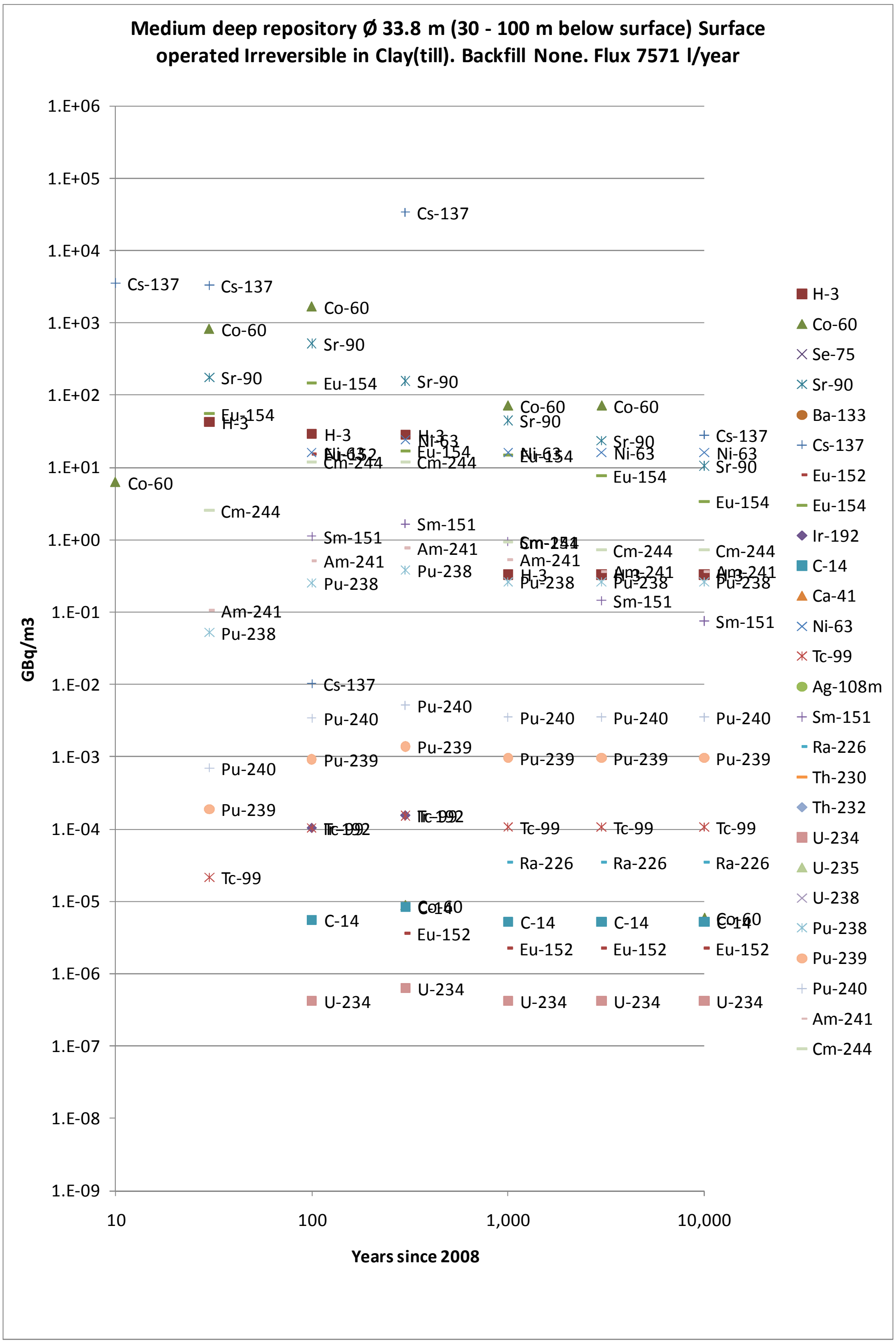


Figure D.1 *Intermediate results for concentration of nuclides in a repository with no backfill and no fill in waste units*

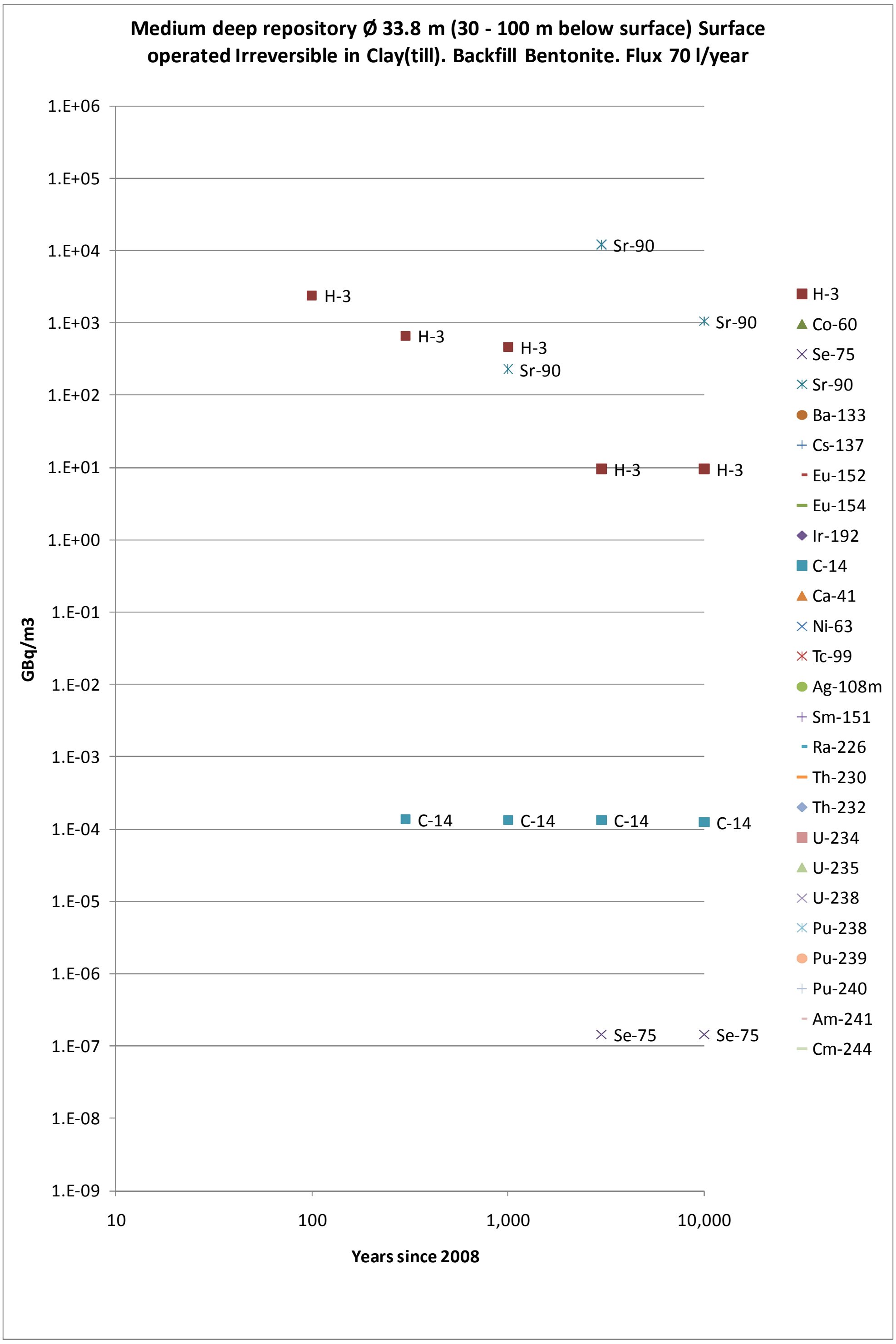


Figure D.13.1 Intermediate results for concentration of nuclides in a repository with bentonite backfill and no fill in waste units.

Figure D.3 to Figure D.13.6 present the sum of the source terms for a medium deep repository versus soil type, backfill and fill.

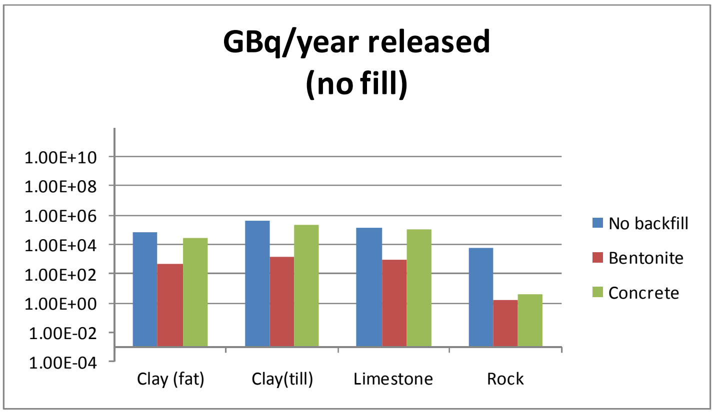


Figure D.13.2 Sum of source terms for a repository of type 3 versus soil and backfill. No additional fill in waste units

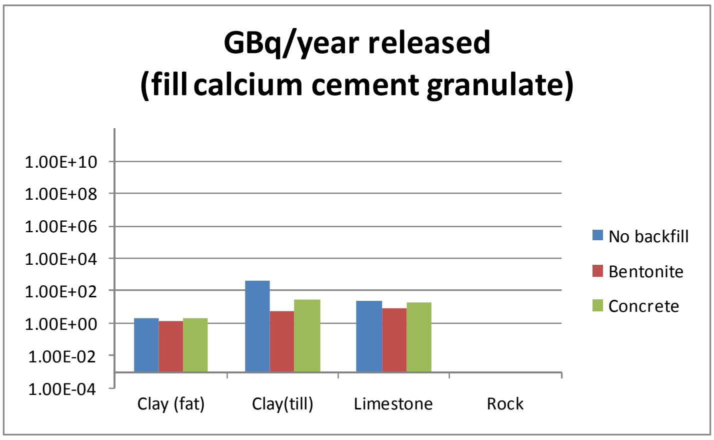


Figure D.13.3 Sum of source terms for a repository of type 3 versus soil and backfill. Voids in waste units are all filled with calcium cement granulate

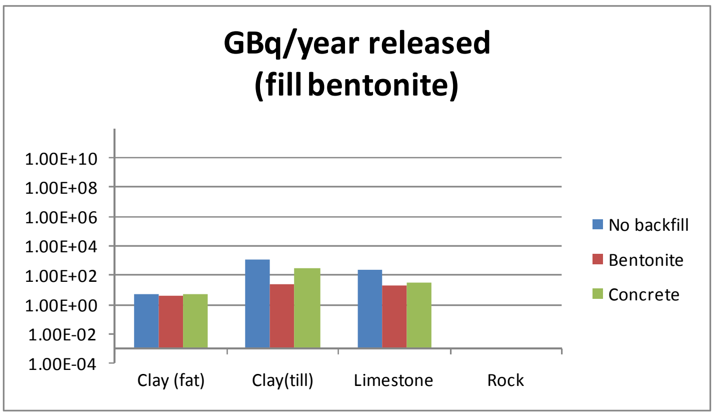


Figure D.13.4 Sum of source terms for a repository of type 3 versus soil and backfill. Voids in waste units are all filled with bentonite

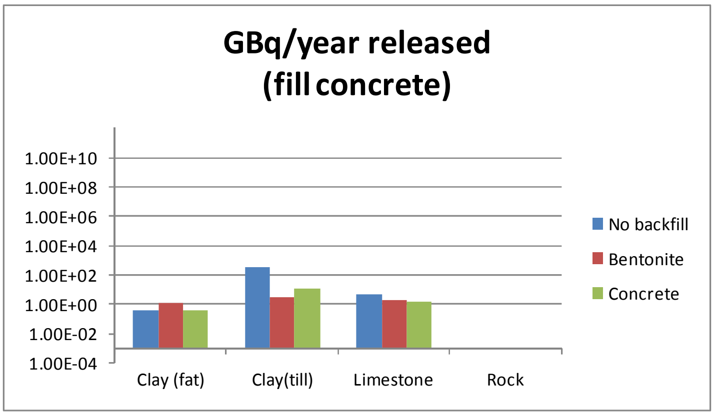


Figure D.13.5 Sum of source terms for a repository of type 3 versus soil and backfill. Voids in waste units are all filled with concrete

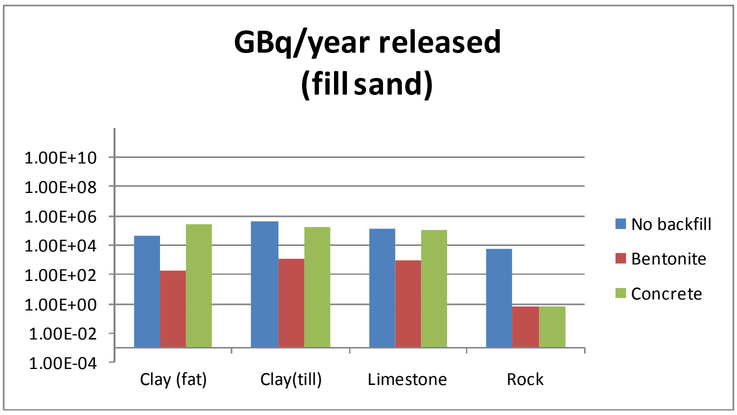


Figure D.13.6 Sum of source terms for a repository of type 3 versus soil and backfill. Voids in ISO containers are filled with sand

Figure D.8 presents the sum of the source terms for the near surface (rep 2) and the medium deep repositories (rep 3, 4 and 5). The values are estimated for repositories with no backfill and no additional fill in waste units.

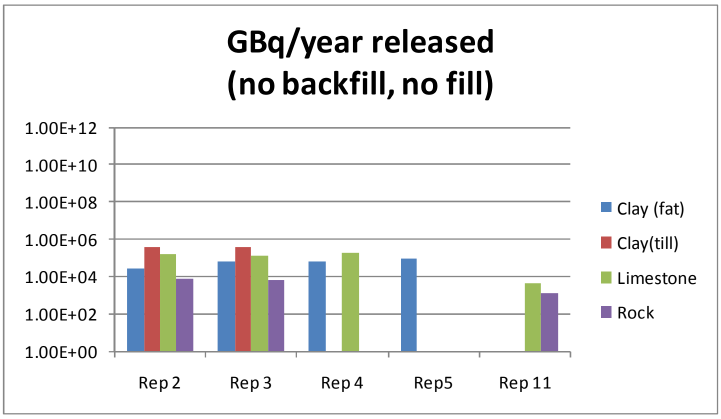


Figure D.13.7 Sum of source terms for near surface and medium deep repositories

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Figure D.9 presents the source terms for a medium deep repository in clay(till) with no backfill and no fill. Results for waste unit distributions for scenario 1, 2 and 3 are indicated.

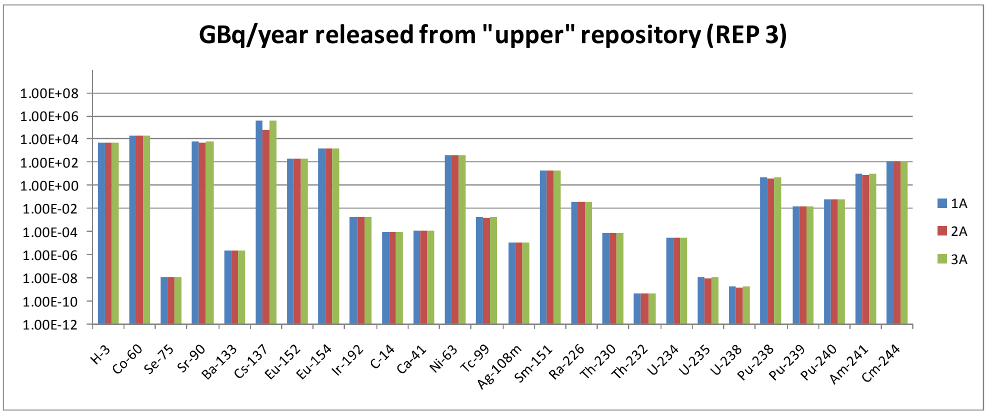


Figure D.13.8 Source terms for medium deep repository located in clay(till)

Figure D.10 presents the results for the remaining waste located in a borehole in limestone.

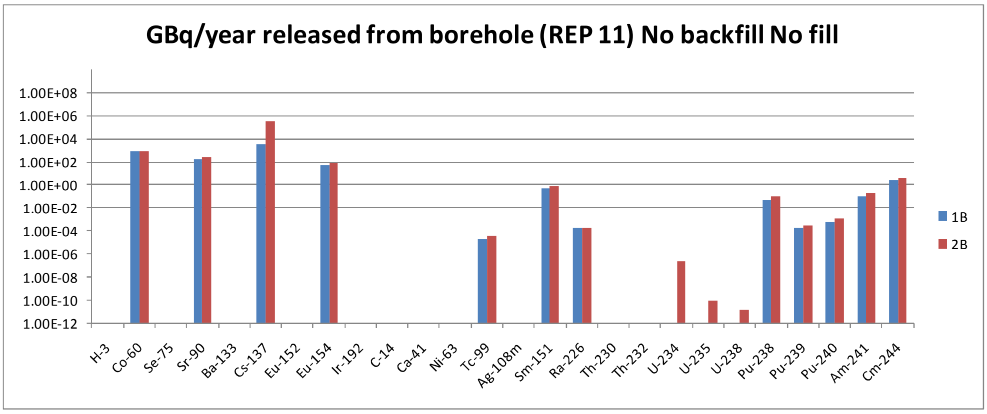


Figure D.13.9 Source terms for remaining waste units in borehole located in limestone.

D.3 Uncertainties

Uncertainty simulations were performed using the EXCEL add-in @RISK. Simulations were performed for a medium deep repository located in clay (till).

Below a number of examples are presented to give an overall understanding of the uncertainties of the values in the source term. The importance of variations in hydraulic conductivity and the variations due to implementation of backfill is examined. The variables considered are the water flux, the concentration of representative elements, the time to washout elements present in small amounts and the retention time in backfill.

It must be noted that the calculations performed generating the actual source term for the subsequent groundwater and dose models are based on most likely values alone and not a simulated mean value.

Geometry

In the simulations geometrical parameters known when the repository is planned are kept fixed. Such may be the size and depth of the repository and the thickness of its concrete walls. Other parameters where variations are foreseen like e.g. the width of backfill and fill layers are described by a three point estimate and a triangular distribution.

Hydraulic conductivity

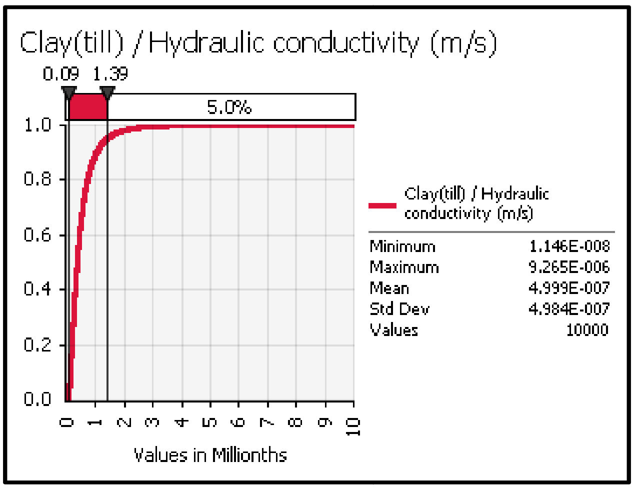
The hydraulic conductivity of the soil around the repository determines the maximum possible groundwater flow rate. The uncertainty on this parameter is very large when nothing is known on the specific conditions at an actual location. The value of this parameter may for the same type of soil be 100 times smaller than the indicated mean or 100 times larger.

When the location of the repository is known the hydraulic conductivity may be determined with a smaller uncertainty. To assess the importance of the parameter simulations were performed using a probability function which vary in the interval 10 times less to 10 times larger than the mean. Also simulations were prepared with a fixed parameter.

The table below indicates the probability that the hydraulic conductivity is the indicated number of times greater than the mean value.

	variable hydraulic conductivity	fixed hydraulic conductivity
p > 10 mean	0.001	0
p > 100 mean	0	0

Variation on hydraulic conductivity



Fixed hydraulic conductivity

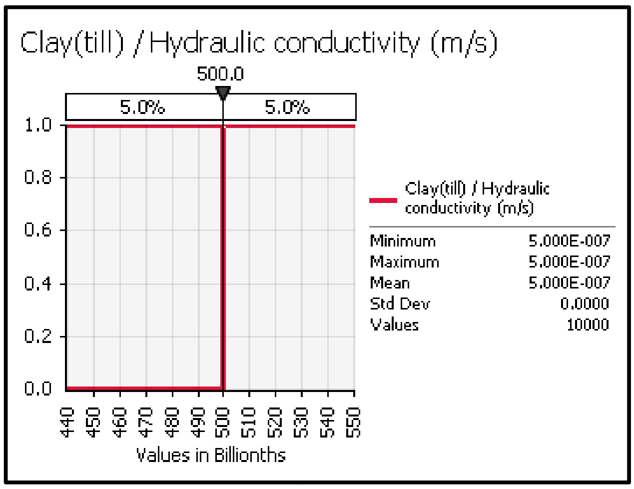


Figure D.13.10 Probability functions for the hydraulic conductivity. Example for clay (till)

Flux of groundwater through repository

The flux of groundwater through the repository depends on the hydraulic conductivity of the soil around the repository and the materials within. When the values of hydraulic conductivities are uncertain and no backfill is implemented the 5% and 95% fractiles for the groundwater flux in the selected repository and soil are 2 m^3/year and 35 m^3/year. When the hydraulic conductivity is fixed the mean value remains the same but the interval is narrowed and the fractiles are 10 m^3/year and 15 m^3/year. When the hydraulic conductivity is allowed to vary, the ratio between maximum and minimum flux is about 1,000, when fixed the ratio is about 2. The table below indicates the probability that the flux of water is the indicated number of times greater than the mean value.

	variable hydraulic conductivity	fixed hydraulic conductivity
p > 10·mean	0.001	0
p > 100· mean	0	0

No backfill in repository, no fill in waste units, variation on hydraulic parameters

No backfill in repository, no fill in waste units, fixed hydraulic parameters

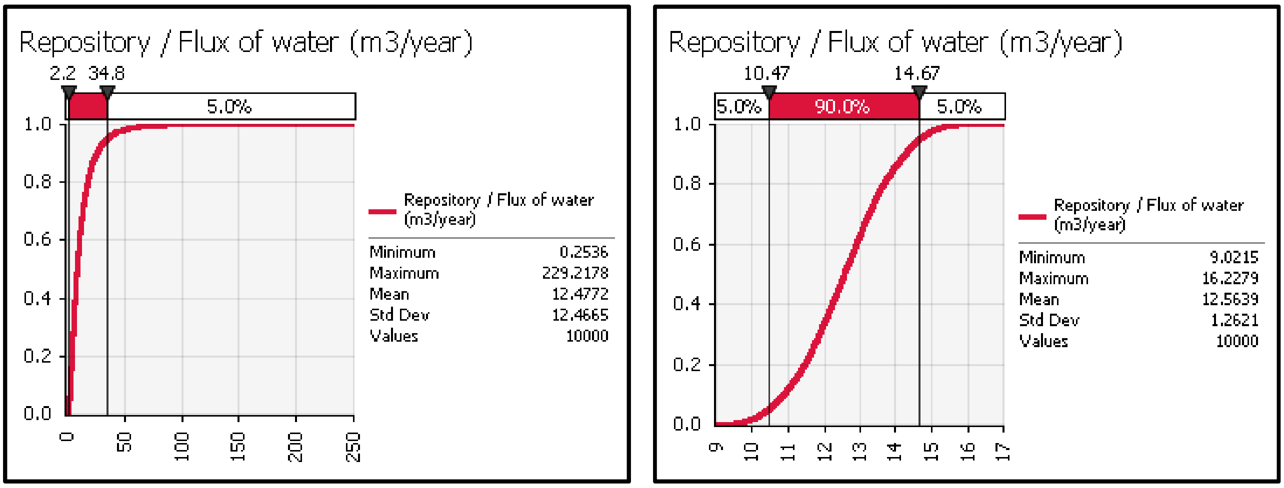
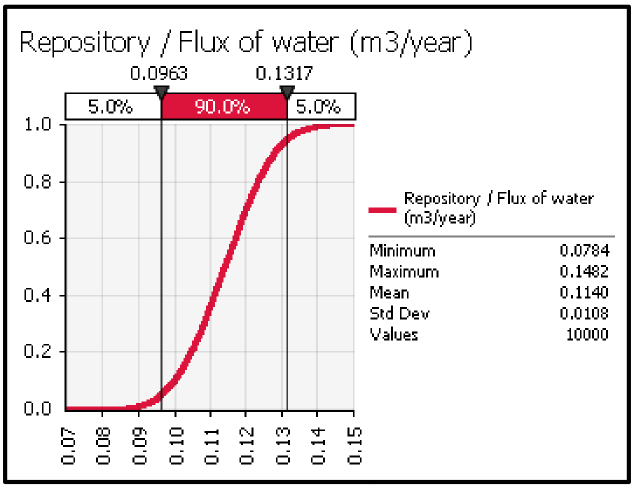


Figure D.13.11 Variations in groundwater flux in repository without backfill

Introducing backfill reduces the groundwater flux considerably as the overall porosity of the repository decreases. Bentonite backfill reduces the flux with a factor of about 100. The factor for concrete is about 15. The simulations were performed with a fixed hydraulic conductivity. In both simulations the ratio between the minimum and maximum values is still about 2.

Bentonite backfill in repository, no fill in waste units, fixed hydraulic parameters



Concrete backfill in repository, no fill in waste units, fixed hydraulic parameters

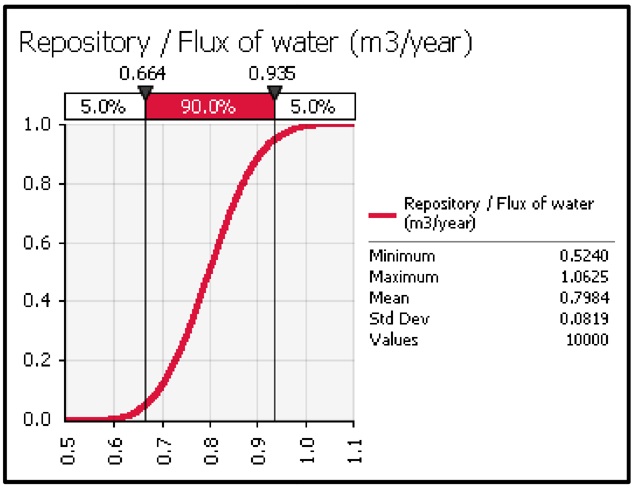


Figure D.13.12 Variations in groundwater flux in repository with backfill

Concentrations

The source term is an intermediate concentration in the unit GBq/m3. For the reasons described above, this intermediate concentration does not yet consider decay of the radionuclide. The concentration has subsequently to be multiplied with the water flux to indicate the amount washed out per year.

H-3

The radionuclide H-3 is only present in small amounts. Although the ratio of H-3 versus ordinary hydrogen is considered to be 1 in a million, the amount of water required to transport the radionuclide is but a few litres.

Figure D.14 and Figure D.15 presents the results for the simulation of the source term concentration of H-3.

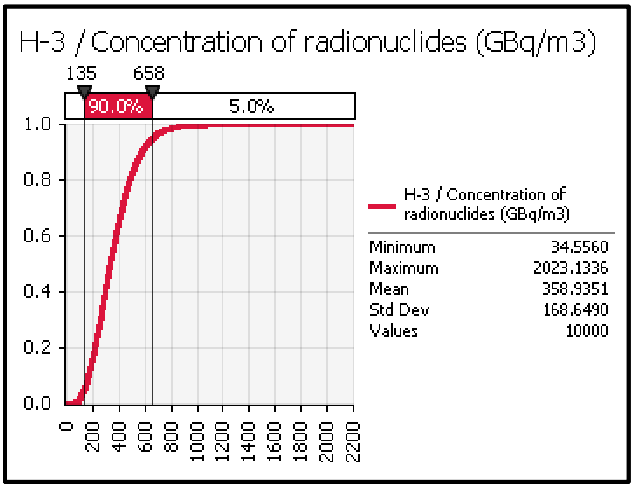
Comparing the results in Figure D.14 and Figure D.15, the concentration is not much influenced by the magnitude of the hydraulic conductivity of the soil and materials. Mean value and also the fractiles remain nearly the same when the hydraulic conductivity is fixed.

The ratio between the maximum and minimum values is about 60 when the value of the hydraulic conductivity is allowed to vary and about 30 when fixed.

The table below indicates the probability that the parameter is the indicated number of times greater than the mean value.

	variable hydraulic conductivity	fixed hydraulic conductivity
p > 10·mean	0	0
p > 100·mean	0	0

No backfill in repository, no fill in waste units, variation on hydraulic parameters



No backfill in repository, no fill in waste units, fixed hydraulic parameters

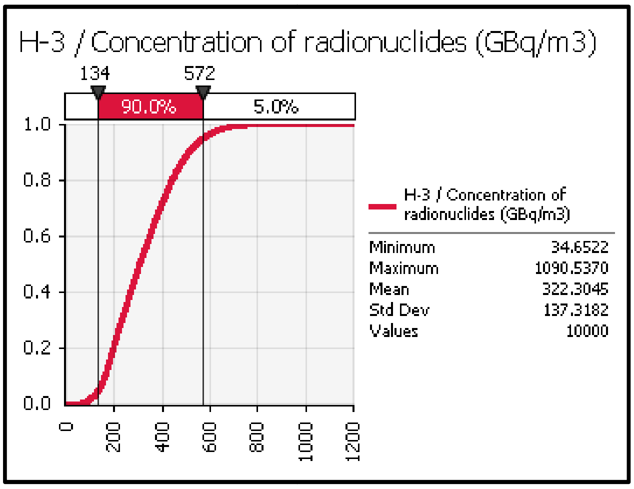


Figure D.13.13 Variations in intermediate result for the H-3 concentration

Introducing backfill appears to cause the H-3 concentration to increase. The amount of H-3 present is small and will be washed out within a short period. Considering the water flux the amount washed out is the same. The model considerers the total amount in each waste unit and do not wash out more than present.

The ratio between the maximum and minimum values is about 10 when the backfill is bentonite and about 30 when the backfill is concrete. The table below indicates the probability that the parameter is the indicated number of times greater than the mean value.

	bentonite fixed hydraulic conductivity	concrete fixed hydraulic conductivity
p > 10 · mean	0	0
p > 100· mean	0	0

Bentonite backfill in repository, no fill in waste units, fixed hydraulic parameters

Concrete backfill in repository, no fill in waste units, fixed hydraulic parameters

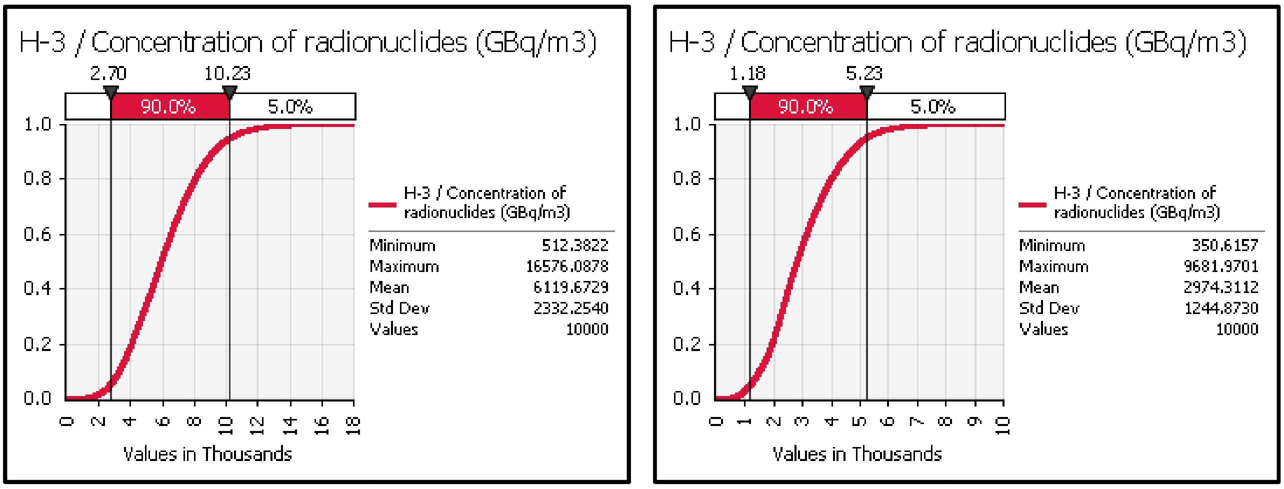


Figure D.13.14 Variations in intermediate result for the H-3 concentration when introducing backfill

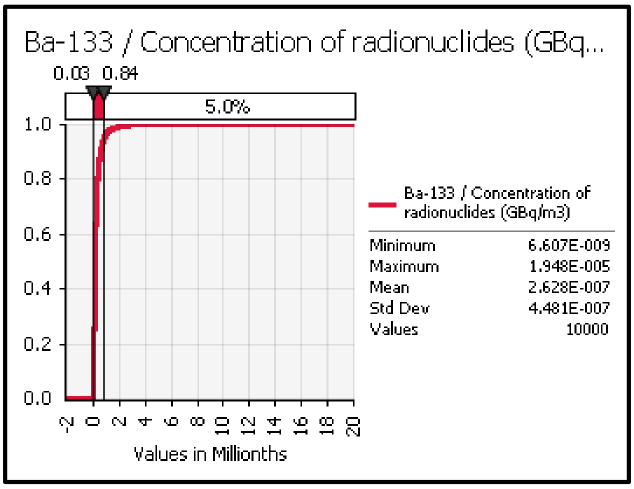
Ba-133

Figure D.16 and Figure D.17 presents the results of a simulation for the source term concentration of Ba-133.

Like for H-3 this variation is little influenced by variations in the hydraulic conductivities. However the ratio between maximum and minimum values estimated when the hydraulic conductivity varies is in the order of 3,000. The ration for a fixed conductivity is in the order of 600. The table below indicates the probability that the parameter is the indicated number of times greater than the mean value.

	variable hydraulic conductivity	fixed hydraulic conductivity
p > 10 · mean	0.005	0.001
p > 100· mean	0	0

No backfill in repository, no fill in waste units, variation on hydraulic parameters



No backfill in repository, no fill in waste units, fixed hydraulic parameters

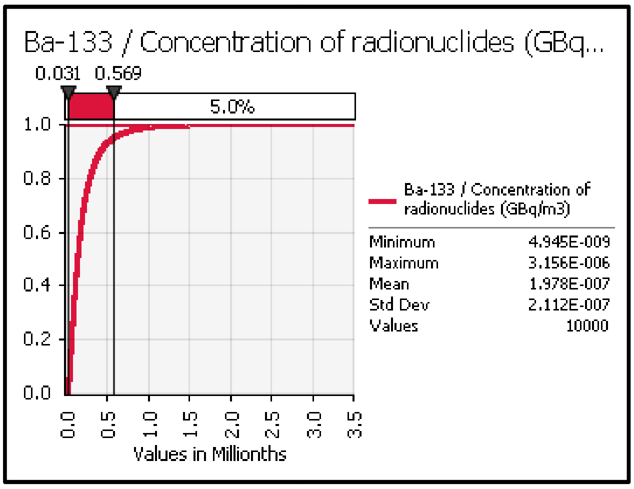
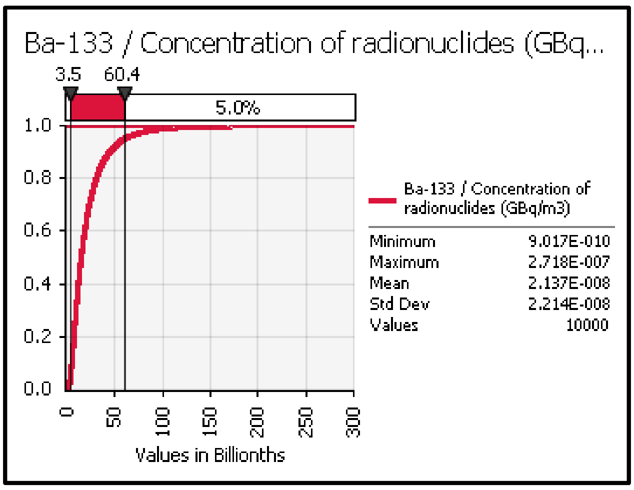


Figure D.13.15 Variations in intermediate result for the Ba-133 concentration

Introducing bentonite or concrete backfill in the repository retains the Ba-133 inside the repository. However Ba-133 is still released within the period of interest. The table below indicates the probability that the parameter is the indicated number of times greater than the mean value.

	bentonite fixed hydraulic conductivity	concrete fixed hydraulic conductivity
p > 10 · mean	0.001	0.001
p > 100· mean	0	0

Bentonite backfill in repository, no fill in waste units, fixed hydraulic parameters



Concrete backfill in repository, no fill in waste units, fixed hydraulic parameters

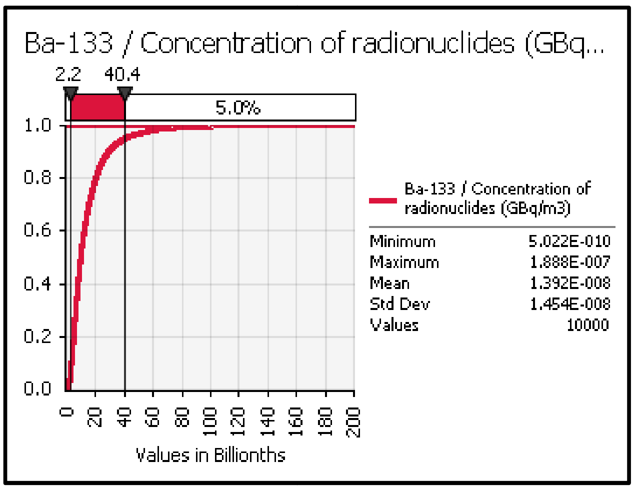


Figure D.13.16 Variations in intermediate result for the Ba-133 concentration introducing backfill

U-238

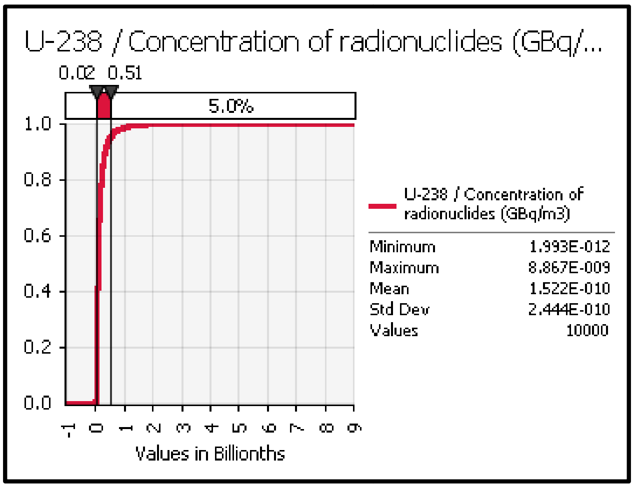
Figure D.18 and Figure D.19 present the results of a simulation for the source term concentration of U-238.

When the repository is not backfilled the variation in the estimated concentration is large. The variation between maximum and minimum estimates is a factor of about 1,000.

Like for H-3 and Ba-133 this variation is little influenced by variations in the hydraulic conductivities. The table below indicates the probability that the parameter is the indicated number of times greater than the mean value.

	variable hydraulic conductivity	fixed hydraulic conductivity
p > 10 · mean	0.007	0.007
p > 100· mean	0	0
p > 1000·mean	0	0

No backfill in repository, no fill in waste units, variation on hydraulic parameters



No backfill in repository, no fill in waste units, fixed hydraulic parameters

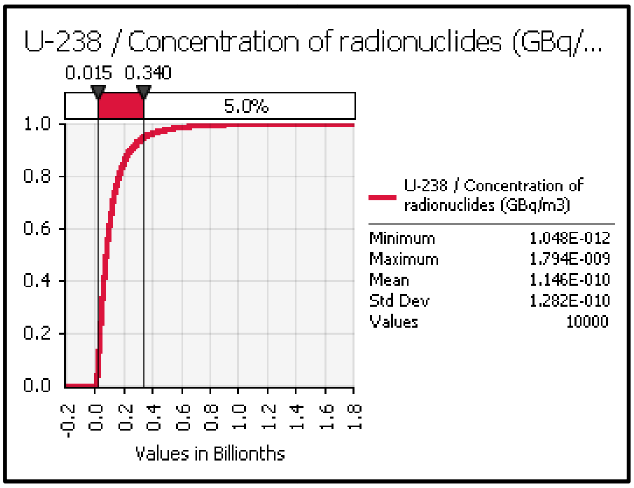


Figure D.13.17 Variations in intermediate result for the U-238 concentration

Time to depletion of radionuclide

The time to wash out radionuclides present in small amounts from specific waste units depends on the water flux and thus on the hydraulic conductivity of soil and materials.

Figure D.19 presents the time to washout H-3 from the waste units of type 4.The ratio between maximum and minimum is 2000 when the hydraulic conductivity is allowed to vary and 50 when fixed. The mean value is within the same order of magnitude. The table below indicates the probability that the parameter is the indicated number of times greater than the mean value.

	variable hydraulic conductivity	fixed hydraulic conductivity
p > 10 · mean	0.005	0.002
p > 100· mean	0	0

No backfill in repository, no fill in waste units, variation on hydraulic parameters

No backfill in repository, no fill in waste units, fixed hydraulic parameters

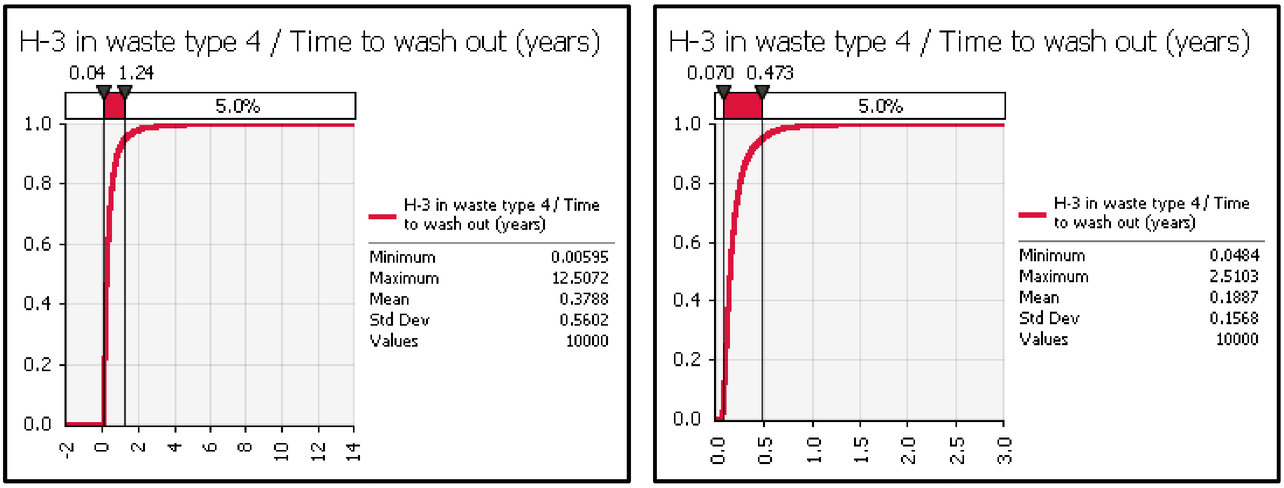


Figure D.13.18 Variations in time to wash out H-3

Retention time

The variability in retention time is illustrated by the simulated values for Ba-133 on Figure D.10. The mean retention time is about 1800 years for bentonite backfill while only about 300 years, when concrete is used. In both cases the ratio between maximum and minimum is a factor of 2. These calculations do not fully take the absolute retention capacity of a given amount of fill into account, since this will depend on the actual configuration.

Bentonite backfill in repository, no fill in waste units, fixed hydraulic parameters

Concrete backfill in repository, no fill in waste units, fixed hydraulic parameters

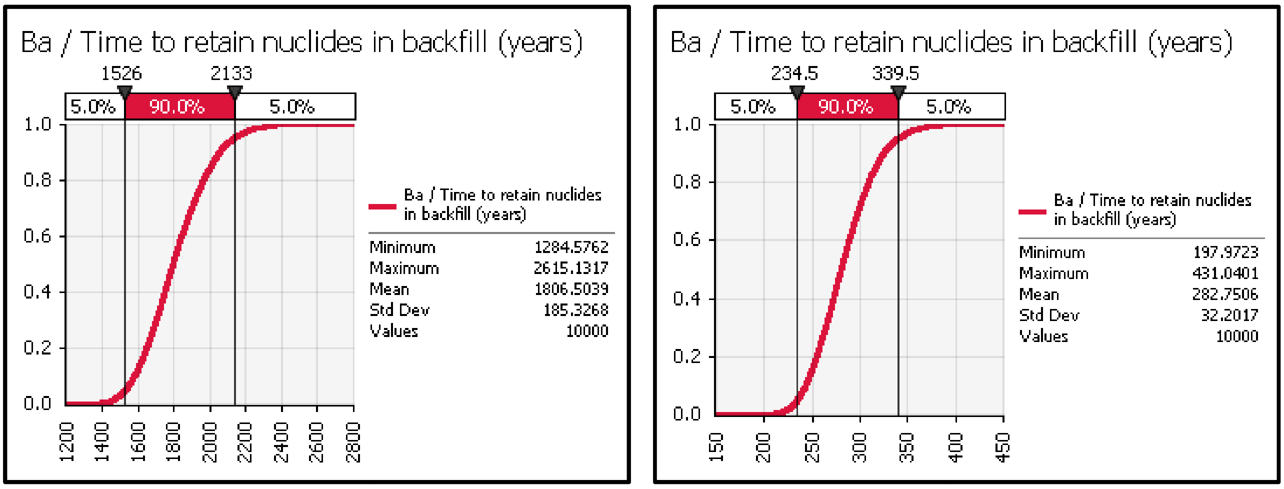


Figure D.13.19 Variations in retention time for Ba-133

Appendix E: Groundwater model, details

In this appendix, the specific details related to the groundwater modelling are described.

E.1 Assumptions that apply for the geosphere model

During this prefeasibility phase of the project the following assumptions are proposed:

- Steady flow conditions. The flow field does not change over time, which again results in short simulation times for the groundwater flow model.
- Constant head boundaries (upstream/downstream, river, wetland). The head level does not change over time which is necessary to obtain steady flow.
- Constant flux boundaries (groundwater wells). The abstraction from the groundwater wells does not change over time which is necessary to obtain steady flow.
- Isotropic conditions in the horizontal plane, i.e. hydraulic conductivity and porosity are the same in all directions in the horizontal plane.
- Stationary and isotropic transport conditions for each radionuclide e.g. exponential decay described by a half-life, adsorption described by the distribution coefficient K_D.
- Isotropic transport conditions in each geological formation, i.e. constant adsorption coefficient and constant dispersion conditions.

These assumptions may not always coincide with worst-case assumptions, but since the same conditions are considered for all models, it will not change the ranking of the models. It will also have minor impact compared to the parameter uncertainty and variability of hydrogeological and hydro geochemical parameters.

Furthermore, the geometrical setup is rigid, but will anyway take various issues into account. The assumptions that deter-mine the geometrical setup of the risk assessment model are listed and explained in Table E.13.7.

Table E.13.7 General assumptions that apply during the risk assessment calculations

Issue	Assumption	Comments
Thickness of *formation*	10 m	Each layer of the *formation*
Distance to deep groundwater well	1,000 m (down stream)	Since the location of the repository will respect areas with special groundwater interests, the distance to the nearest water supply well is relatively large
Distance to shallow groundwater well	1,500 m (down stream)	Since the location of the repository will respect areas with special groundwater interests, the distance to the nearest water supply well is relatively large
Depth of shallow groundwater well	20-30 m	This well will typically abstract water from a quaternary sand aquifer
Depth of deep groundwater well	60-70 m	This well will typically abstract water from a Danien limestone aquifer
Groundwater intake - shallow well	15,000 m^3/year	It is assumed that the nearest shallow water supply well will supply a smaller waterworks or is used for irrigation
Groundwater intake - deep groundwater well	200,000 m^3/year	It is assumed that the nearest deep water supply well will supply a larger waterworks
Distance to stream	1,000 m	
Gradient between repository location and stream	about 5 m	
Distance to final recipient	2,500 m	
Gradient between repository and final recipient	about 15 m	
Distance to wetland	1,300 m (down stream)	
Gradient between repository and wetland	about 10 m	
Net precipitation close to wetlands, lakes and rivers	-100 mm	Evaporation assumed to be higher than precipitation in these areas
Net precipitation in "other areas"	400 mm (varies between 200 and 600 in Denmark)	Typical average net precipitation value in Denmark (Henriksen and Sonnenborg, 2003)
Drainage	depth = 1 m	In "other areas"

The radionuclide source is given as a fixed (unit) flux having a fixed (unit) concentration. This means that the groundwater modelling results can be combined with the results from the repository model by scaling to the calculated fluxes and concentrations.

A simple retardation model to describe contaminant ad-sorption has been adopted, which is the empirical distribution coefficient K_D, that relates the equilibrium concentration of a species adsorbed on a given mineral to the concentration in solution. It is thus assumed that retardation is independent of the concentration levels groundwater chemistry, etc. which is in general not true (see e.g. Criscenti et al., 2006). However, the determination of K_D should be site specific (Criscenti et al., 2006), which is not possible in this pre-feasibility phase. In stead, K_D values have been chosen that are representative of the geological conditions in the proposed conceptual models and are as much is possible results of lab of field experiments with a mixture of nuclides. Where estimates are made, they are on the conservative side.

Since the repositories will be located in typically low permeable environments, the flow velocities will be low. As such the dispersion, which depends on the flow velocity, will be low as well. Molecular diffusion may therefore be an important process and will be included in the calculations. This was also concluded in the studies of transport in fractured clay by Harrar et al. (2007) and Huysmans and Dassargues (2005).

The timing of the source, i.e. when will dissolved matter be released to the groundwater system as well as the concentration of the source will be assessed in the repository model. The groundwater model will determine the dilution and retardation of species and determine concentrations at the points of interest as a function of time.

E.2 Parameterisation of the combinations

Parameters concerning hydrogeological, geochemical, hydraulic and geotechnical conditions will be of importance as part of the preliminary safety assessments. Hydraulic and hydrogeological characteristics of the geological formations to be used in the safety assessment are suggested below as low, mean and high values.

The numbers given in the table below are based on the fact that one of the selection criteria for GEUS, when proposing actual locations, is that the geologic formations should have a low permeability (Danish Decommissioning, 2009). Parameter values for hydraulic conductivities are mainly based on literature values in Freeze and Cherry (1979) combined with more specific literature references described below and experience. It should be emphasized that one of the main controlling factors for the hydraulic conductivity of the geologic formations listed in Table E.13.8 is the presence of fractures. This may cause preferential flow directions, where the hydraulic conductivities are in the high end of the given range. The mean values are estimated.

Table E.13.8 Important parameters for the geological repositories considered are given as low, mean and high values (see justification below)

		Clay, plastic, fat	*Clay till, etc.*	*Limestone*	*Rock*
Horizontal hydraulic conductivity (m/s)	Low	10^{-10}	10^{-9}	10^{-9}	10^{-12}
	Mean	10^{-8}	$5x10^{-7}$	10^{-7}	10^{-10}
	High	10^{-6}	$5x10^{-5}$	$5x10^{-6}$	10^{-8}
Vertical hydraulic conductivity (m/s)	Low	$5x10^{-11}$	10^{-9}	10^{-9}	10^{-12}
	Mean	$5x10^{-9}$	10^{-7}	10^{-7}	10^{-10}
	High	$5x10^{-7}$	$5x10^{-5}$	$5x10^{-6}$	10^{-8}
Porosity (-)	Low	0.15	0.10	0.20	0.01
	Mean	0.20	0.15	0.30	0.02
	High	0.30	0.20	0.40	0.04
Effective porosity (-)	Low	0.05	0.01	0.02	0.005
	Mean	0.10	0.02	0.04	0.01
	High	0.15	0.05	0.10	0.02

E.2.1 Clay till formations

Clay till formations are very common in the Danish underground, and several studies have focused on the determination of the bulk hydraulic conductivity of clay till. A number of studies have also investigated fractures and their influence on transport of water and solutes in clay till formations.

The Danish EPA (Miljøstyrelsen, 1998) suggests that clay vertical hydraulic conductivities are in the range of $2x10^{-7}$ m/s to $2x10^{-5}$ m/s in the upper 1 to 5 m of the soil column.

In Miljøstyrelsen (2007) a number of clay till samples from 8 different locations in Denmark at various depths down to 10 m below ground level are reported. It is stated that the visible number of macro pores decreases considerably with increasing depth. The hydraulic conductivity of the clay till is in the range of 10^{-5} m/s in the near surface zone decreasing to 10^{-6} m/s at 3-5 m depth and 10^{-9} m/s at 6-7 m depth. The samples are taken from vertical boreholes and the hydraulic values represent as such the vertical hydraulic conductivity.

In Harrar et al. (2003) a study area situated within a dissected glacial-till plain on the Jutland peninsula was subject to intensive geological and numerical modelling. Hydraulic parameters are determined by use of inverse modelling techniques applying different geological models. The vertical hydraulic conductivity of till clay is in this study determined to be $3.5x10^{-9}$ to $4.8x10^{-5}$ m/s within a 95 % confidence interval with an estimated mean of $4.1x10^{-7}$ m/s.

In Henriksen og Nyegaard (2003), which is based on Henriksen and Sonnenborg (2003) calibrated model parameters from a large scale Danish water resource model study are reported. Un-weathered clay till vertical hydraulic conductivities are reported in the range of $2x10^{-9}$ to $2x10^{-8}$ m/s. It should be noted that the modelling approach in Henriksen and Sonnenborg (2003) is that parameter values are kept constant throughout an entire model domain, e.g. for all clay layers on Funen – and local scale heterogeneities are not taken into account. These parameter values are as such comparable to mean values.

E.2.2 Plastic clay formations

The Danish EPA (Miljøstyrelsen, 1998) suggests that the horizontal hydraulic conductivity in deep clay formations varies from 10^{-8} to 10^{-2} m/s while the conductivity in surficial clay formations varies from 10^{-8} to 10^{-6} m/s. However, these figures do not refer specifically to plastic clay formations but to clay formations in general.

In Harrar et al. (2003) the vertical hydraulic conductivity of tertiary clay is determined to be $4.3x10^{-10}$ to $1.7x10^{-6}$ m/s within a 95% confidence interval with an estimated mean of $2.7x10^{-8}$ m/s.

In Henriksen og Nyegaard, (2003) plastic clay and silt vertical hydraulic conductivities are reported in the range of $3x10^{-9}$ to $1.5x10^{-7}$ m/s.

E.2.3 Limestone formations

The Danish EPA (Miljøstyrelsen, 2007) suggests that limestone horizontal hydraulic conductivities are in the range of 10^{-7} to 10^{-5} m/s. However, most of the measurements of the hydraulic conductivity of limestone formations have been carried out in high permeable and/or highly fractured formations in order find suitable aquifer conditions for water withdrawal. The parameters in the Table 3 3 represent less permeable parts of a limestone formation and as such represents parts where the repository may be located.

E.2.4 Rock formations

The Danish EPA (Miljøstyrelsen, 2007) suggests that horizontal hydraulic conductivities in weathered rock formations are in the range of 10^{-8} m/s to 10^{-4} m/s. It has not been possible to identify hydraulic parameters for deep, un-weathered Danish rock formations. However, other studies - in Sweden and elsewhere - have been considered. As for clay till the hydraulic conductivity of a rock formation decreases with increasing depth both because the number of fractures typically decreases, and because un-weathered rock in general has a lower hydraulic conductivity.

In a highly fractured granite formation in France, it is suggested in Cacas et al. (1990a) and Cacas et al. (1990b) that the effective hydraulic conductivity in a continuum model approach is in the order of $2x10^{-8}$ m/s.

In Zhao (1998) the hydraulic conductivity in the massive part of the Bukit Timah granite is measured to be in the range of 10^{-10} to 10^{-8} m/s and in Wilson et al. (1983) the large scale rock hydraulic conductivity at the experimental site Stripa, Sweden, was measured to be in the order of 10^{-10} m/s.

In Watkins (2003) the representative hydraulic conductivity of the Carnmenellis granite of Cornwall, UK, is estimated. It is suggested that the granite aquifer can be characterised by four vertically distributed zones: an uppermost extremely weathered intergranular flow zone of very high hydraulic conductivity up to 2-3 m thick; a high permeable upper zone, 30 to 100 m deep with hydraulic conductivity around 10^{-5} 10^{-6} m/s; a moderately permeable middle zone covering a few hundred meters in depth and exhibiting the hydraulic conductivity in the region of 10^{-7} to 10^{-9} m/s. Beneath this, the granite may be considered effectively impermeable, with hydraulic conductivity of around 10^{-9} m/s - 10^{-10} m/s, though water flow at depth can still be considerable within major zones of discontinuities, providing local zones with hydraulic conductivity of 10^{-5} m/s or more.

E.3 Retardation

The species to be assessed in the preliminary assessment will be retarded due various geochemical reactions. Retardation is typically described conceptually by a retardation factor that can be determined from the distribution coefficient K_D and the bulk density of the porous media.

Retention of the nuclides in the different geologies has a major impact on the resulting concentrations at the different recipients considered. A literature study has been carried out of studies from a number of existing and planned repositories on nuclide and matrix related K_D-values. These values vary substantially with both nuclide and soil/rock type, and based on this literature study, K_D-values given in Table E.13.9 have been used in the calculations.

As can be seen from the table, different K_D-values have been used dependent on nearness to the soil surface. This is due to connection between K_D and the redox conditions. Values based on aerobic conditions have been used as a basis for the suggestions for the upper layers, while anaerobic conditions are assumed for the deeper layers. Generally, values relating to non-saline conditions and a pH above 8 have been used as a basis, since some form of conditioning creating relatively high pH is assumed together with the presence of concrete in the construction itself.

Table E.13.9 Suggested K_D-values, m^3/kg

	Above Surface and Near Surface			Medium deep and Borehole		
	Rock	Limestone	Soil*	Rock	Limestone	Soil*
H	0	0	0	0	0	0
C	0.001	0.001	0.001	0.001	0.001	0.001
Ca	10	10	10	1.000	1.000	1.000
Co	0.001	1	0.1	0.01	10	1
Ni	0.1	1	0.1	1	1	1
Se	0	0.01	0.01	0.001	0.1	0.1
Sr	0.001	0.1	0.01	0.01	1	0.01
Tc	0.1	0.1	0.001	1	1	0.01
Ag	0.1	0.1	0.1	1	1	1
Ba	0.001	0.001	0.001	0.001	0.001	0.001
Cs	0.05	0.1	0.1	0.05	1	1
Sm	0.1	0.1	0.1	1	1	1
Eu	0.1	0.1	10	1	0.1	10
Ir	0.001	1	0.1	0.01	10	1
Pb	1	1	1	1	1	1
Ra	0.1	0.1	0.1	1	1	1
Ac	0.02	0.1	0.1	0.2	0.1	1
Th	1	1	1	1	1	1
Pa	0.1	0.1	1	0.1	0.1	1
U	0.002	0.02	0.2	0.2	0.2	2
Np	0.02	0.02	0.2	0.2	0.2	0.2
Pu	0.2	0.2	0.2	2	2	2
Am	0.2	0.2	0.2	2	2	2
Cm	0.2	0.2	0.2	2	2	2

* Soil refers to sand, clay and clay till

The distribution coefficients have been grouped as a basis for the groundwater modelling. The groups are listed in Table E.13.10.

Table E.13.10 K_D groups applied in the solute transport model

Group #	K_D (m^3/kg) min-max	K_D (m^3/kg) best estimate
1	100-1,000	200
2	10-100	20
3	1-10	2
4	0.1-1	0.2
5	0.01-0.1	0.02
6	0.001-0.01	0.002
7	0	0

In the safety assessment modelling, the K_D values for soil have been adopted for clay, sand and clay till while the K_D values for rock have been adopted for limestone and rock. The following 14 combinations have been modelled (K_D(rock)/K_D(soil)):

7/7, 6/6, 6/4, 6/3, 5/5, 4/5, 4/4, 3/5, 2/2, 4/3, 3/4, 3/3, 3/1 and 2/3.

The calculation of dose in the biosphere model is dependent on both the nuclide in question and the geosphere scenario related to the placement of the repository. To combine the dose calculation for each nuclide with the relevant geosphere scenario and K_D combination the matrix shown in Table E.13.11 and Table E.13.12 has been set up.

As can be seen from Table E.13.11 and Table E.13.12, all nuclides are represented in all geosphere scenarios (as they should), but they may have different K_D's dependent on the geology and thus be placed in a different combination group. This will be important for their resulting concentration at the points of interest and thus for the overall dose. It is also seen that based on the final K_D's suggested, combination 9 turned out to be irrelevant.

Table E.13.11 Combinations of nuclides and KD scenarios

	Near surface					Medium deep			Borehole	
Geological formation	Rock	Rock/soil	Limestone	Limes-tone/soil	Soil	Rock/soil	Lime-stone/soil	Soil	Rock/soil	Lime-stone/Soil
Geosphere scenarios K_D group (rock | soil)	G1M1	G4M4	SK1M1	No combi-nation	L1M2 & L1M4 & ML1M1 & ML4M1 & ML4M3	G3M6	SK2M5 & SK4M7 & SK7M7	L1M6 &L5M6 &L5M8 & L5M10 & ML5M5	G4M11 & G4M14	SK8M11 & SK8M14
7 | 7	H, Se	H	H	H	H	H	H	H	H	H
6 | 6	C, Co, Sr, Ir, U, Ba	C, Se, Sr, Ba	C, U, Ba	C, Ba	C, Tc, Ba, Np	C, Ba	C, Ba	C, Ba	C, Ba	C, Ba
6 | 4		Tc, Ac, U, Np		Tc, U, Np		Co, Cs, Ir			Co, Cs, Ir	
6 | 3	Ni, Tc, Ag, Cs, Sm, Eu, Ra, Ac, Pa, Pu, Am, Cm, Np	Co, Ni, Ag, Cs,Ir,	Se, Sr, Tc, Ag, Cs, Sm, Eu, Ra, Ac, Pa, Pu, Am, Cm, Np	Se, Sr, Ag, Cs	Co, Ni, Se, Sr, Ag, Sm, Ir, U,	Se, Sr, U, Np	Se, Sr, U, Np	Se, Sr, Tc, Ag, U, Np	Se, Sr, U, Np	Se, Sr, Tc, Ag, U, Np

Table E.13.12 Combinations of nuclides and K_D scenarios (continued)

	Near surface					Medium deep			Borehole	
Combination no.	Rock	Rock/soil	Limestone	Lime-stone/soil	Soil	Rock/soil	Lime-stone/soil	Soil	Rock/soil	Lime-stone/Soil
Geosphere scenarios K_D groups (rock | soil)	G1M1	G4M4	SK1M1	No combi-nation	L1M2 & L1M4 & ML1M1 & ML4M1 & ML4M3	G3M6	SK2M5 & SK4M7 & SK7M7	L1M6 &L5M6 &L5M8 & L5M10 & ML5M5	G4M11 & G4M14	SK8M11 & SK8M14
5 | 5					Eu, Pu, Am, Cm					
4 | 5		Eu, Sm, Ra, Pu, Am, Cm		Eu, Sm, Ra, Ac, Pu, Am, Cm		Ac,	Ac		Ac,	Ac
4 | 4		Pa		Co, Ni, Ir, Pa		Tc, Pa	Tc, Pa		Tc, Pa	Tc, Pa
3 | 5	Pb, Th	Pb, Th	Co, Ni, Ir, Pb, Th	Pb, Th	Pb, Th, Pa, Cs, Ra, Ac	Ni, Ag, Sm, Eu, Pb, Ra, Th, Pu, Am, Cm	Ni, Ag, Cs, Sm, Pb, Ra, Th, Pu, Am, Cm	Co, Ni,Sm, Eu, Ir, Pb, Ra, Th, Pa, Pu, Am, Cm, Cs, Ac	Ni, Ag, Cs, Sm, Eu, Pb, Ra, Th, Pu, Am, Cm	Ni, Ag, Cs, Sm, Pb, Ra, Th, Pu, Am, Cm
2 | 2										
4 | 3	Ca	Ca	Ca	Ca	Ca	Ca	Co, Eu, Ir, Ca	Ca	Ca	Co, Eu, Ir, Ca

E.4 Diffusion

A number of studies have dealt with diffusion of various species in aqueous solutions and in rock and fill environments. The tables below give an overview of these findings.

Table E.13.13 Summary of diffusion parameters reported in literature (Pohlmann, et al, 2007)

Radionuclide	D_e, m^2/s	Method	Reference
Sr	1×10^{-13} - 7×10^{-11}	Core section	Yamaguchi et al. (1993)
Cs, Sr	1×10^{-12} - 3.7×10^{-11}	Crushed granite	Skagius et al. (1982)
U	1.5×10^{-13} - $1.5 X 10^{-12}$	Field experiment	Birgersson and Neretnieks (1990)

Table E.13.14 Recommended Rock Matrix Diffusion Coefficients of Radionuclides (Baik, et al, 2008)

Radionuclide	Molecular diffusion coefficient in water, D_x, m^2/s	Effective diffusion coefficient in rock matrix, D_e, m^2/s	Apparent diffusion coefficient in rock matrix, D_a, m^2/s
Cs	2.1×10^{-9}	8.8×10^{-14}	6×10^{-16}
Sr	7.9×10^{-10}	$3.3 - 27 \times 10^{-14}$	6×10^{-14}
U	$4.3 - 10 \times 10^{-10}$	3.6×10^{-14}	3×10^{-18}
Th	1.5×10^{-10}	6.3×10^{-15}	5×10^{-19}
Am	1×10^{-9}	4×10^{-14}	5×10^{-18}
Pu	1×10^{-9}	4×10^{-14}	3×10^{-18}
Pa	1×10^{-9}	4×10^{-14}	1×10^{-17}
Ac	1×10^{-9}	4×10^{-14}	5×10^{-18}
Cm	1×10^{-9}	4×10^{-14}	5×10^{-18}
Ra	8.9×10^{-10}	3.7×10^{-14}	7×10^{-16}
Np	1×10^{-9}	$4 - 25 \times 10^{-14}$	3×10^{-18}
C	1.2×10^{-9}	5×10^{-14}	2×10^{-14}
Ni	6.8×10^{-10}	2.8×10^{-14}	5×10^{-16}
Se	1×10^{-9}	4×10^{-14}	1×10^{-14}
Tc(IV)	1×10^{-9}	4×10^{-14}	1×10^{-17}
$Tc(VII)O_4^-$	1×10^{-9}	4×10^{-14}	8×10^{-12}
Sm	1×10^{-9}	4×10^{-14}	7×10^{-18}

Table E.13.15 Matrix porosity and effective diffusion coefficients in the rock matrix with different groundwater types (Vieno and Nordman, 1999). Se is assumed to be present as anion in all cases. Tc, Sm, U, Np and Pu are only anions in glacial water

Parameter	Distance from fracture, cm	Species	Brackish/dilute and glacial	Saline
Porosity, ε, %	0 - 1	Anions	0.1	0.2
		Neutral and cations	0.5	0.5
	1 - 10	Anions	0.02	0.04
		Neutral and cations	0.1	0.1
Effective diffusion coefficient, D_e, m^2/s	0 - 1	Anions	10^{-14}	5×10^{-14}
		Neutral and cations	10^{-13}	10^{-13}
	1 - 10	Anions	10^{-15}	5×10^{-15}
		Neutral and cations	10^{-14}	10^{-14}

Table E.13.16 Diffusion in fill materials (Nykyri, et al., 2009). Se is assumed to behave as anion in all cases, and Sm, Am and Cm only in glacial water cases. Grain density of fill: 2700 kg/m^3

Speciation	Neutral	Anion	Cation
Bentonite buffer			
Porosity, ε	0.43	0.17	0.43
Effective diffusion coefficient, De, m^2s	1.2×10^{-12}	1×10^{-11}	3×10^{-10}
Bentonite/ballast backfill			
Porosity, ε	0.23	0.092	0.23
Effective diffusion coefficient, D_e, m^2/s	5×10^{-11}	4.2×10^{-12}	1.3×10^{-10}

It has not been possible to find references describing the diffusion coefficient for all species included in the risk assessment. However, it seems that there is relatively little variability in the diffusion coefficient. Based on these findings the transport models have applied a diffusion coefficient (in rock and limestone formations) of 1.5×10^{-10} m^2/s in all simulations, which is considered a conservative estimate.

Table E.13.17 Diffusion in fill materials (Nykyri, et al., 2009). Se is assumed to behave as anion in all cases, and Sm, Am and Cm only in glacial water cases. Grain density of fill: 2700 kg/m3

Radionuclide	Bentonite type	Effective diffusion coefficient D_e, m^2/s	Apparent diffusion coefficient D_a, m^2/s
Tc	Ca-bentonite	-	9.5×10^{-11}
Ni	Ca-bentonite	-	2.0×10^{-13}
Am	Ca-bentonite	-	7.3×10^{-15}
U	Ca-bentonite	-	1.8×10^{-13}
Co	Ca-bentonite	-	3.4×10^{-13}
C	Ca-bentonite	-	7.6×10^{-11}
Sr	Na-bentonite	2.4×10^{-11}	8.5×10^{-12}
	Ca-bentonite	1.9×10^{-11}	4.1×10^{-12}
Cs	Na-bentonite	7.0×10^{-12}	5.7×10^{-13}
	Ca-bentonite	1.1×10^{-11}	1.4×10^{-12}

E.5 Model code inventory and selection

Model codes for continuum approaches and discrete fracture network model codes are described and compared. Based on a model inventory, it is suggested to apply a commonly used groundwater model code, namely the MODFLOW GMS package, to simulate flow and transport processes. It includes the MODFLOW flow model, and a multispecies reactive transport code called MT3DMS, which was suggested as the transport code. The model code includes a number of simulation packages; groundwater flow will be simulated using as a continuum approach covering the upper 300 m of the groundwater system.

Model codes for continuum approaches assumes that flow and transport processes are described using local averaged hydraulic parameter such as transmissivity and porosity. In discrete fracture network model codes, the flow and transport in fractures are described as the transport in connected tubes of different width, length and connectivity.

Apparently, there is a trend that safety assessments at underground nuclear waste repositories are carried out using a discrete fracture networks (DFN) or hybrid approach in the very near field of the repository, and a (stochastic) continuum or equivalent porous medium (EPM) approach at larger scales. This is both the case in investigations at Swedish locations of deep storage in rock formations (SKB, 2006a), in Canadian locations and at the Sellafield site, UK (Jeong and Song, 2002).

Results in Harrar et al. (2007) also indicate that vertical transport of solutes in oxidized and reduced zones of the till can be adequately simulated using an equivalent porous media under Danish conditions.

The choice for the prefeasibility study is supported by various scientists and studies; in Cacas et al. (1990a) and Cacas et al (1990b) it is stated that it is not realistic to interpret tracer experiments in the fractured crystalline rocks at Fanay-Augères, France, without accounting for discrete channels in the model. On the other hand, it also stated that the work confirms the suitability of stochastic continuum modelling for analysis of flow and transport at the site on scales larger than 10 m. Also Wilson et al. (1983), which report flow test results from a Swedish rock environment, indicate that test results have a behaviour approximating radial flow in a porous medium.

In Huysmans and Dassargues (2005), the Boom Clay, which is considered host geology for a radioactive repository in Belgium, is investigated. The purpose of that study was to investigate the effect of geological heterogeneity and fractures on transport through the clay itself by comparing stochastic modelling using a fracture model approach and a continuum modelling approach. On a very small scale, i.e. a 20 m long section, it was concluded that the difference between the fluxes of the heterogeneous simulations and the homogeneous model is also rather small. The output fluxes of the heterogeneous model differ at most 8 % from the fluxes of the homogeneous model.

E.6 Results of the groundwater modelling

It is obviously not possible (or relevant) to present all results from the groundwater modelling. In the following selected results will be presented which highlight some of the differences and similarities between the models.

The near surface location of the repository is compared for different depths but in the same geological formation, see Figure E.1. The geological formation is clay till and the depths are 0 - 10 metres, 0 - 10 metres and 20 - 30 metres below ground surface, respectively. The K_D group is 3/3 and the models have all run for 1,000,000 years.

From Figure E.1 it can be seen:

- In one of the models (ML1M1) the shallow well and the beach recipient are not present and therefore not represented in the graphs.
- Break through curves are a little more steep in the model without the shallow sand aquifer present (ML1M1). This is probably due to the fact, that transport of solutes primarily takes place in only one layer, namely the Danien limestone.

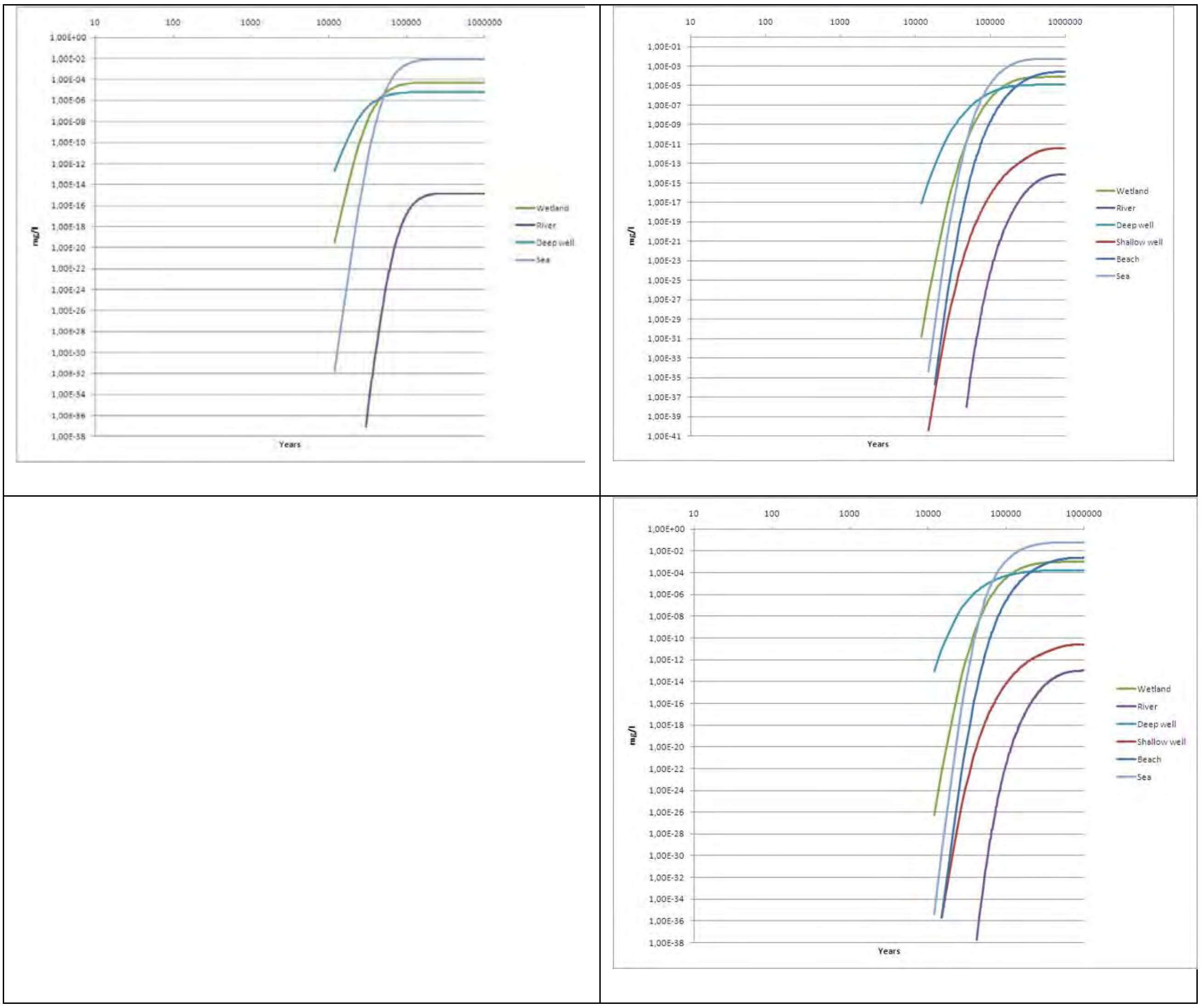


Figure E.1 Break through curves of concentration versus time for near surface location of the repository. Upper left: ML1M1, upper right: ML4M1 and lower right: ML4M3

- The break through curves from the ML4M1 and ML4M3 are very similar though the concentration levels in ML4M3 are slightly higher than in ML4M1. This is due to the fact that the repository in ML4M3 is located closer to the high permeable Danien limestone layer which is the layer the deep well is pumping from.

The medium deep location of the repository is compared for different depths but in the same geological formation, see Figure E.2. The geological formation is Maastrichtien limestone and the depths are 40-50 metres, 60-70 metres and 60-70 metres below ground surface, respectively. The K_D group is 3/3 and two of the models have run for 1,000,000 years while one of them has only run for 100,000 years.

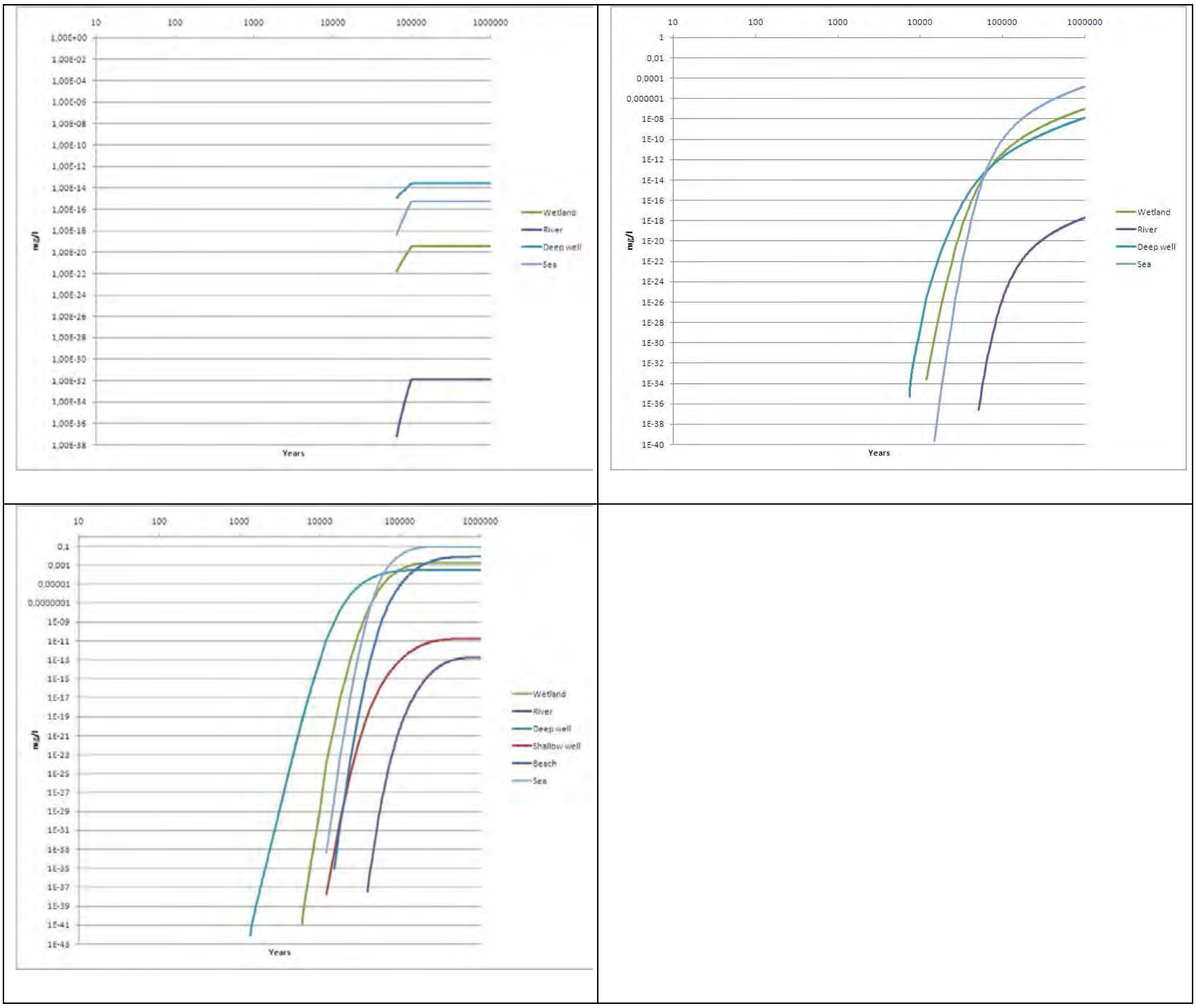


Figure E.2 Break through curves of concentration versus time for medium deep location of the repository in limestone. Upper left: SK2M5, upper right: SK4M7 and lower left: SK7M7.

From Figure E.2 the following can be seen:

- It is obvious that the break through in SK2M5 is somewhat later than the two other models and that the model should have run for more than 100,000 years in order to reach steady state;

- The break through is also considerably earlier in SK7M7 than in SK4M7 even though the repositories are located at the same depth. This is due to the fact that the repository in SK7M7 is located very close to the high permeable Danien limestone layer.

The borehole solution is compared for different depths and different geological formations, see Figure E.3.

The geological formations are Maastrichtien limestone and rock and the depths are 100-150 metres, 250-300 metres and 100-150 metres below ground surface, respectively. The K_D group is 3/3 and all models have run for 1,000,000.

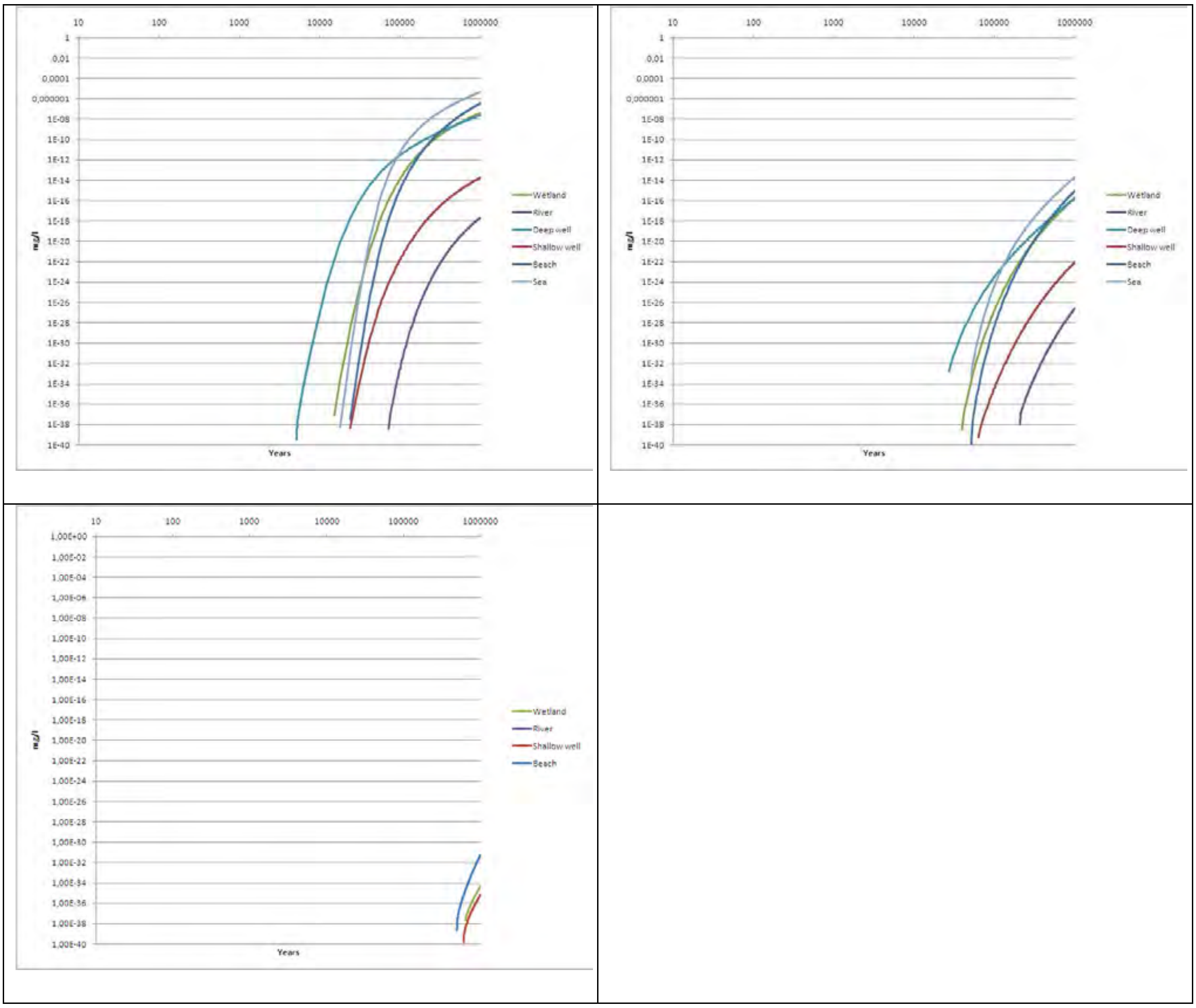


Figure E.3 Break through curves of concentration versus time for borehole solution. Upper left: SK8M11, upper right: SK8M14 and lower left: G4M11

From Figure E.3 the following can be seen:

- By comparing SK8M11 and SK8M14 it is obvious that the deeper the borehole is the later the break through happens and the lower the concentration becomes (delay in the range of 20.000 years and about 8 decades lower concentrations);
- The break through is considerably later in the rock formation than in limestone formations.

The medium deep repository location is compared for all the geological formations, see Figure E.4. The depths are 100-150 metres, 250-300 metres and 100-150 metres below ground surface, respectively. The K_D group is 3/3 and all models have run for 1.000.000 years.

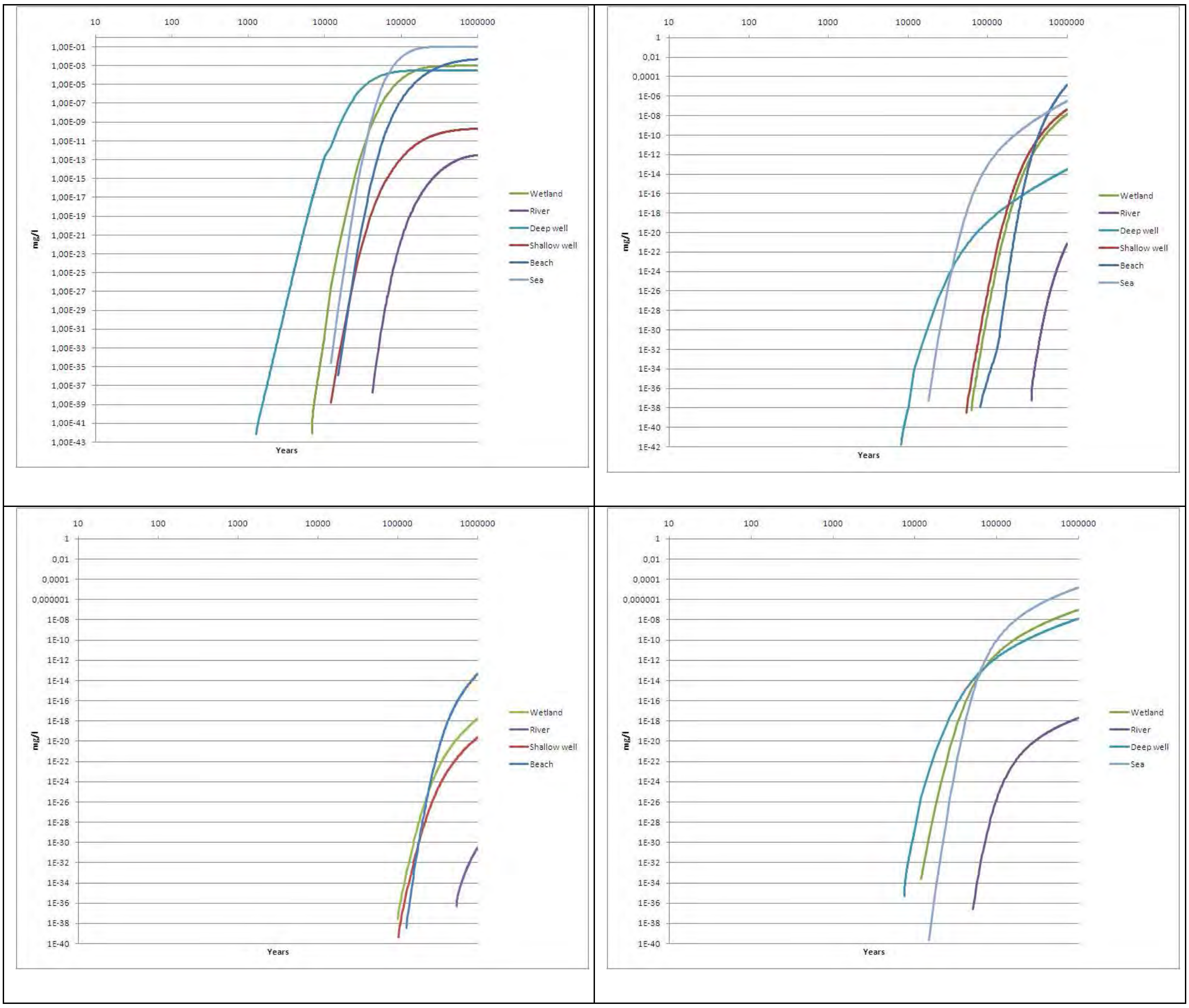


Figure E.4 Break through curves of concentration versus time for medium deep location of the repository in different geologies. Upper left: ML5M5, upper right: L5M6, lower left: G3M6 and lower right: SK4M7

From Figure E.4 the following can be seen:

- There are considerable differences between the break through time and the concentration levels. Lowest concentrations and latest break through appears in the rock formation while earliest break through and highest concentration levels appears in clay till formations;
- Break through is a little later in fat clay than in limestone formations and concentration levels are smaller;

- Steady state with respect to concentration levels is only reached for the clay till location of the repository within the 1,000,000 years simulation time.

Repositories located in fat clay at different depths are compared, see Figure E.5. The depths are 30 - 40 metres, 50 - 60, metres 50 - 60 metres and 70 - 80 metres below ground surface, respectively. The K_D group is 3/3 and all models have run for 1,000,000 years.

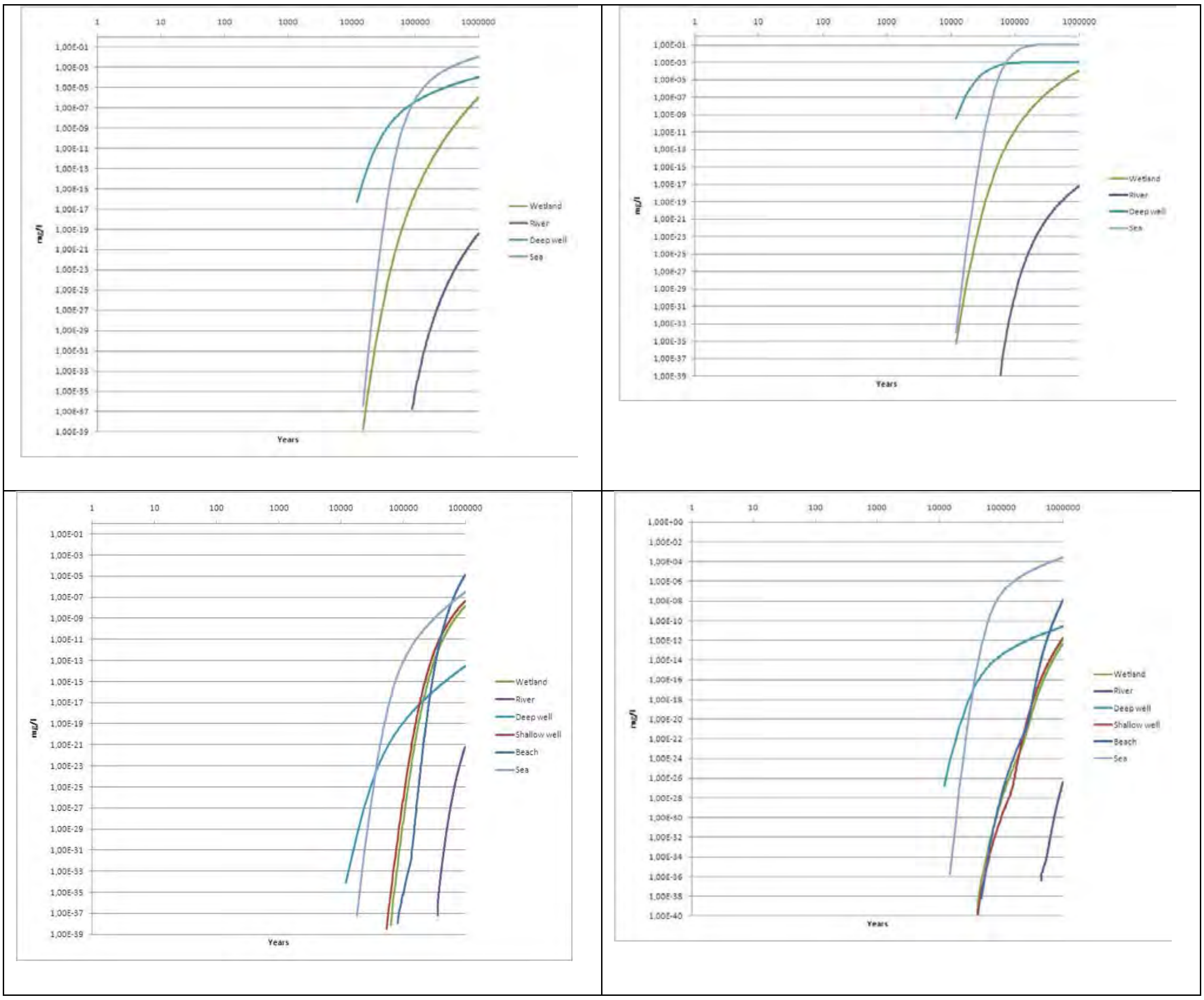


Figure E.5 Break through curves of concentration versus time for a repository located in fat clay at different depths. Upper left: L1M4 (near surface), upper right: L1M6 (medium deep), lower left: L5M6 (medium deep) and lower right: L5M8 (medium deep)

From Figure E.5 the following can be seen:

- There are some differences between the break through time and the concentration levels. Lowest maximum concentration and latest break through appears in the medium deep location, L5M5, and highest maximum concentration appears in the near surface location. This is due to the location of the high permeable limestone layer related to the repository;

- The shape and ranking of the break through curves related to different recipients varies considerably. This is also due to the location of layers around the fat clay layer.

E.7 Parameter variability

Two models were selected to analyses the effect of parameter variability, namely a near surface repository location (L1M4 - fat clay, repository at 30-40 metres below surface) and a medium deep repository location (SK4M7 - limestone, repository at 60-70 metres below surface). The K_D group is 12.

Twenty simulations have been carried out for each repository location changing the hydraulic conductivities of the repository formation uniformly between high and low values defined in Table E.13.8. All other parameters are kept constant. The maximum concentration after 10.000 years at the deep well is extracted from the results and plotted against the horizontal hydraulic conductivity.

The resulting graph from the near surface repository in L1M4 is shown in Figure E.6. As can be seen from the figure, the maximum concentration varies almost four decades as a result of changing the hydraulic conductivity about four decades.

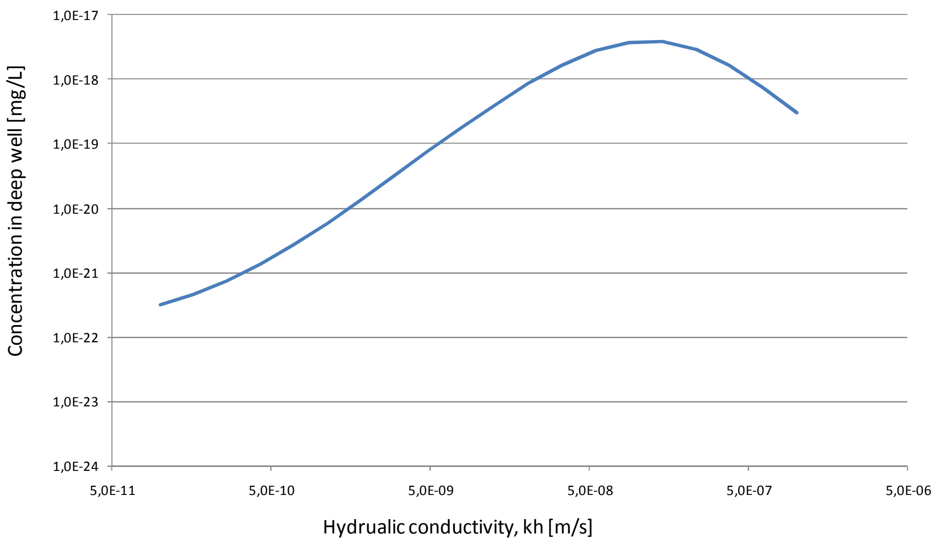


Figure E.6 The maximum concentration in the deep well as a function of the hydraulic conductivity for the near surface repository location

The resulting graph from the medium deep repository in SK4M7 is shown in Figure E.7. As can be seen from the figure the maximum concentration varies almost eight decades as a result of changing the hydraulic conductivity less than four decades.

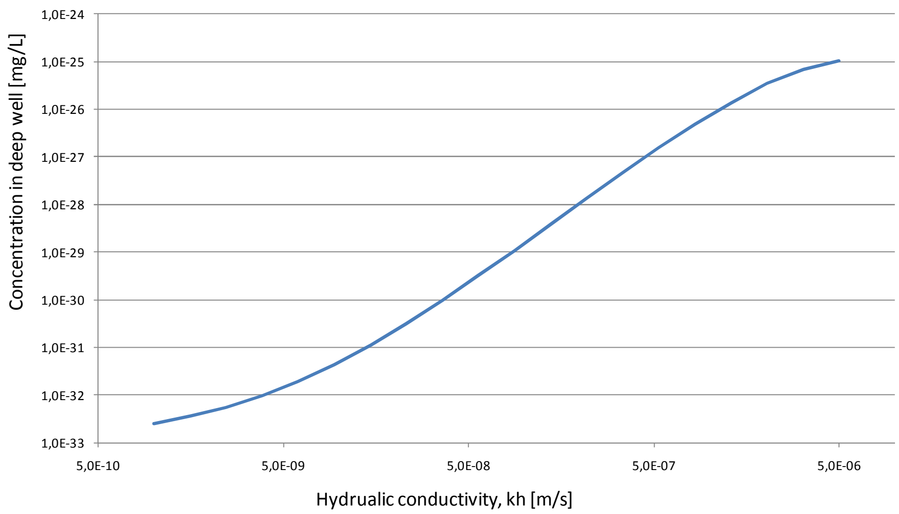


Figure E.7 The maximum concentration in the deep well as a function of the hydraulic conductivity for the medium deep repository location

It is noted that only the repository hydraulic conductivities are varied in the above simulations. All other parameters are subject to variability and thus the results will vary even more than indicated in the above figures. However, our conclusions from the above results are:

- In a prefeasibility phase with hypothetical geological and other settings it is not meaningful to include parameter variability in the safety analyses since it is very high and will result in unrealistic large concentration intervals;

- On the other hand it is very important to include parameter variability in the feasibility analyses to be carried out at a later stage;

- It is also very important to decrease the parameter variability through intensive monitoring campaigns in the selected areas for the repository to narrow down the result variability;

- It should also be emphasized that the conditions in the geological formation where the repository is located are important, but conditions in adjacent and surrounding geological formations are also very important to map.

Appendix F: Biosphere model, details

In this appendix, the specific equations for the different exposure pathways are listed. The method for dose calculation follows the principles recommended by the IAEA for radiological impact assessment (IAEA, 1994). This gives the following general equations for dose via ingestion, inhalation and general dose:

Ingestion:

$$D_{ing} = HC_i \times U_i \times DC_{ing},$$

Inhalation:

$$D_{inh} = C_a \times IH \times DC_{inh},$$

External dose:

$$D_{ext} = C_s \times \rho \times H_i \times DC_{ext},$$

where

DC_{ing}, DC_{inh} and DC_{ext} are dose coefficients for ingestion [Sv/Bq], inhalation [Sv/Bq], and external dose [(Sv/h)/(Bq/m3)] respectively according to ICRP recommendationsand Avila & Bergström (2006)[86].

HC_i = Consumption rate for pathway i [kg or litre per year], see Table 7.2.

U_i = Concentration in foodstuff i [Bq per kg or litre]

C_a = Concentration of radionuclides in air [Bq/m3]

IH = Inhalation rate [m3/h]

H_i = Exposure time [h/year]

C_s = Concentration of radionuclides in soil [Bq/kg], see F.1.4

ρ = Soil density [kg/m3]

[86] External dose coefficients are originally calculated in [(Sv/h)/(Bq/kg)] and then converted to [(Sv/h)/(Bq/m3)].

The modelling of pathways as described in the following is based on Karlsson et al (2001), Bergström et al (1999) and Avila & Bergström (2006), which have been used in the preliminary and final safety assessments carried out for the Swedish repositories for nuclear waste[87]. The equations used are thus based on assumptions and principles commonly used in risk assessment also of non-radioactive substances. Where special consideration has to be taken to the decay properties of the radionuclides, this has been incorporated.

The values of the dose coefficients for external exposure from the ground depend, among other things on the radionuclide vertical distribution in the ground. Hence, concentrations of radionuclides in soil that are consistent with the chosen DC_{ext} should be used. In practice, a homogeneous radionuclide distribution in a soil layer of infinite depth is assumed with concentrations representative of the most contaminated soil layer, which is conservative[88]. For radionuclides with decay chains, the values include the contribution from short-lived daughter radionuclides, assuming equilibrium.

F.1 Ingestion

Intake through ingestion encompasses ingestion of water, milk and meat, crops and of fish and crustaceans.

Concentrations in drinking water are direct results of the geosphere model, while the other concentrations require specific supplementary calculations.

F.1.1 Concentrations in milk and meat

Concentrations of radionuclides in milk and meat can be calculated from the following equations[89]:

Milk:

$U_m = In \times F_m.$

Meat:

$U_f = In \times F_f,$

where

[87] Information on validity of the model and the choice of data can be found in (Bergström, et al, 1995), (Karlsson, et al, 2001) and (Avila & Bergström, 2006).

[88] The actual values for a layer of 15 cm and infinite depth are almost equal. The values are based on a silty soil with 20 % air and 30 % water content, which is deemed a reasonably conservative estimate for use in the prefeasibility study. In the final safety assessments, specific values for the actual locations should be used.

[89] Excretion of nuclides from the cow before consumption is not taken into account for reasons of simplicity, neither is possible effects of food processing and storage time.

In = The cow's daily intake of nuclides [Bq/day]

F = Element specific distribution coefficient for milk and meat, respectively [day/litre, day/kg].

The cow's fodder is assumed to consist of pasturage and cereals. Cereals represent concentrated food. Additionally, some inadvertent consumption of soil when grazing is expected. The fodder is potentially contaminated through root uptake and retention of radionuclides on vegetation surfaces, if the crop is irrigated[90].

The cow's daily intake of radionuclides (In) can then be calculated as:

$$In = MC_w \times UC_w + MC_p \times UC_p + MC_c \times UC_c + MC_s \times UC_s,$$

where

w stands for water, p for pasturage, c for cereal and s for soil

MC_i = The daily consumption [kg or litre per day], see Table 7.3.

UC_i = The concentration of radionuclides in foodstuff, water and soil, respectively, eaten by the cow [Bq per kg or litre]

The concentrations of radionuclides in water are directly taken from the geosphere model, while the concentrations in soil are calculated as shown in F.1.4.

The concentration in cereals that are used as concentrate to cattle is assumed to have the same concentration as in cereals for human consumption, see F.1.2. Pasturage is not assumed to be irrigated, so the concentration of radionuclides in pasturage can be calculated as:

$$UC_p = C_s \times B_p$$

where

C_s = Concentration of radionuclides in soil [Bq/kg], see F.1.4
B_p= Nuclide specific soil to pasturage root uptake factors [(Bq/kg)/Bq/kg)]

F.1.2 Concentration in crops

Crops are represented by cereals, root crops and green vegetables. Grain and root crops are contaminated from root uptake, retention on vegetation surfaces and translocation. The resulting concentration in cereals and root-crop products can be calculated as

$$U_i = C_s \times B_i + \sum_0^{N_{IRR}} I_n \times TL \times C_w,$$

---

[90] Only assumed for cereals

where

C_s = Concentration of radionuclides in soil [Bq/kg], see F.1.4

B_i = Nuclide specific soil to plant transfer factors [(Bq/kg)/Bq/kg)]

n = Number of irrigation occasions, see Table F.1.

I_n = Remaining water on the vegetation after each irrigation occasion [m^3/m^2], see Table

TL = Nuclide specific translocation from surface to edible parts of plant, [(Bq/kg)/(Bq/m^2)],

C_w = Concentration of radionuclides in irrigation water [Bq/m^3]

The nuclide specific transfer and translocation factors from Karlsson et al (2001) are suggested used with relevant updates taken from IAEA (2009). In principle, the transfer coefficients will also depend on the physicochemical form of the radionuclide, but data on the variation among physicochemical forms are not available[91]. Furthermore, they will depend on soil type and on plant type. In this study, it is assumed that plants are grown in a loam, which will be the typical situation in Denmark[92]. The variation between soil types is taken into account in the modelling of transport with the water flow from the repository (in the saturated zone), where substantial difference will exist between the scenarios. The actual variation in transfer coefficients between soil types are not great compared to other variations handled in the calculations, see IAEA (2009). As for all parameters used in the biosphere modelling variation intervals and distribution functions have been assumed for the calculation of variability of the dose results due to potential parameter variability.

Variations between plant types can be greater, especially for plants with phyto-extractive properties. Values for plant types representing the different crop types are chosen. Evaluation of the variation and uncertainty of these parameters is carried out as part of the variability analysis.

Vegetables are contaminated from root uptake and surface contamination due to retention of contaminated irrigation water. The contamination of the soil in the case of long-term exposure from leaching to groundwater of radionuclides from the repository is through movement of dissolved species. This is taken into account when choosing the transfer factors. Contamination through settling of

91 The estimated released concentrations of the radionuclides from the repository are based on a literature study of solubility evaluations from evaluations of solubility limits of mixtures either from laboratory work or geochemical modeling. This is described further in the chapters regarding the repository model.

92 In the prefeasibility study this assumption is necessary due to lack of detailed knowledge. This could be verified in a further safety assessment for the specific locations.

dust is not taken into account, since this makes up a marginal portion of the exposure in this case.[93]

Retained irrigation water with its contents of radionuclides contaminates the surfaces of the vegetation. The concentration of radionuclides on vegetation surfaces decreases due to growth and effects of wind and precipitation, the effect of which is described by the weathering half-life (IAEA, 2009). A new irrigation occasion causes an additional retention, while there is an exponential decrease during the time passing between the irrigation occasions. The amounts of radionuclides on the surfaces of vegetation are therefore a function of frequencies of irrigation and time for harvest.

The harvest of green vegetables is assumed to occur during the entire growing period. Therefore, the mean concentration of surface contamination is calculated. The resulting content of radionuclides in vegetables can be calculated as:

$$U_v = C_s \times B_v + \frac{C_w}{Y_p} \times \frac{I}{t_{tot}} \times \sum_{N_{IRR}} \int_0^{t_n} e^{-\tau t}\, dt,$$

where

C_s = Concentration of radionuclides in soil [Bq/kg], see F.1.4.

B_V = Nuclide specific soil to plant transfer factor [(Bq/kg)/Bq/kg)]

C_W = Concentration of radionuclides in irrigation water [Bq/l][94]

Y_V = Yield of vegetables [kg dw/m^2]

I = Remaining water on the vegetation after each irrigation occasion [m^3/m^2],

t = Time between irrigation occasion and end of irrigation period [days]

t_n = Length of each irrigation period [days]

t_{tot} = Irrigation period [days]

$\tau = ln2/T_{½w}$ where $T_{½w}$ = weathering constant [day^{-1}]

N_{IRR} = Number of irrigation occasions.

The suggested parameters used are given in Table F.1 (and taken from Karlsson et al, 2001). The nuclide-specific soil to pasturage transfer factors are also taken from Karlsson et al (2001) with relevant updates taken from IAEA (2009).

93 This may be different in for exposure due to some of the accidents, which will be described further in Chapter 8 of the main report.

94 Is the concentration in the water from the short well calculated as described in Working Report 5

Table F.1 Parameters used in calculation of radionuclide concentrations in milk, meat and crops

Parameter	Unit	Value	Reference
I	m^3/m^2	0.003	Persson, 1997*
t_{tot}	days	90	Karlsson el al, 2001*
$T_{½w}$	days	20[95]	IAEA, 2009
N_{IRR}	-	5	Karlsson et al, 2001*
Y_p	kg dw/m2	0.5	Haak. 1983*

* these data will also be relevant for Danish conditions

F.1.3 Concentrations in fish and crustacean

The concentrations of radionuclides in those groups of species are obtained by use of bioaccumulation factors for edible parts of the species relative to total concentration of respective nuclide in water and suspended matter. The bioaccumulation factors are valid for steady-state conditions and implicitly consider all paths from the surrounding environment. The concentrations can be calculated as:

$$U_i = B_i \times C_w,$$

Where

B_i= Radionuclide specific concentration factors water to edible part of the species [(Bq/kg)/(Bq/l)] for fish and crustacean, respectively. Values from Karlsson al (2001) are used.

C_w= Concentration of soluble and suspended matter in surface water (either the stream or the ocean) [Bq/l][96].

F.1.4 Concentrations in soil

Radionuclides can be transferred to vegetation and top soils through irrigation or through capillary water rise, root-uptake and diffusion from the groundwater zone, if this is contaminated as a result of inflow of contaminated groundwater. Contamination through capillary water rise is not considered in the prefeasibility study for simplification reasons, and since it will be quite dependent on the specific local conditions.

[95] According to IAEA (2009) weathering half-lifes depend little on the cation species, but is more dependent on plant species and plant growth stage at time of deposition. Values typically vary between 10 and 40, and an average value is chosen for the calculations.
[96] Is the concentration in surface water, either fresh or coastal, calculated as described in Working Report 5

Irrigation will result in transfer of nuclides from the water in the shallow well to soil and plants. The rate constant governing the transfer of radionuclides from water to soil due to irrigation can be expressed as:

$C_s = C_W \times \frac{I_n \times n}{\varepsilon_{ts} \times D_{ts} \times R}$, where R is the retention coefficient:

$R = 1 + \frac{\rho_p \times K_D}{\varepsilon_{ts}}$,

where

D_{ts}= the thickness of the top soil layer

ε_{ts} = the porosity of the top soil layer

ρ_p = the particle density of the top soil.

The outflow of radionuclides from soil after irrigation is assumed to be to the well from which the irrigation water is taken. The consequence is that no radionuclides are lost from the system due to irrigation.

Suggested parameter values for the above calculations are shown in Table F.2.

Table F.2 Suggested parameters for calculation of soil concentrations

Parameter	Unit	Value	Reference
D_{ts}	m	0.5	Suggested value
ε_{ts}	m^3/m^3	0.4	Suggested typical value
ρ_p	Kg/m3	2650	Typical value

* These values will also be typical for Danish conditions

** Typical value for soil particles

F.2 Inhalation

Radionuclides in air emanate from two sources; dust in air from soil and released gaseous radionuclides. Dust may emanate from incidents at the repository that result in spreading of dust, which is further described in Chapter 8 of the main report and from resuspension of soil. Resuspended dust is assumed to have the same concentration of radioactivity as soil for the scenarios related to contamination of groundwater.

For these scenarios, the dust concentration in air is suggested set to 0.0001 kg/m^3[97]. This value is set conservatively and will be varied as part of the uncertainty evaluation. For the scenarios where dust is released to air directly from the repository, the dust concentration will vary with i.a. distance from source and the meteorological conditions and will be evaluated for each incident[98].

The calculation of concentrations of gaseous radionuclides in air (Ca) is described in Chapter 7.7. The relevant nuclides are ^{14}C (as CO_2) and ^{222}Rn. Emission of gaseous nuclides is primarily related to accidents occurring in the pre-closure phase and intrusion events post-closure. Gaseous emissions can also occur in case of leakage in the containers and repository walls due to e.g. settling and due to long term deterioration of top membranes for the near surface repositories.

F.3 Examples of variation of dose conversion factors

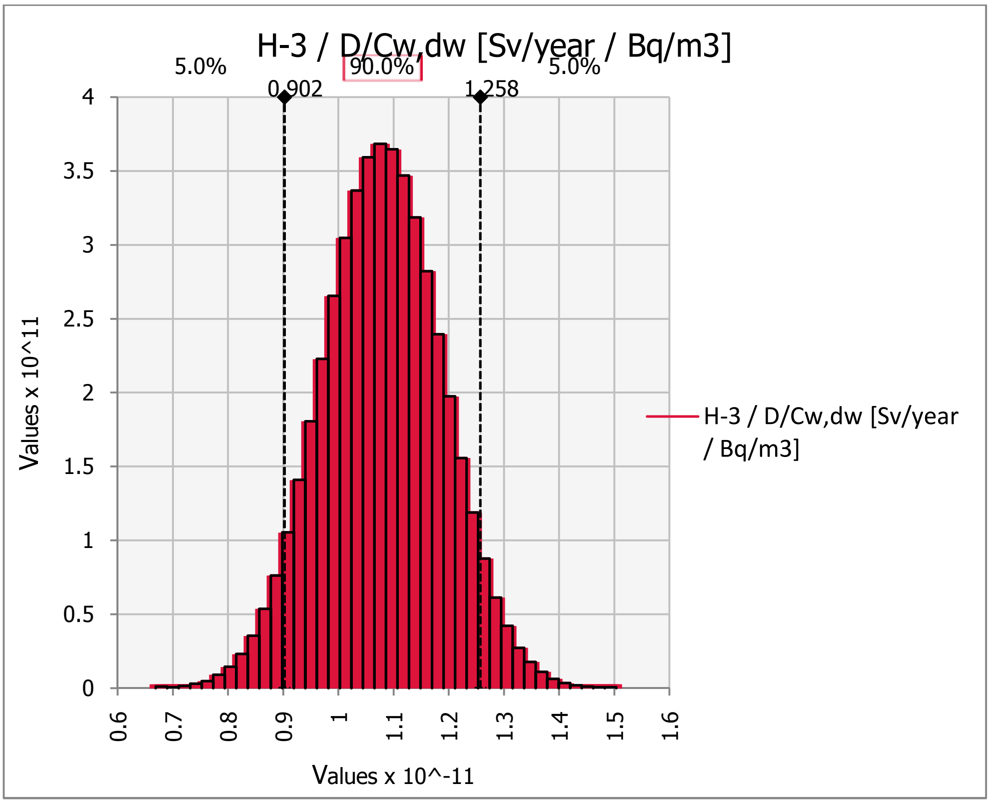
In this chapter examples of the variation of the dose conversion factors based on the possible variation/uncertainty related to the input parameters for a number of nuclides with different properties. The indices in the figures stand for:

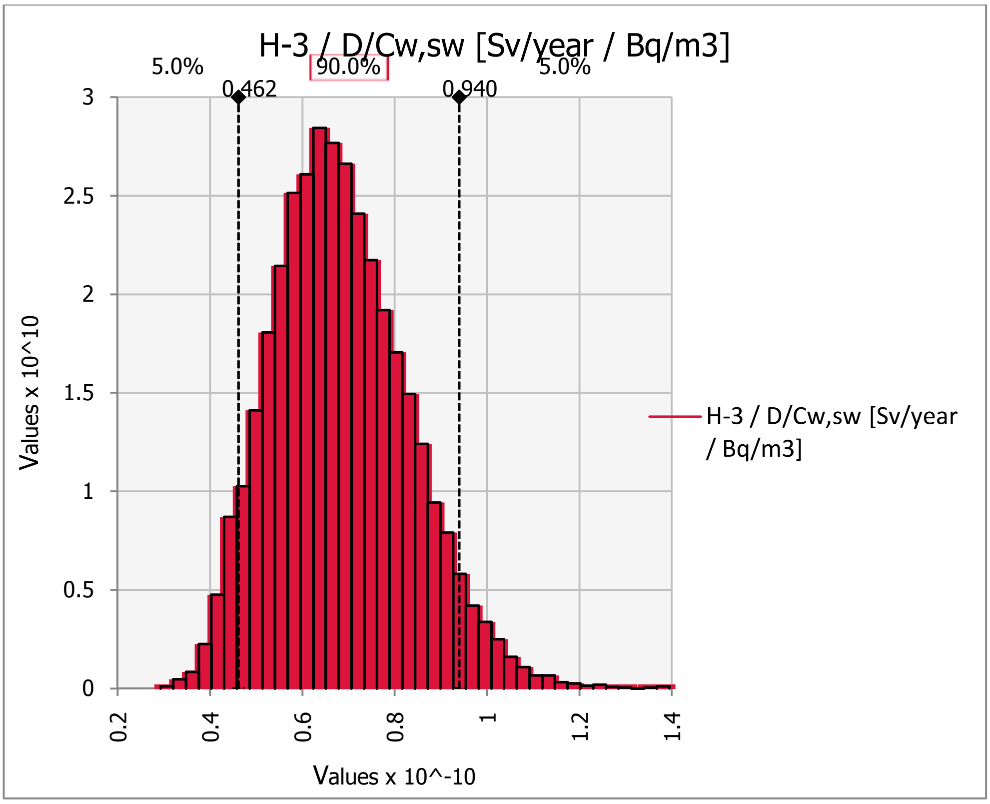
dw = deep well
sw = short well
st = stream
co = coastal waters.

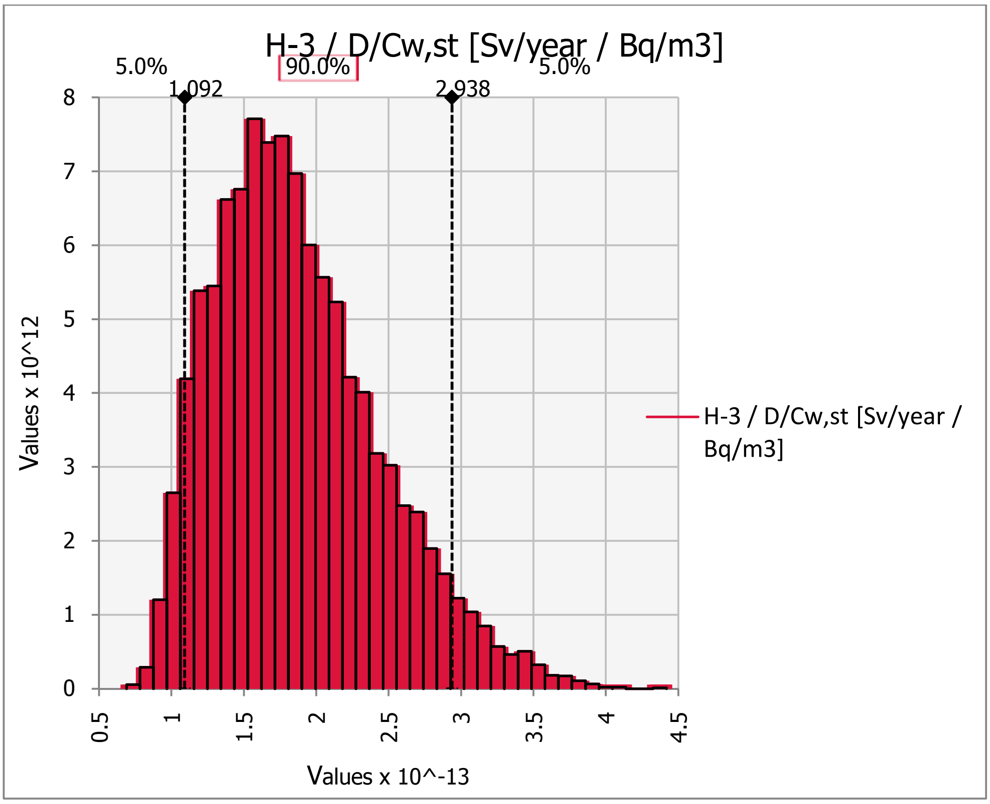
It can be seen from the figures that there is quite a bit variation in the variation between nuclides and recipients. It should be noted that this is partly due to the amount of data available to generate the variation spans for the different input parameters. Typically, the variation is larger, the more data that is available.

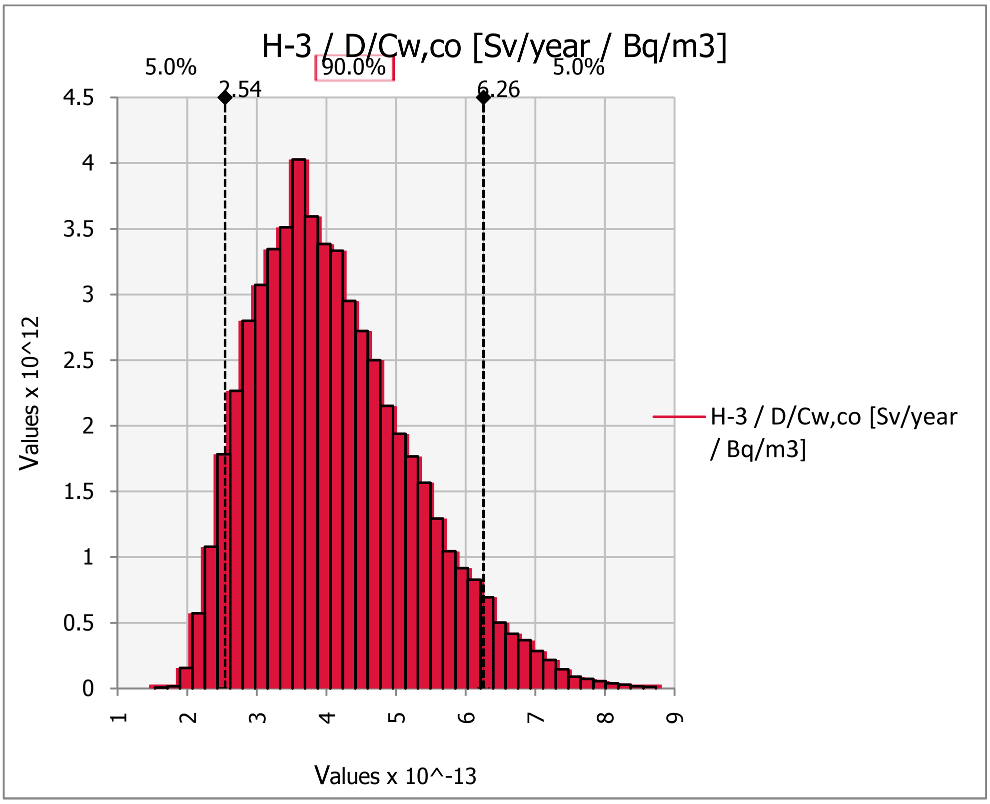
[97] (Karlsson, et al, 2001)
[98] Chapter 8.

F.3.1 H-3

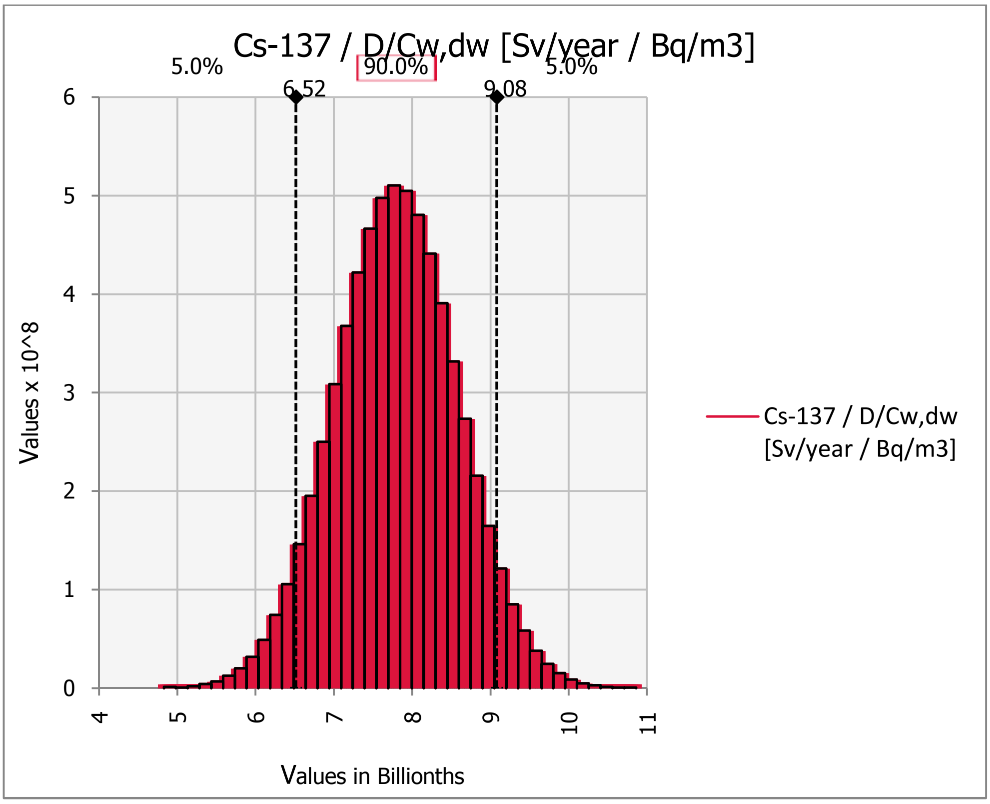

H-3 / D/Cw,dw [Sv/year / Bq/m3]
5.0%
90.0%
5.0%
0.902
1.258
4
3.5
3
2.5
2
1.5
1
0.5
0
Values x 10^11
0.6
0.7
0.8
0.9
1
1.1
1.2
1.3
1.4
1.5
1.6
Values x 10^-11
H-3 / D/Cw,dw [Sv/year / Bq/m3]

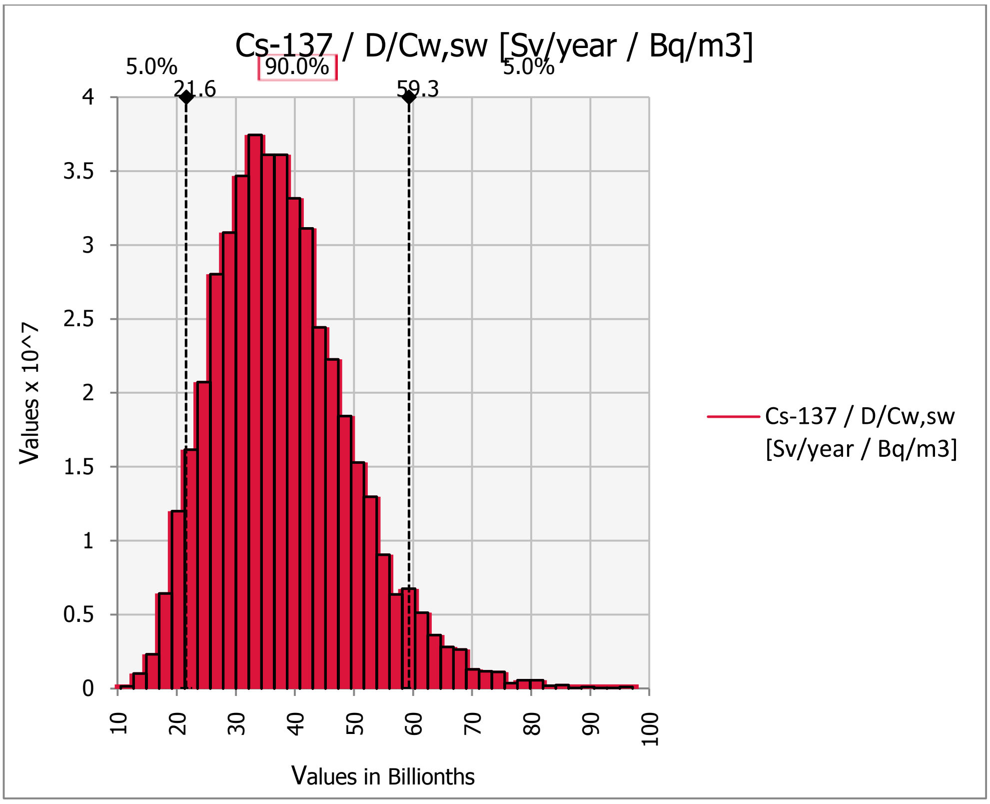

H-3 / D/Cw,sw [Sv/year / Bq/m3]
5.0%
90.0%
5.0%
0.462
0.940
3
2.5
2
1.5
1
0.5
0
Values x 10^10
0.2
0.4
0.6
0.8
1
1.2
1.4
Values x 10^-10
H-3 / D/Cw,sw [Sv/year / Bq/m3]

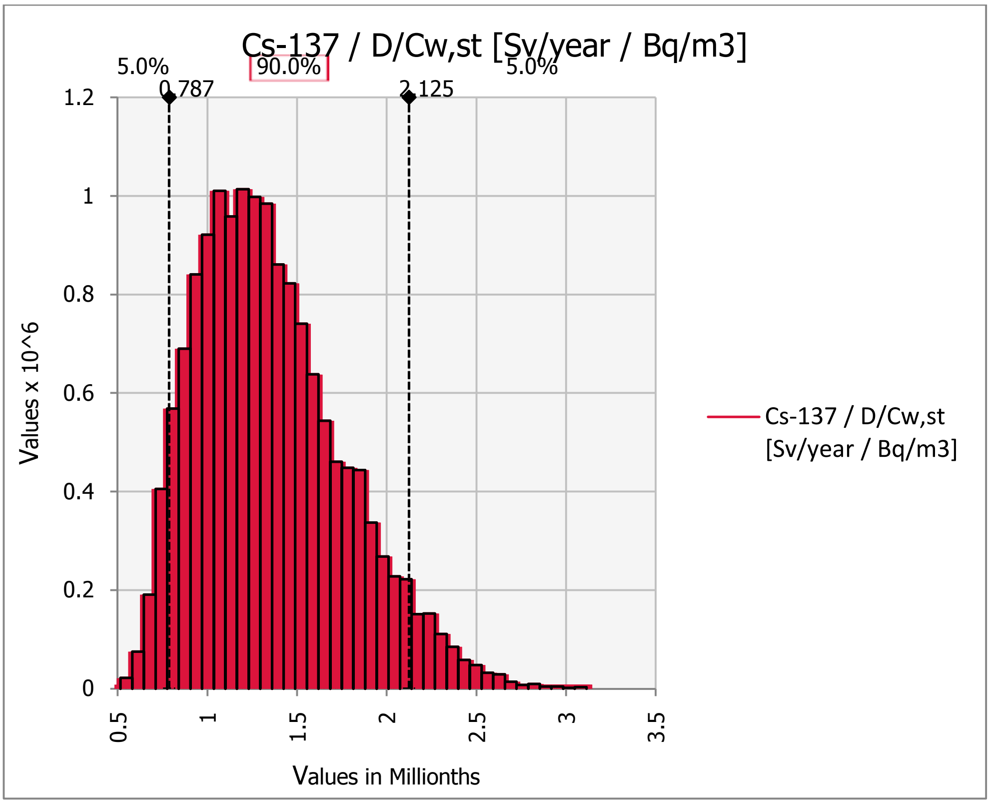

H-3 / D/Cw,st [Sv/year / Bq/m3]
5.0%
90.0%
5.0%
1.092
2.938
Values x 10^12
Values x 10^-13
H-3 / D/Cw,st [Sv/year / Bq/m3]

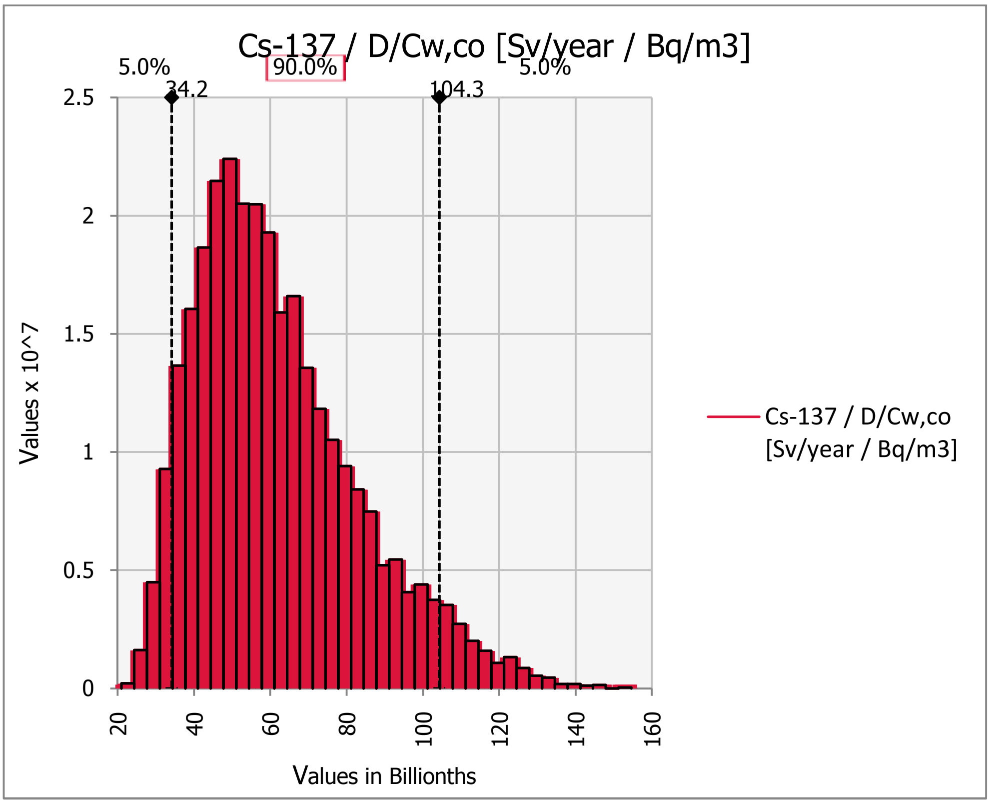

H-3 / D/Cw,co [Sv/year / Bq/m3]
5.0%
90.0%
5.0%
2.54
6.26
Values x 10^12
Values x 10^-13
H-3 / D/Cw,co [Sv/year / Bq/m3]

F.3.2 Cs-137

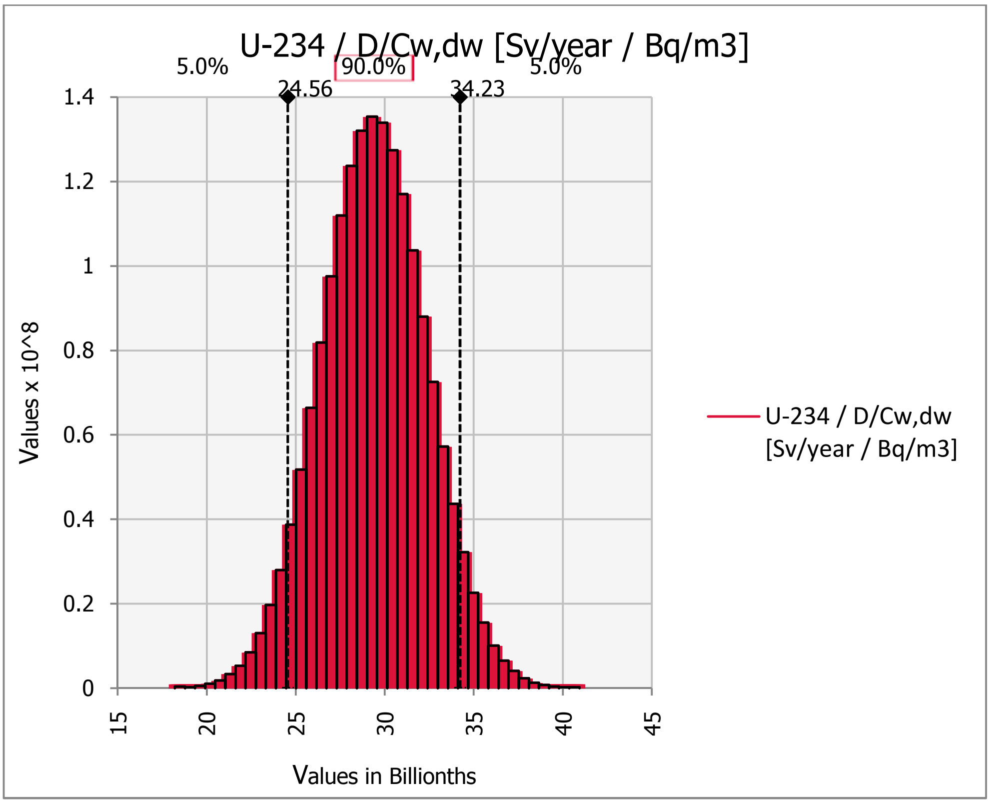

Cs-137 / D/Cw,dw [Sv/year / Bq/m3]
5.0%
90.0%
5.0%
6.52
9.08
Values x 10^8
0
1
2
3
4
5
6
4
5
6
7
8
9
10
11
Values in Billionths
Cs-137 / D/Cw,dw [Sv/year / Bq/m3]

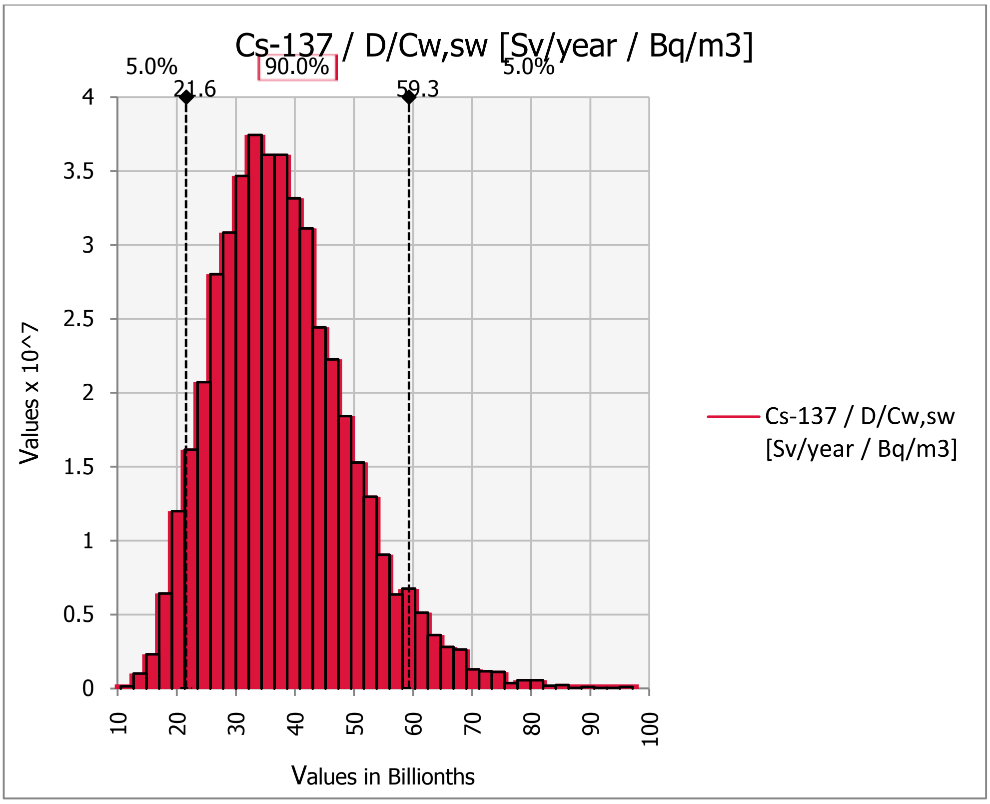

Cs-137 / D/Cw,sw [Sv/year / Bq/m3]
5.0%
90.0%
5.0%
21.6
59.3
Values x 10^7
0
0.5
1
1.5
2
2.5
3
3.5
4
10
20
30
40
50
60
70
80
90
100
Values in Billionths
Cs-137 / D/Cw,sw [Sv/year / Bq/m3]

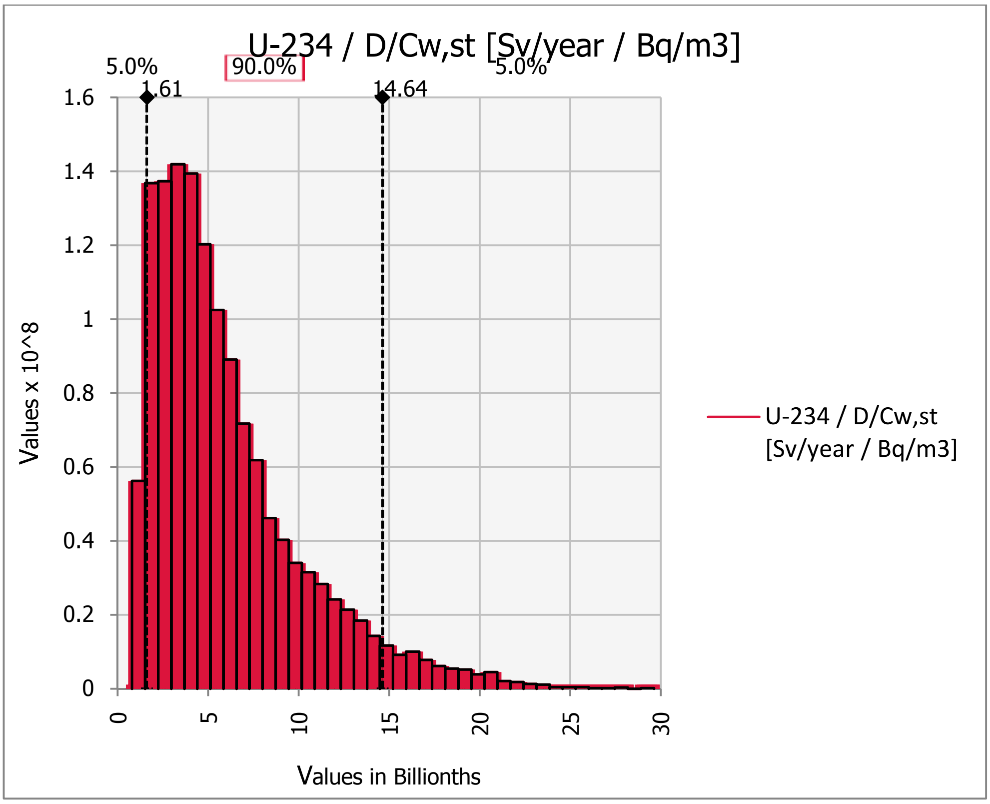

Cs-137 / D/Cw,st [Sv/year / Bq/m3]
5.0%
90.0%
5.0%
0.787
2.125
Values x 10^6
Values in Millionths
Cs-137 / D/Cw,st [Sv/year / Bq/m3]

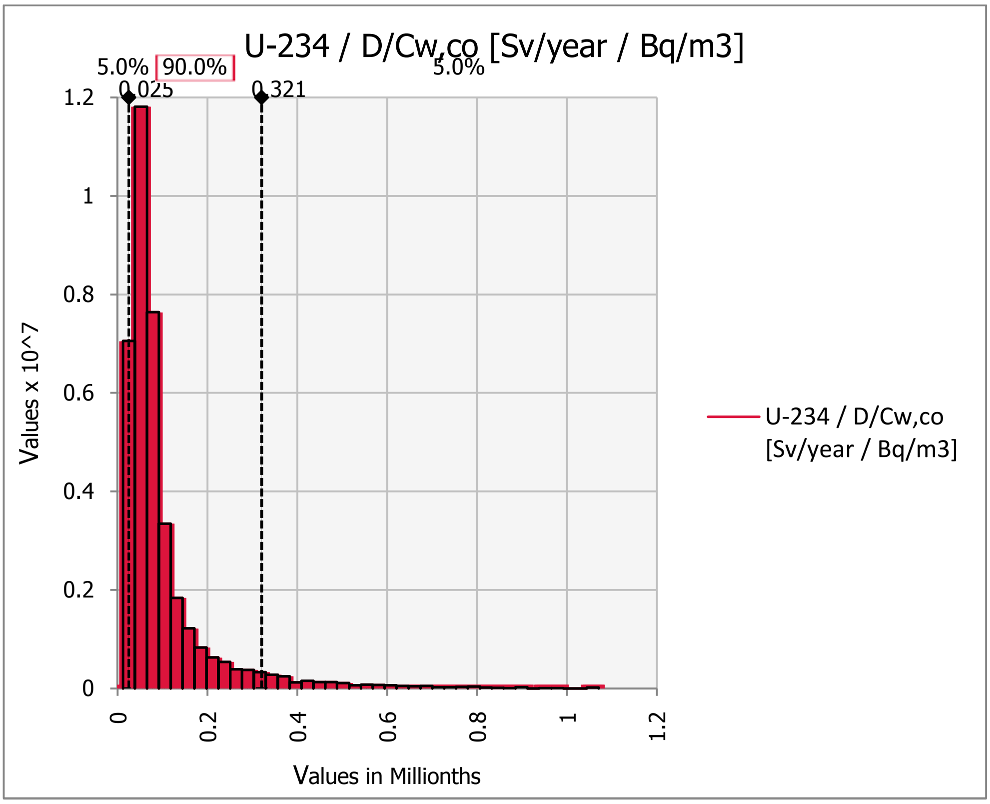

Cs-137 / D/Cw,co [Sv/year / Bq/m3]
5.0%
90.0%
5.0%
34.2
104.3
Values x 10^7
Values in Billionths
Cs-137 / D/Cw,co [Sv/year / Bq/m3]

F.3.3 U-234


U-234 / D/Cw,dw [Sv/year / Bq/m3]
5.0%
90.0%
5.0%
24.56
34.23
Values x 10^8
1.4
1.2
1
0.8
0.6
0.4
0.2
0
15
20
25
30
35
40
45
Values in Billionths
U-234 / D/Cw,dw [Sv/year / Bq/m3]


Cs-137 / D/Cw,sw [Sv/year / Bq/m3]
5.0%
90.0%
5.0%
21.6
59.3
Values x 10^7
4
3.5
3
2.5
2
1.5
1
0.5
0
10
20
30
40
50
60
70
80
90
100
Values in Billionths
Cs-137 / D/Cw,sw [Sv/year / Bq/m3]


U-234 / D/Cw,st [Sv/year / Bq/m3]
5.0%
90.0%
5.0%
1.61
14.64
1.6
1.4
1.2
1
0.8
0.6
0.4
0.2
0
Values x 10^8
0
5
10
15
20
25
30
Values in Billionths
U-234 / D/Cw,st [Sv/year / Bq/m3]


U-234 / D/Cw,co [Sv/year / Bq/m3]
5.0%
90.0%
5.0%
0.025
0.321
1.2
1
0.8
0.6
0.4
0.2
0
Values x 10^7
0
0.2
0.4
0.6
0.8
1
1.2
Values in Millionths
U-234 / D/Cw,co [Sv/year / Bq/m3]

Appendix G: Modelling of gaseous releases, details

G.1 Spreading in the atmosphere

The nuclide-specific concentration (C_a; Bq/m^3) in air at a point (x,y) at the ground level for a plume during conditions with a constant wind speed and direction will be modelled using the gaussian plume model (Pasquill & Smith, 1983):

$$C_a(x,y) = \frac{Q}{\pi \cdot u \cdot \sigma_y(x) \cdot \sigma_z(x)} \cdot exp\left[-\frac{1}{2}\cdot\left(\frac{y}{\sigma_y(x)}\right)^2\right] \cdot exp\left[-\frac{1}{2}\cdot\left(\frac{h}{\sigma_z(x)}\right)^2\right]$$

where

Q is the nuclide-specific release rate (Bq/s)
u is the wind speed (m/s)
h is the release height (m)

When the release is from the ground, $h=0$, $y=0$, such as in the case of resuspension simplification is made:

$$C_a(x) = \frac{Q}{\pi \cdot u \cdot \sigma_y(x) \cdot \sigma_z(x)}$$

The dispersion parameters σ_y and σ_z are functions of distance and atmospheric stability. A commonly used division of the stability is the so called Pasquill classes. The parameterisation that will be used, is ”Briggs' open-country” (Pasquill & Smith, 1983), see Table G.13.18.

Table G.13.18 Dispersion parameters according to Briggs (Pasquill & Smith, 1983).

Pasquill class	$\sigma_y(x)$	$\sigma_z(x)$
A	$\frac{0.22x}{\sqrt{1+0.0001x}}$	$0.20x$
B	$\frac{0.16x}{\sqrt{1+0.0001x}}$	$0.12x$
C	$\frac{0.11x}{\sqrt{1+0.0001x}}$	$\frac{0.08x}{\sqrt{1+0.0002x}}$
D	$\frac{0.08x}{\sqrt{1+0.0001x}}$	$\frac{0.06x}{\sqrt{1+0.0015x}}$
E	$\frac{0.06x}{\sqrt{1+0.0001x}}$	$\frac{0.03x}{1+0.0003x}$
F	$\frac{0.04x}{\sqrt{1+0.0001x}}$	$\frac{0.016x}{1+0.0003x}$

Some of the material in the plume will deposit on the ground, with the exception of noble gases. In the screening stage, wet deposition due to rain will be disregarded[99].

Dry deposition (q_{dry}; Bq/m^2) will be modelled as:

$$q_{dry}(x, y) = v_d \cdot C(x, y)t$$

where

v_d is a nuclide-specific deposition velocity (m/s), and

t is the duration of the release (s).

Table G.13.19 Data for calculation of dry deposition (Hallberg, 2001)

Substances	**Dry deposition velocity, v_d (m/s)**
Noble gases	0
Other substances	0.003

G.2 Indoor vapour intrusion

The model used is the newest version of the JAGG model from the Danish EPA for modelling of transport of volatile contaminants in soil on contaminated sites (Miljøstyrelsen, 2010)[100].

In this model only diffusive transport of volatiles is included in the transport through the soil. This is a reasonable assumption in cases where no substantial pressure difference exists between the source (here the groundwater plume) and the surface or the basement of the house. Steady state conditions are also assumed. Under these conditions transport of volatiles can be modeled using Fick's law:

$$\frac{dC}{dt} = 0 = N \cdot D_L \cdot \frac{d^2C}{dt^2} \div k_1 \cdot C,$$

where C is the pore air concentration, t is time, N a soil constant, D_L is the diffusion coefficient of the volatile compound, and k_1 is the decay constant of the compound.

---

99 Inclusion of wet deposition will enhance the exposure from the ground and decrease the exposure from cloud shine and inhalation. This is not expected to alter the difference in overall impact calculated, which is the main objective of the pre-feasibility study.

100 For further description of the model see this report. This model is built on the same principles and assumptions as the models described in for instance Andersen (2001)

Assuming the background concentration is zero, and the soil layers are homogeneous, the flux from the groundwater upwards can be expressed as a function of the source concentration:

$$J_z = \div N \cdot D_L \cdot \frac{dC}{dz},$$

$$J_z = C_L \cdot \sqrt{(N \cdot D_L) \cdot k_1} \cdot \frac{\cosh\left[(x \div z) \cdot \sqrt{k_1 / (N \cdot D_L)}\right]}{\sinh\left[x \cdot \sqrt{k_1 / (N \cdot D_L)}\right]},$$

where x is the depth to the source concentration (the depth to the groundwater), C_L is the concentration in the soil air at the source, and z is the height over the source, where the flux is determined.

The flux out of the ground and into the atmosphere is determined by ($z = x$):

$$J_z = C_L \cdot \sqrt{(N \cdot D_L) \cdot k_1} \cdot \frac{1}{\sinh\left[x \cdot \sqrt{k_1 / (N \cdot D_L)}\right]}.$$

If the diffusion passes through several soil layers of thickness x_n, N can be expressed as N':

$$N' = \frac{N_1 \cdot N_2 \cdot N_3 \cdot \ldots \cdot N_n}{(N_2 \cdot N_3 \cdot \ldots \cdot N_n) \cdot x_1 + (N_1 \cdot N_3 \cdot \ldots \cdot N_n) \cdot x_2 + (N_1 \cdot N_2 \cdot \ldots \cdot N_n) \cdot x_3 + (N_1 \cdot N_2 \cdot \ldots \cdot N_{n-1}) \cdot x_n},$$

where N_n is the soil constant for the nth soil layer. N can be determined by:

$$N = \left(V_L^{3,33} / (V_L + V_V)^2\right),$$

where V_L is the soil's air content, V_V is the soil's water content, and V_L+V_V is the soils porosity.

In the influence zone of a building, both diffusive and advective transport is relevant, since there normally will exist a pressure difference between a building and the surrounding ground due to temperature differences and ventilation of the house.

The diffusive flux can again be described by Fick's law (with the same assumptions as described above):

$$J = \div D \cdot \frac{dC}{dx} \approx \div D \cdot \frac{\Delta C}{\Delta x} = \div D \cdot \frac{C_{indoor} \div C_{pore\ air}}{x_b},$$

Where C is the concentration indoors and in the pore air beneath the house, respectively, and x_b is the distance (the thickness of the floor construction including capillary layers). The rest of the symbols are equal to the description given for flux through the soil layers.

Similarly as for the soil, a combined material constant can be calculated for the floor construction.

The total mass flow into the building is the product of the flux and the influenced area.

The advective transport (mass flow) through the floor construction can be described as:

$$Q_a = q_a \cdot C_{pore\ air} = \frac{l_{tot} \cdot w^3}{12 \cdot \mu} \cdot \frac{\Delta P \cdot C_{pore\ air}}{x_b},$$

where Q_a is the mass flow of the nuclide into the building. q_a is the amount of air entering the building through cracks in the floor construction, l_{tot} is the total length of the cracks, w is the width of the cracks, ΔP is the pressure difference across the floor construction, x_b is the thickness of the floor construction and μ is the dynamic viscosity of the air.

Complex equations exist for the calculation of l_{tot}, [101]but a simplified version is used here assuming only a crack along the walls of the house:

$$l_{tot} = 2 \cdot l_b + 2 \cdot w_b,$$

where l_b is the length and w_b the width of the building.

The width of the cracks can be calculated as:

$$w = \frac{\varepsilon_f \cdot A}{l_{tot}},$$

where ε_f is the shrinkage of the concrete and A is the area of the building. Complex equations also exist for the calculation of ε_f dependent the humidity, the cement and water content of the concrete used in the floor construction and the drying time for the concrete[102]. A typical figure is 0.03%, which will be used in the calculations here.

Since steady state is assumed, the mass balance directly under the floor can be expressed as:

[101] Also in the JAGG model, see for instance Miljøstyrelsen (1998, a & b)

[102] Also in the JAGG model, see for instance Miljøstyrelsen (1998, a & b)

$$Q_{diffusion,soil} = Q_{diffusion,floor} + Q_{advection\ floor} =>$$

$$\div K_{soil} \cdot \left(C_{pore\ air} \div C_L\right) \cdot A = \left(\div K_N \cdot \left(C_{indoor} \div C_{pore\ air}\right)\right) \cdot A + q_a \cdot C_{pore\ air},$$

where

$$K_{soil} = x \cdot \sqrt{(N' \cdot D_L) \cdot k_1} \times \frac{\cosh\left[(x \div z) \cdot \sqrt{k_1 / (N' \cdot D_L)}\right]}{\sinh\left[x \cdot \sqrt{k_1 / (N' \cdot D_L)}\right]},$$ [103]

and

$$K_N = \div \frac{(N_1 \cdot N_2 \cdot N_3) \cdot D_L}{N_2 \cdot N_3 \cdot x_1 + N_1 \cdot N_3 \cdot x_2 + N_1 \cdot N_2 \cdot x_3},$$

assuming a 3 layer floor construction.

The mass balance for the flow out of the building is:

$$Q_{out,building} = Q_{difusion,floor} + Q_{advection,floor} =>$$

$$C_{indoor} \cdot h_b \cdot R_b \cdot A = \left(\div K_N \cdot \left(C_{indoor} \div C_{pore\ air}\right)\right) \cdot A + q_a \cdot C_{pore\ air},$$

where h_b is the height of the room affected, R_b is the air removal rate in the room due to ventilation. For the rest of the symbols see previous equations.

From the two mass balances an expression for C_{indoor} can be derived:

$$C_{indoor} = \frac{\left(K_N + \frac{q_a}{A}\right) \cdot C_L}{(h_b \cdot R_b) + \frac{K_N \cdot (h_b \cdot R_b)}{K_{soil}} + K_N + \frac{(h_b \cdot R_b)}{K_{soil}} \cdot \frac{q_a}{A}}$$

G.3 Diffusion of gases directly from the repository

Diffusion of gaseous compounds through the covering soil of the repository (and through the repository construction and waste packages) is modelled similarly to the diffusion of gases described in the beginning of chapter G.2.

[103] See previously for the background for the equation

Appendix H: Assessments of accidents, details

H.1 Selected parameters used in frequency and dose probability estimation

Critical release versus waste type and drop height						
	Waste type 4		Waste type 8		Waste type 21	
Drop height	P(release, fill)	P(release, no fill)	P(release, fill)	P(release, no fill)	P(release, fill)	P(release, no fill)
0	0	0	0	0	0	0
1.5	0	0	0	0	0	0
3	0.05	0.005	0	0.02	0.05	0.005
4.5	0.1	0.01	0	0.1	0.1	0.01
10	0.5	0.5	0.02	0.2	0.5	0.5
20	1	1	0.1	0.5	1	1
30			0.2	1		
40			0.5	1		
50			1	1		

Weather conditions		
wind speed [m/s]	P (weather)	
	D	F
0.5	0.10	0.05
2	0.30	0.10
5	0.30	0.00
10	0.15	0.00

Dose factor for retention within repository (building and repository effect)

Repository 0

min	most	max
0.1	0.1	0.1

Repository 1

min	most	max
1	1	1

Repository 2

min	most	max
0.1	0.1	0.1

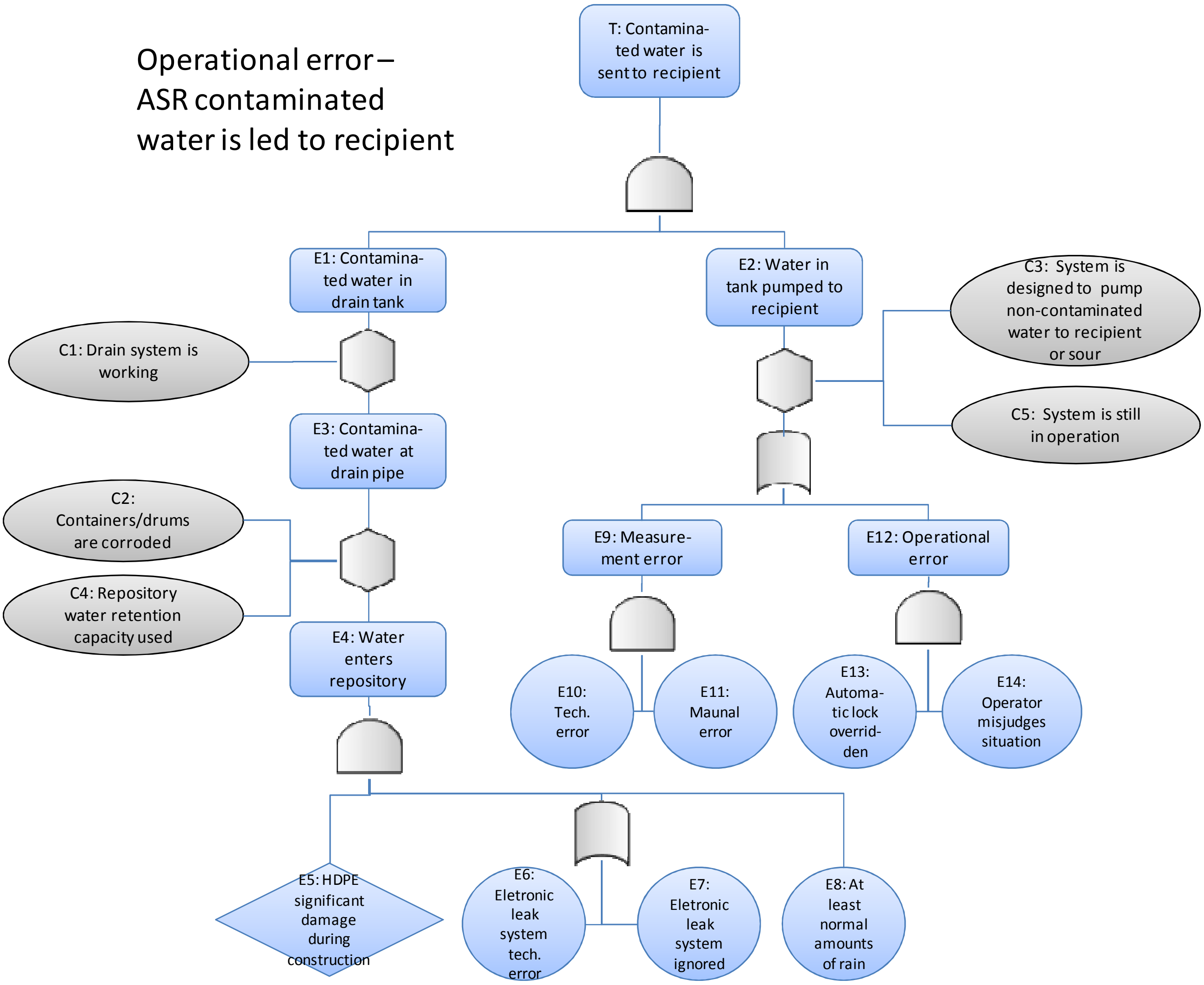
	Repository 3, 4, 5, 6, 7, 8			Repository 9			Repository 10		
Drop height	min	most	max	min	most	max	min	most	max
0	0.1	0.1	0.1	0.1	0.055	0.01	0.1	0.055	0.01
20	0.09	0.09	0.09	0.09	0.0495	0.009	0.09	0.0495	0.009
30	0.08	0.08	0.08	0.08	0.044	0.008	0.08	0.044	0.008
40	0.07	0.07	0.07	0.07	0.0385	0.007	0.07	0.0385	0.007
50	0.06	0.06	0.06	0.06	0.033	0.006	0.06	0.033	0.006
60	0.05	0.05	0.05	0.05	0.0275	0.005	0.05	0.0275	0.005
70	0.04	0.04	0.04	0.04	0.022	0.004	0.04	0.022	0.004
80	0.03	0.03	0.03	0.03	0.0165	0.003	0.03	0.0165	0.003
90	0.02	0.02	0.02	0.02	0.011	0.002	0.02	0.011	0.002
100	0.01	0.01	0.01	0.01	0.0055	0.001	0.01	0.0055	0.001

Drop heights versus repository

Repository	min	max
1	1	4.5
2	1	7
3	1	56.5
4	1	77.5
5	1	104.5
6	1	56.5
7	1	77.5
8	1	104.5
9	1	56.5
10	1	4.5
11	1	200

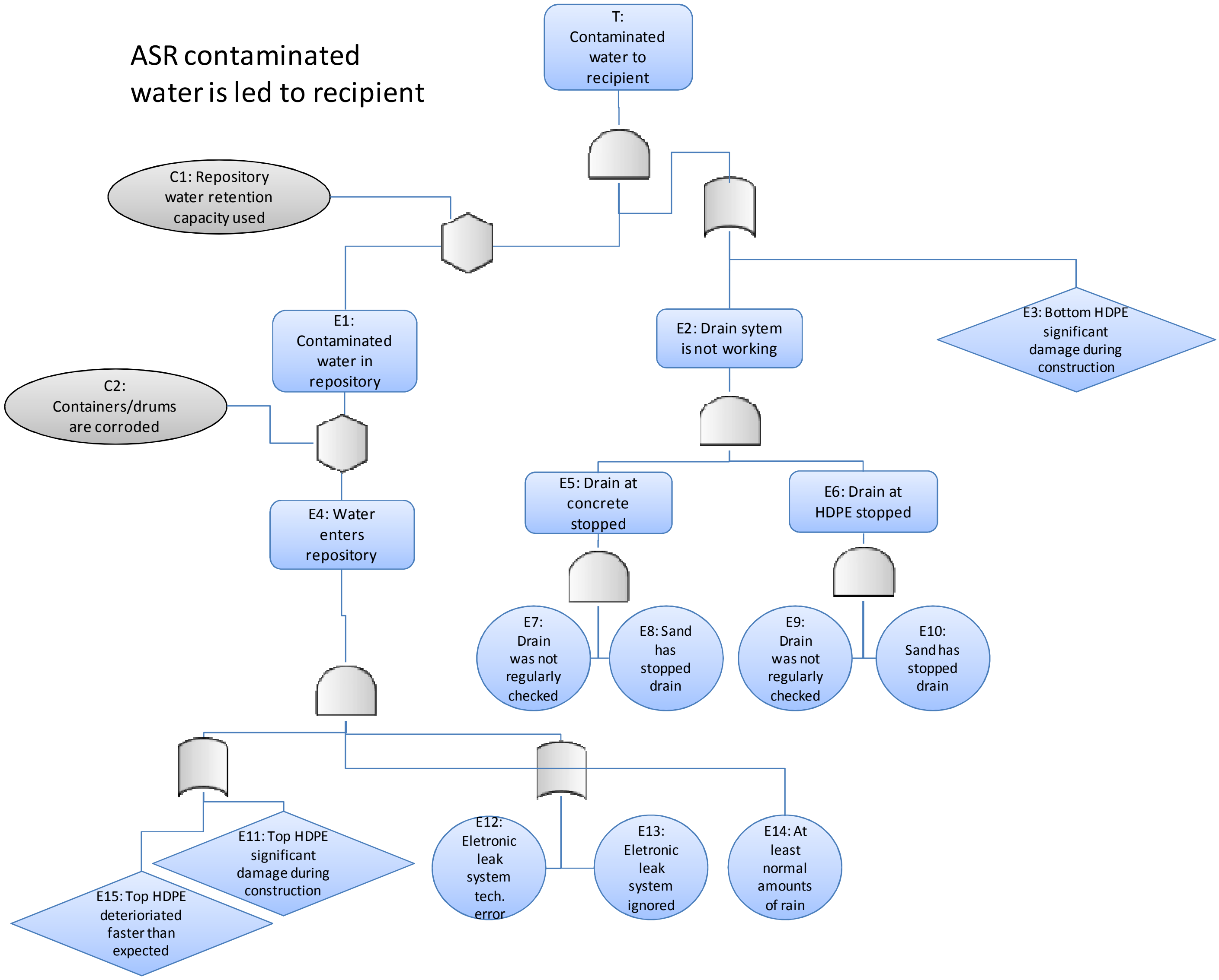
Danger is noticed when digging in repository	
Excavation year	P danger not noticed
300	0.5
1,000	1
10,000	1

H.2 Fault tree analysis of operation error leading to discharge of contaminated water to recipient.



	Time			
Event	**10**	**30**	**100**	**300**
T (per year)	0.0E+00	7.6E-07	1.9E-05	8.4E-06
E1	0.0E+00	1.3E-04	5.6E-04	6.8E-04
E2	2.6E-03	6.0E-03	3.4E-02	1.3E-02
E3	0.0E+00	1.7E-04	2.3E-03	6.8E-03
E4	6.0E-04	2.7E-03	6.0E-03	7.5E-03
E5	1%	1%	1%	1%
E6	5%	25%	50%	50%
E7	1%	2%	10%	25%
E8	100%	100%	100%	100%
E9	1.0E-04	2.0E-03	2.5E-02	1.3E-01
E10	1%	10%	25%	50%
E11	1%	2%	10%	25%
E12	5.0E-03	1.0E-02	5.0E-02	1.3E-01
E13&E14	1%	1%	5%	13%
C1	90%	75%	25%	10%
C2	1%	25%	50%	90%
C3	50%	50%	50%	50%
C4	0%	25%	75%	100%
C5	100%	100%	90%	10%

H.3 Fault tree analysis of drain system failure



	Time			
Event	**10**	**30**	**100**	**300**
T (per year)	0.0E+00	1.9E-06	1.7E-04	5.3E-03
E1	6.0E-06	7.4E-04	1.8E-02	7.4E-02
E2	1.0E-06	1.6E-05	2.5E-03	6.1E-02
E3	1%	1%	1%	1%
E4	6.0E-04	3.0E-03	3.6E-02	8.3E-02
E5	1.0E-03	4.0E-03	5.0E-02	2.5E-01
E6	1.0E-03	4.0E-03	5.0E-02	2.5E-01
E7	1%	2%	10%	25%
E8	10%	20%	50%	99%
E9	1%	2%	10%	25%
E10	10%	20%	50%	99%
E11	1%	1%	1%	1%
E12	5%	25%	50%	50%
E13	1%	2%	10%	25%
E14	100%	100%	100%	100%
E15	0%	0%	5%	10%
C1	0%	25%	75%	100%
C2	1%	25%	50%	90%

H.4 Additional results – event based exposure from accidents

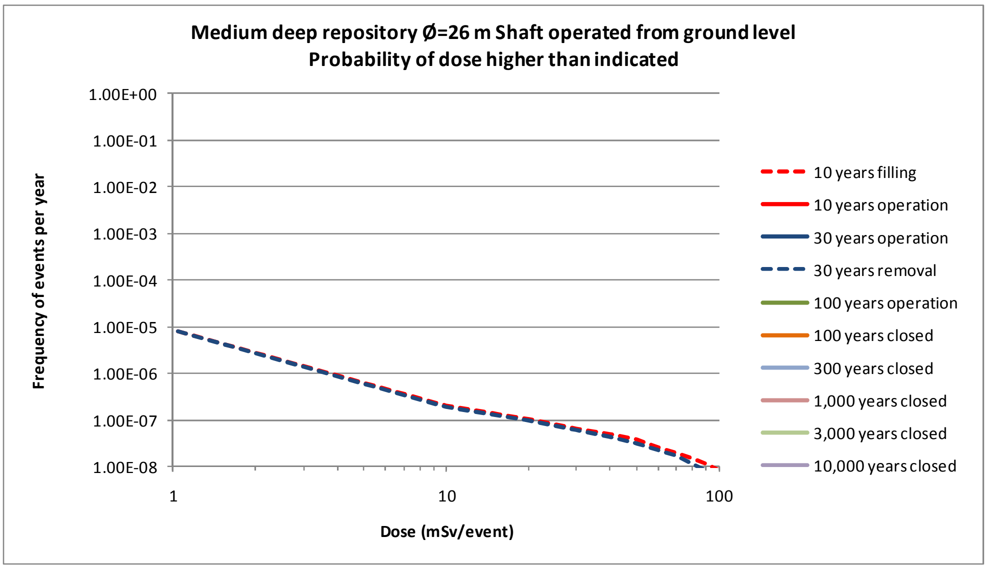


Figure H.1 Frequency-dose diagram for medium deep repository (diameter 26 m) shaft operated from ground level.

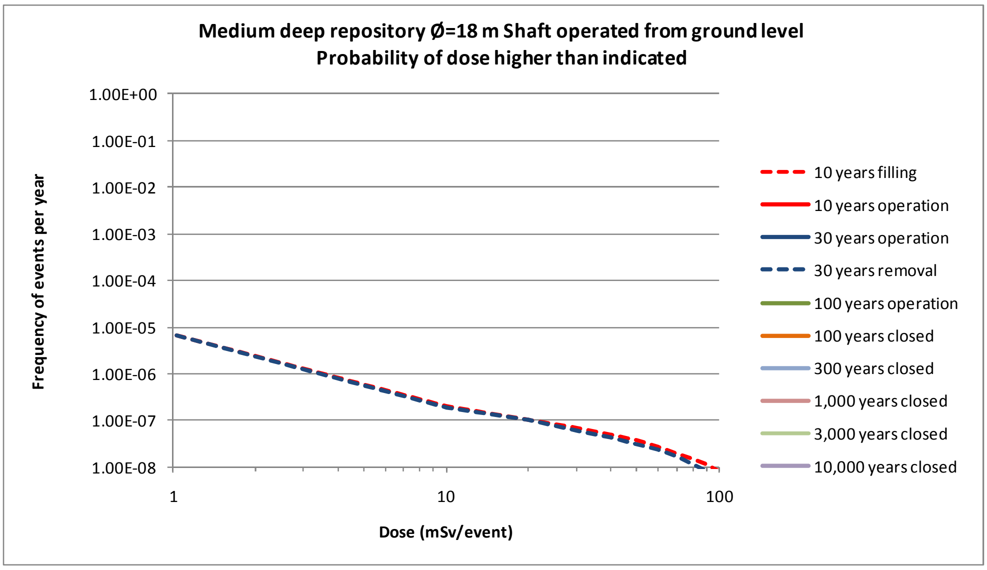


Figure H.2 Frequency-dose diagram for medium deep repository (diameter 18 m) shaft operated from ground level.

H.5 Additional results – long term exposure from accidents

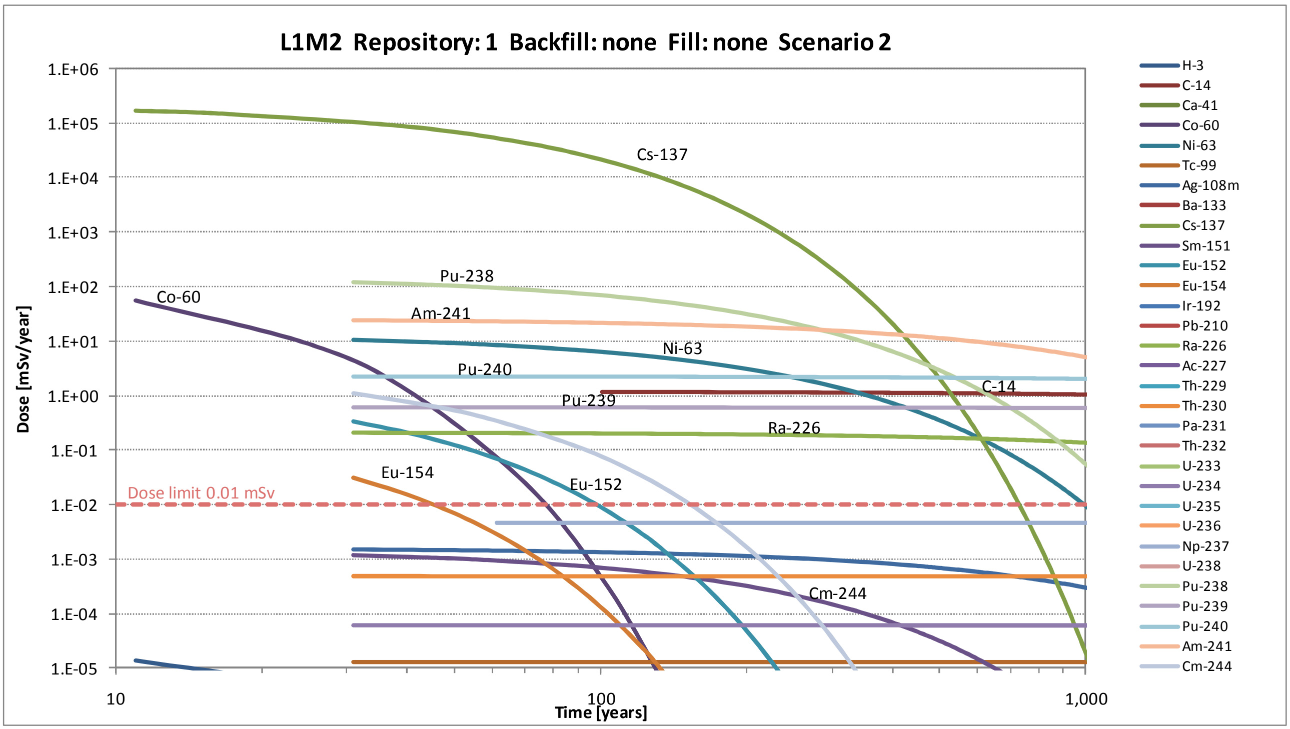


Figure H.3 Frequency-dose diagram for events with long term effects, ASR placed on fat clay.

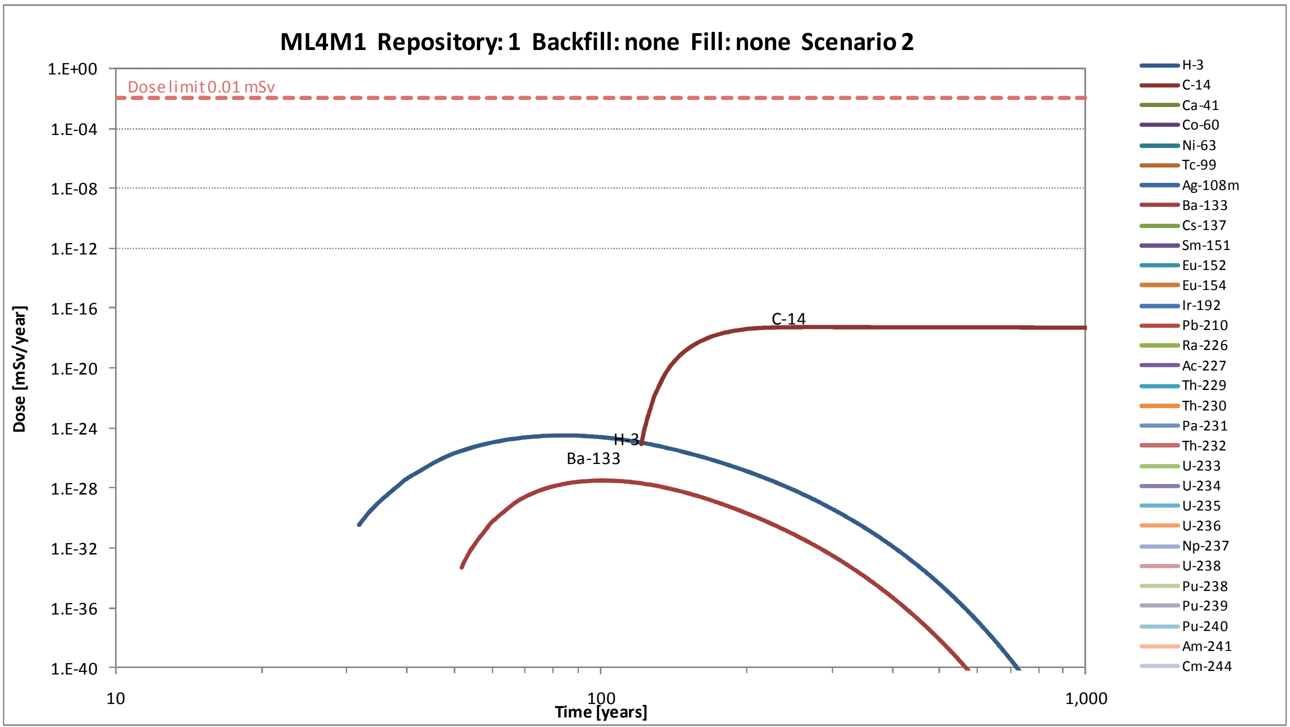


Figure H.4 Frequency-dose diagram for events with long term effects, ASR placed on clay till (geology M4M1).

Appendix I: Cost estimates, details

On the next three pages (in landscape format) the outcome of the cost estimates is summarised in tabulated forms for:

- The most likely cost (no variation in prices)
- The minimum costs (min variation/percentage for all prices)
- The maximum costs (max variation/percentage for all prices).

Repository facility type no.	Main repository concept [1]	Cut-off structure [2]	Internal diameter	INITIAL costs					ADDITIONAL costs									TOTAL costs
				Land	Facilities	Borehole	Main repository [5]	SUM initial costs	Additional operation 1st year borehole	Additional operation 1st year main repository	Basic operation 31 years	SUBSUM operation	Closure borehole	Closure main repository	SUBSUM closure	Monitoring	SUM additional cost	
MOST LIKELY COSTS General uncertainty given as total min and max percentage of estimated cost or as plus/minus variation				Total: Min: 25% Max: 250%	Total: Min: 75% Max: 150%	Variation: ±17.5%	Variation: ±17.5%		Total: Min: 75% Max: 150%	Total: Min: 75% Max: 150%	Total: Min: 75% Max: 150%		Variation: ±17.5%	Variation: ±17.5%		Total: Min: 75% Max: 150%		
1	ASR	–	–	5,000,000	13,000,000	6,621,000	22,031,000	46,652,000	2,653,000	7,228,000	229,605,000	239,486,000	500,000	3,958,000	4,458,000	11,427,000	255,371,000	302,023,000
2	NSR	Sheet piles	–	5,000,000	13,000,000	6,621,000	45,730,000	70,351,000	2,653,000	11,277,000	229,605,000	243,535,000	500,000	18,582,000	19,082,000	11,427,000	274,044,000	344,395,000
3	MDR, GI [3]	DW	33,8 m	5,000,000	13,000,000	6,621,000	194,666,000	219,287,000	2,653,000	13,555,000	229,605,000	245,813,000	500,000	92,909,000	93,409,000	11,427,000	350,649,000	569,936,000
4		DW	26 m	5,000,000	13,000,000	6,621,000	159,074,000	183,695,000	2,653,000	13,555,000	229,605,000	245,813,000	500,000	48,788,000	49,288,000	11,427,000	306,528,000	490,223,000
5		DW	18 m	5,000,000	13,000,000	6,621,000	129,551,000	154,172,000	2,653,000	13,555,000	229,605,000	245,813,000	500,000	23,714,000	24,214,000	11,427,000	281,454,000	435,626,000
6		SP&SCL	33,8 m	5,000,000	13,000,000	6,621,000	190,963,000	215,584,000	2,653,000	13,555,000	229,605,000	245,813,000	500,000	92,909,000	93,409,000	11,427,000	350,649,000	566,233,000
7		SP&SCL	26 m	5,000,000	13,000,000	6,621,000	156,124,000	180,745,000	2,653,000	13,555,000	229,605,000	245,813,000	500,000	48,788,000	49,288,000	11,427,000	306,528,000	487,273,000
8		SP&SCL	18 m	5,000,000	13,000,000	6,621,000	127,706,000	152,327,000	2,653,000	13,555,000	229,605,000	245,813,000	500,000	23,714,000	24,214,000	11,427,000	281,454,000	433,781,000
9	MDR, GR [3]	DW	33,8 m	5,000,000	13,000,000	6,621,000	190,228,000	214,849,000	2,653,000	15,078,000	229,605,000	247,336,000	500,000	16,989,000	17,489,000	11,427,000	276,252,000	491,101,000
10		DW	26 m	5,000,000	13,000,000	6,621,000	238,928,000	263,549,000	2,653,000	15,078,000	229,605,000	247,336,000	500,000	11,276,000	11,776,000	11,427,000	270,539,000	534,088,000
11		DW	18 m	5,000,000	13,000,000	6,621,000	241,419,000	266,040,000	2,653,000	15,078,000	229,605,000	247,336,000	500,000	6,967,000	7,467,000	11,427,000	266,230,000	532,270,000
12		SP&SCL	33,8 m	5,000,000	13,000,000	6,621,000	186,525,000	211,146,000	2,653,000	15,078,000	229,605,000	247,336,000	500,000	16,989,000	17,489,000	11,427,000	276,252,000	487,398,000
13		SP&SCL	26 m	5,000,000	13,000,000	6,621,000	235,978,000	260,599,000	2,653,000	15,078,000	229,605,000	247,336,000	500,000	11,276,000	11,776,000	11,427,000	270,539,000	531,138,000
14		SP&SCL	18 m	5,000,000	13,000,000	6,621,000	239,574,000	264,195,000	2,653,000	15,078,000	229,605,000	247,336,000	500,000	6,967,000	7,467,000	11,427,000	266,230,000	530,425,000
15	MDR, IR [3], [4]	DW	33,8 m	5,000,000	13,000,000	6,621,000	272,039,000	296,660,000	2,653,000	49,511,000	229,605,000	281,769,000	500,000	16,989,000	17,489,000	11,427,000	310,685,000	607,345,000
16		SP&SCL	33,8 m	5,000,000	13,000,000	6,621,000	268,336,000	292,957,000	2,653,000	49,511,000	229,605,000	281,769,000	500,000	16,989,000	17,489,000	11,427,000	310,685,000	603,642,000
17	MDR, CA	SP&SCL	–	5,000,000	13,000,000	6,621,000	163,654,000	188,275,000	2,653,000	15,078,000	229,605,000	247,336,000	500,000	11,892,000	12,392,000	11,427,000	271,155,000	459,430,000
18		SP&RO	–	5,000,000	13,000,000	6,621,000	77,967,000	102,588,000	2,653,000	15,078,000	229,605,000	247,336,000	500,000	11,892,000	12,392,000	11,427,000	271,155,000	373,743,000

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[2] DW: diaphragm wall, SP: secant piles, SCL: sprayed concrete lining, RO: rock

[3] The construction costs of all shaft based MDRs are based on the assumption that the bottom slab is located at the deepest depth possible determined by the structural capacity with given internal diameter and thickness of external walls. In particular, the diameter 33,8 m MDRs could be placed less deep, which would lead to a decrease in the construction costs.

[4] The diameter 26 m and 18 m MDR, IR concepts interfere with the minimum volume requirements and have thus been excluded from the cost estimate.

[5] The costs of the main repository include costs of cut-off structures and excavation.

Repository facility type no.	Main repository concept [1]	Cut-off structure [2]	Internal diameter	INITIAL costs					ADDITIONAL costs									TOTAL costs
				Land	Facilities	Borehole	Main repository [5]	SUM initial costs	Additional operation 1st year borehole	Additional operation 1st year main repository	Basic operation 31 years	SUBSUM operation	Closure borehole	Closure main repository	SUBSUM closure	Monitoring	SUM additional cost	
MIN percentage of most likely costs				25%	75%	82.5%	82.5%		75%	75%	75%		82.5%	82.5%		75%		
1	ASR	–	–	1,250,000	9,750,000	5,462,000	18,176,000	34,638,000	1,990,000	5,421,000	172,204,000	179,615,000	413,000	3,265,000	3,678,000	8,570,000	191,863,000	226,501,000
2	NSR	Sheet piles	–	1,250,000	9,750,000	5,462,000	37,727,000	54,189,000	1,990,000	8,458,000	172,204,000	182,652,000	413,000	15,330,000	15,743,000	8,570,000	206,965,000	261,154,000
3	MDR, GI [3]	DW	33,8 m	1,250,000	9,750,000	5,462,000	160,599,000	177,061,000	1,990,000	10,166,000	172,204,000	184,360,000	413,000	76,650,000	77,063,000	8,570,000	269,993,000	447,054,000
4		DW	26 m	1,250,000	9,750,000	5,462,000	131,236,000	147,698,000	1,990,000	10,166,000	172,204,000	184,360,000	413,000	40,250,000	40,663,000	8,570,000	233,593,000	381,291,000
5		DW	18 m	1,250,000	9,750,000	5,462,000	106,880,000	123,342,000	1,990,000	10,166,000	172,204,000	184,360,000	413,000	19,564,000	19,977,000	8,570,000	212,907,000	336,249,000
6		SP&SCL	33,8 m	1,250,000	9,750,000	5,462,000	157,544,000	174,006,000	1,990,000	10,166,000	172,204,000	184,360,000	413,000	76,650,000	77,063,000	8,570,000	269,993,000	443,999,000
7		SP&SCL	26 m	1,250,000	9,750,000	5,462,000	128,802,000	145,264,000	1,990,000	10,166,000	172,204,000	184,360,000	413,000	40,250,000	40,663,000	8,570,000	233,593,000	378,857,000
8		SP&SCL	18 m	1,250,000	9,750,000	5,462,000	105,357,000	121,819,000	1,990,000	10,166,000	172,204,000	184,360,000	413,000	19,564,000	19,977,000	8,570,000	212,907,000	334,726,000
9	MDR, GR [3]	DW	33,8 m	1,250,000	9,750,000	5,462,000	156,938,000	173,400,000	1,990,000	11,309,000	172,204,000	185,503,000	413,000	14,016,000	14,429,000	8,570,000	208,502,000	381,902,000
10		DW	26 m	1,250,000	9,750,000	5,462,000	197,116,000	213,578,000	1,990,000	11,309,000	172,204,000	185,503,000	413,000	9,303,000	9,716,000	8,570,000	203,789,000	417,367,000
11		DW	18 m	1,250,000	9,750,000	5,462,000	199,171,000	215,633,000	1,990,000	11,309,000	172,204,000	185,503,000	413,000	5,748,000	6,161,000	8,570,000	200,234,000	415,867,000
12		SP&SCL	33,8 m	1,250,000	9,750,000	5,462,000	153,883,000	170,345,000	1,990,000	11,309,000	172,204,000	185,503,000	413,000	14,016,000	14,429,000	8,570,000	208,502,000	378,847,000
13		SP&SCL	26 m	1,250,000	9,750,000	5,462,000	194,682,000	211,144,000	1,990,000	11,309,000	172,204,000	185,503,000	413,000	9,303,000	9,716,000	8,570,000	203,789,000	414,933,000
14		SP&SCL	18 m	1,250,000	9,750,000	5,462,000	197,649,000	214,111,000	1,990,000	11,309,000	172,204,000	185,503,000	413,000	5,748,000	6,161,000	8,570,000	200,234,000	414,345,000
15	MDR, IR [3], [4]	DW	33,8 m	1,250,000	9,750,000	5,462,000	224,432,000	240,894,000	1,990,000	37,133,000	172,204,000	211,327,000	413,000	14,016,000	14,429,000	8,570,000	234,326,000	475,220,000
16		SP&SCL	33,8 m	1,250,000	9,750,000	5,462,000	221,377,000	237,839,000	1,990,000	37,133,000	172,204,000	211,327,000	413,000	14,016,000	14,429,000	8,570,000	234,326,000	472,165,000
17	MDR, CA	SP&SCL	–	1,250,000	9,750,000	5,462,000	135,015,000	151,477,000	1,990,000	11,309,000	172,204,000	185,503,000	413,000	9,811,000	10,224,000	8,570,000	204,297,000	355,774,000
18		SP&RO	–	1,250,000	9,750,000	5,462,000	64,323,000	80,785,000	1,990,000	11,309,000	172,204,000	185,503,000	413,000	9,811,000	10,224,000	8,570,000	204,297,000	285,082,000

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[2] DW: diaphragm wall, SP: secant piles, SCL: sprayed concrete lining, RO: rock

[3] The construction costs of all shaft based MDRs are based on the assumption that the bottom slab is located at the deepest depth possible determined by the structural capacity with given internal diameter and thickness of external walls. In particular, the diameter 33,8 m MDRs could be placed less deep, which would lead to a decrease in the construction costs.

[4] The diameter 26 m and 18 m MDR, IR concepts interfere with the minimum volume requirements and have thus been excluded from the cost estimate.

[5] The costs of the main repository include costs of cut-off structures and excavation.

Repository facility type no.	Main repository concept [1]	Cut-off structure [2]	Internal diameter	INITIAL costs					ADDITIONAL costs									TOTAL costs
				Land	Facilities	Borehole	Main repository [5]	SUM initial costs	Additional operation 1st year borehole	Additional operation 1st year main repository	Basic operation 31 years	SUBSUM operation	Closure borehole	Closure main repository	SUBSUM closure	Monitoring	SUM additional cost	
MAX percentage of most likely costs				250%	150%	117.5%	117.5%		150%	150%	150%		117.5%	117.5%		150%		
1	ASR	–	–	12,500,000	19,500,000	7,780,000	25,886,000	65,666,000	3,980,000	10,842,000	344,408,000	359,230,000	588,000	4,651,000	5,239,000	17,141,000	381,610,000	447,276,000
2	NSR	Sheet piles	–	12,500,000	19,500,000	7,780,000	53,733,000	93,513,000	3,980,000	16,916,000	344,408,000	365,304,000	588,000	21,834,000	22,422,000	17,141,000	404,867,000	498,380,000
3	MDR, GI [3]	DW	33,8 m	12,500,000	19,500,000	7,780,000	228,733,000	268,513,000	3,980,000	20,333,000	344,408,000	368,721,000	588,000	109,168,000	109,756,000	17,141,000	495,618,000	764,131,000
4		DW	26 m	12,500,000	19,500,000	7,780,000	186,912,000	226,692,000	3,980,000	20,333,000	344,408,000	368,721,000	588,000	57,326,000	57,914,000	17,141,000	443,776,000	670,468,000
5		DW	18 m	12,500,000	19,500,000	7,780,000	152,222,000	192,002,000	3,980,000	20,333,000	344,408,000	368,721,000	588,000	27,864,000	28,452,000	17,141,000	414,314,000	606,316,000
6		SP&SCL	33,8 m	12,500,000	19,500,000	7,780,000	224,382,000	264,162,000	3,980,000	20,333,000	344,408,000	368,721,000	588,000	109,168,000	109,756,000	17,141,000	495,618,000	759,780,000
7		SP&SCL	26 m	12,500,000	19,500,000	7,780,000	183,446,000	223,226,000	3,980,000	20,333,000	344,408,000	368,721,000	588,000	57,326,000	57,914,000	17,141,000	443,776,000	667,002,000
8		SP&SCL	18 m	12,500,000	19,500,000	7,780,000	150,055,000	189,835,000	3,980,000	20,333,000	344,408,000	368,721,000	588,000	27,864,000	28,452,000	17,141,000	414,314,000	604,149,000
9	MDR, GR [3]	DW	33,8 m	12,500,000	19,500,000	7,780,000	223,518,000	263,298,000	3,980,000	22,617,000	344,408,000	371,005,000	588,000	19,962,000	20,550,000	17,141,000	408,696,000	671,994,000
10		DW	26 m	12,500,000	19,500,000	7,780,000	280,740,000	320,520,000	3,980,000	22,617,000	344,408,000	371,005,000	588,000	13,249,000	13,837,000	17,141,000	401,983,000	722,503,000
11		DW	18 m	12,500,000	19,500,000	7,780,000	283,667,000	323,447,000	3,980,000	22,617,000	344,408,000	371,005,000	588,000	8,186,000	8,774,000	17,141,000	396,920,000	720,367,000
12		SP&SCL	33,8 m	12,500,000	19,500,000	7,780,000	219,167,000	258,947,000	3,980,000	22,617,000	344,408,000	371,005,000	588,000	19,962,000	20,550,000	17,141,000	408,696,000	667,643,000
13		SP&SCL	26 m	12,500,000	19,500,000	7,780,000	277,274,000	317,054,000	3,980,000	22,617,000	344,408,000	371,005,000	588,000	13,249,000	13,837,000	17,141,000	401,983,000	719,037,000
14		SP&SCL	18 m	12,500,000	19,500,000	7,780,000	281,499,000	321,279,000	3,980,000	22,617,000	344,408,000	371,005,000	588,000	8,186,000	8,774,000	17,141,000	396,920,000	718,199,000
15	MDR, IR [3], [4]	DW	33,8 m	12,500,000	19,500,000	7,780,000	319,646,000	359,426,000	3,980,000	74,267,000	344,408,000	422,655,000	588,000	19,962,000	20,550,000	17,141,000	460,346,000	819,772,000
16		SP&SCL	33,8 m	12,500,000	19,500,000	7,780,000	315,295,000	355,075,000	3,980,000	74,267,000	344,408,000	422,655,000	588,000	19,962,000	20,550,000	17,141,000	460,346,000	815,421,000
17	MDR, CA	SP&SCL	–	12,500,000	19,500,000	7,780,000	192,293,000	232,073,000	3,980,000	22,617,000	344,408,000	371,005,000	588,000	13,973,000	14,561,000	17,141,000	402,707,000	634,780,000
18		SP&RO	–	12,500,000	19,500,000	7,780,000	91,611,000	131,391,000	3,980,000	22,617,000	344,408,000	371,005,000	588,000	13,973,000	14,561,000	17,141,000	402,707,000	534,098,000

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[2] DW: diaphragm wall, SP: secant piles, SCL: sprayed concrete lining, RO: rock

[3] The construction costs of all shaft based MDRs are based on the assumption that the bottom slab is located at the deepest depth possible determined by the structural capacity with given internal diameter and thickness of external walls. In particular, the diameter 33,8 m MDRs could be placed less deep, which would lead to a decrease in the construction costs.

[4] The diameter 26 m and 18 m MDR, IR concepts interfere with the minimum volume requirements and have thus been excluded from the cost estimate.

[5] The costs of the main repository include costs of cut-off structures and excavation.

Appendix J: Detailed activity plan

(In Danish)

J.1 Planlægning af det videre forløb

Som den første aktivitet laves en detaljeret planlægning af det videre forløb, hvor eventuelle udestående spørgsmål søges afklaret.

Dette kan kræve involvering af andre parter end rådgiver og bygherre, herunder f.eks. Naturstyrelsen vedrørende tilrettelæggelse af VVM - processen og stillingtagen til, hvordan et slutdepot forholder sig til nationale love og bekendtgørelser, herunder risikobekendtgørelsen, samt internationale konventioner, direktiver og traktater, herunder IAEA's konvention om sikkerhed i forhold til radioaktivt affald og EURATOM - traktaten om det europæiske atomenergifællesskab. Det skal understreges, at der ikke forventes nogen grænseoverskridende miljøpåvirkninger.

Det foreslås, at der udarbejdes en myndighedsplan, som skitserer, hvornår de enkelte myndigheder og aktiviteter skal indgå.

Det skal desuden afklares, hvorvidt VVM - undersøgelsen og den tilhørende skitseprojektering samt efterfølgende faser skal udbydes og i givet fald hvordan.

Myndighedsplan

I myndighedsplanen afklares det bl.a., hvornår og hvorledes EURATOM - traktatens artikel 37 håndteres i forhold til VVM - processen, herunder hvornår Kommissionen orienteres om eventuel påvirkning af nabolande:

Artikel 37:

"Hver medlemsstat skal forsyne Kommissionen med alle almindelige oplysninger vedrørende planer om bortskaffelse af radioaktivt spild i enhver form, for at det derved kan afgøres, om iværksættelsen af denne plan kan antages at medføre en radioaktiv kontaminering af en anden medlemsstats vande, jord eller luftrum.

Efter høring af den i artikel 31 omhandlede ekspertgruppe afgiver Kommissionen sin udtalelse inden for en frist af 6 måneder."

Udbud af VVM-undersøgelsen

Såfremt VVM-undersøgelsen udbydes til rådgiver, skal følgende overvejes og afklares:

- Valg af udbudsform
- Gennemførelse af selve udbuddet
- Bedømmelse af indkomne tilbud
- Valg af rådgiver
- Tildeling og overdragelse af opgaven til rådgiver
- Eventuel bygherrerådgivning/styring af VVM-rådgiver.

J.1.1 Udvælgelse af lokaliteter

Som et led i deres del af forstudiet udpeger GEUS ca. 20 potentielle lokaliteter for placering af slutdepot. Det videre mulige antal lokaliteter skal dernæst udpeges på grundlag af de samlede forstudier. Antallet antal vil om muligt blive begrænset til 5 - 6 lokaliteter.

J.1.2 Projekteringslov

Inden det forberedende arbejde for en anlægslov igangsættes, skal der fremsættes og vedtages et forslag om projekteringslov i Folketinget og afsættes finanser på finanslov.

Et slutdepot for radioaktivt affald er opført på VVM-bekendtgørelsens bilag 1, stk. 3b) *Anlæg, der er bestemt udelukkende til deponering (planlagt at vare mere end 10 år) af bestrålet nukleart brændsel eller radioaktivt affald på et andet sted end produktionsstedet.*

Da anlægget forudsættes vedtaget efter anlægslov, er det undtaget VVM-bekendtgørelsen. Det betyder, at miljøspørgsmålene vil blive adresseret i en anlægslov på tilsvarende niveau, som havde det fulgt VVM-bekendtgørelsen under planloven, herunder afholdelse af tilsvarende offentlighedsfaser etc.

Dansk Dekommissionering anmoder i samarbejde med SIS og GEUS den ansvarlige minister (Sundhedsministeren) om at fremsætte, behandle og få vedtaget forslag til projekteringslov i Folketinget, hvilket er en forudsætning for at igangsætte det lovforberedende arbejde, herunder VVM.

Parallelt med fremsættelse af projekteringslov i Folketinget foreslås det, at EU-kommissionen (og eventuelle andre internationale instanser) orienteres om, at planlægningen af slutdepotet vil blive igangsat efter lovens vedtagelse, at man ikke forventer nogen grænseoverskridende miljøpåvirkning, og at Kommissionen vil blive orienteret om resultatet af VVM-undersøgelsen.

J.1.3 VVM-proces

Det pågældende ressortministerium igangsætter VVM-undersøgelsen, når projekteringsloven er vedtaget.

Der nedsættes arbejdsgruppe, styregruppe og/eller følgegruppe med repræsentanter for de væsentligst berørte myndigheder og interessenter (kommunale og statslige).

Parallelt med/som en del af VVM-undersøgelsen gennemføres en skitseprojektering.

Indkaldelse af ideer og forslag

Udover at undgå utilsigtede miljøpåvirkninger er et af formålene med VVM at inddrage offentligheden ved to offentlige høringer.

I den første offentlige høring i en VVM-undersøgelse indkaldes ideer og forslag. Dette kan f.eks. gøres ved at udsende et debatoplæg, der samtidig orienterer om anlægget, de udvalgte lokaliteter, og om at planlægningen af dette nu er gået i gang.

Normalt afholdes borgermøder lokalt i de områder, der kan blive berørt af anlægget. Det skal overvejes, om man eventuelt skal holde dette mere overordnet, især hvis man opstarter processen på et tidligt niveau med mange mulige lokaliteter.

De indkomne ideer og forslag skal gennemgås og – hvis de er relevante – indgå i den videre VVM-undersøgelse. Man kan eksempelvis udarbejde en hvidbog med fuld gengivelse eller resume af alle indkomne ideer og forslag.

VVM-scoping

Når ideer og forslag er færdigbehandlet, skal omfanget af VVM-undersøgelsen endeligt vurderes, herunder hvilke metoder, der vil blive anvendt i undersøgelsen. Dette betegnes som en VVM-scoping.

Man kan gennemføre en VVM-scoping på et indledende niveau, f.eks. i forbindelse med planlægning og eventuelt udbud af VVM-undersøgelsen. Som alternativ kan man afvente indkaldelse af ideer og forslag, før man igangsætter undersøgelser og dermed kender omfanget i større detalje. Der kan under 1. offentlighedsfase fremkomme nye alternativer, som kræver yderligere undersøgelser.

I VVM-scopingen fastlægges (om muligt), hvilke alternativer, der indgår, herunder en klar definition af 0-alternativet (den situation, at der ikke tages beslutning om et slutdepot, og det mellem- og lavradioaktive affald må blive liggende på nuværende placering). Allerede i scopingen bør der skæres ned i antallet af alternativer, der kræver fuld VVM.

Influensområdet afgrænses geografisk, og der identificeres interessenter i en interessentanalyse.

VVM-undersøgelse

I VVM-undersøgelsen indgår dels en undersøgelse af de eksisterende miljøforhold, dels en vurdering af anlæggets påvirkninger. Såfremt der stadig indgår 20 lokaliteter (og ikke 5 - 6 som vil være mest hensigtsmæssigt), vil første del af VVM-undersøgelsen indeholde en miljømæssig screening af disse og fravalg af lokaliteter med største miljøkonsekvenser.

Når der er indhentet detaljerede oplysninger om de eksisterende miljøforhold, kan der foretages en yderligere udvælgelse af de alternativer (placering og depotform), der umiddelbart ser ud til at have færrest konflikter med omgivelserne. Dette suppleres med oplysninger om f.eks. anlægsøkonomioverslag.

I undersøgelsen af eksisterende forhold indgår detaljerede feltundersøgelser af miljøforhold. Desuden indsamles øvrige oplysninger, og der udføres detaljerede beregninger og vurderinger af konsekvenserne af anlægget inden for alle miljøemner (se neden for).

På denne baggrund sammenlignes alternativer, hovedforslag vælges, og der opstilles afværgeforanstaltninger.

Når der er gennemført de første vurderinger og beregninger af miljøpåvirkningerne, kan der foretages en yderligere udvælgelse af de alternativer (placering og depotform), der umiddelbart ser ud til at få de færrest konsekvenser for omgivelserne. Dette suppleres med oplysninger om f.eks. anlægsøkonomioverslag.

VVM-redegørelse

Selve afrapporteringen i en VVM-redegørelse vil som minimum omfatte:

- En beskrivelse af det påtænkte anlæg
- En oversigt over de væsentligste alternativer (herunder 0-alternativet) og en begrundelse for valg og fravalg af alternativer under hensyn til påvirkninger af miljøet
- En beskrivelse af de metoder og principper, der er anvendt i VVM-undersøgelsen
- En beskrivelse af de eksisterende miljøforhold i det område der kan blive påvirket af anlægget
- En beskrivelse af de kort- og langsigtede miljøpåvirkninger
- En beskrivelse af foranstaltninger, der indarbejdes i projektet for at skadelige miljøpåvirkninger undgås, mindskes, eller der kompenseres for dem
- En overvågningsplan for udvalgte miljøfaktorer (er ikke krævet i VVM, men i miljøvurdering)
- En oversigt over eventuelle mangler ved oplysningerne eller vurderingerne i VVM-undersøgelsen, inklusive en vurdering af manglernes betydning for konklusionerne i VVM-redegørelsen.

De miljøforhold, der vil indgå, omfatter virkninger under anlæg og drift for:

- Planforhold
- Landskab og jordbund
- Plante- og dyreliv, herunder eventuelle påvirkninger af Natura 2000-områder og arter opført på habitatdirektivets bilag IV eller fuglebeskyttelsesdirektivets bilag I. Såfremt der kan være en potentiel påvirkning ind i et natura 2000 område, skal der indgå en separat Natura 2000 konsekvensvurdering
- Kulturmiljø og materielle goder
- Overfladevand og grundvand, herunder dybe, geologiske boringer
- Luft og klima
- Støj og vibrationer
- Befolkning, herunder beboelse, erhverv, friluftsliv og menneskers sundhed (sidstnævnte er ikke krævet i VVM, men i miljøvurdering)
- Afledte socioøkonomiske effekter.

Indkaldelse af bemærkninger og indsigelser

Når VVM-redegørelsen er færdig, udsendes denne i den 2. offentlige høring, hvor borgere og interessenter kan komme med indsigelser og bemærkninger til redegørelsen.

Bemærkninger og indsigelser skal behandles færdigt, inden der kan indstilles et hovedforslag til anlægslovforslag.

J.1.4 Forslag til anlægslov

Der skal udarbejdes forslag til anlægslov, som skal behandles og vedtages af Folketinget. Forinden skal opdragsholderen/den ansvarlige styrelse/myndighed indstille den løsning, de foretrækker (placering og depotform), samt begrundelser herfor til det ansvarshavende ministerium.

J.1.5 Skitseprojektering

Parallelt med VVM-undersøgelsen foretages en skitseprojektering af slutdepotet i en iterativ proces, således at det er et miljøoptimeret anlæg, der vedtages og detailprojekteres. Det betyder, at miljøundersøgelserne kan få indflydelse på anlæggets udformning, etablering og drift, såfremt nogle af aktiviteterne vurderes at kunne få væsentlig indflydelse på miljøet, og det af den årsag vil være hensigtsmæssigt at ændre aktiviteterne med henblik på at mindske miljøpåvirkningerne.

Desuden foretages der løbende justeringer af anlægsøkonomioverslag, herunder priser for de indarbejdede afværgeforanstaltninger og overvågningstiltag samt den nødvendige arealerhvervelse.

J.1.6 Vedtagelse af anlægslov

Når anlægsloven er vedtaget, kan detailprojektering, udbud og arealerhvervelse påbegyndes.

J.1.7 Detailprojektering og udbud

Når anlægget er vedtaget, skal der gennemføres en detailprojektering. Som en del af dette kan der udarbejdes et udbudsmateriale, og projektet kan udbydes, og anlægsentreprenør udvælges.

J.1.8 Arealerhvervelse

Parallelt med og på basis af detailprojekteringen kan ekspropriationsforretning og arealerhvervelse påbegyndes.

J.1.9 Udførelse

Herefter kan anlægsarbejdet påbegyndes.

J.1.10 Ibrugtagning, drift og vedligeholdelse af slutdepotet

Efter deponering af affaldet fra anlæggene på Risø skal der i en årrække være adgang til slutdepotet, idet der til stadighed genereres mindre mængder nukleart affald i Danmark. Det er i de øvrige dele af forstudierne forudsat, at slutdepotet drives i en 30-årig periode, hvorefter det lukkes for modtagelse af affald.

Det skal afklares, om depotet skal være bemandet, om der skal være adgang, publikumsfaciliteter etc.

Desuden skal almindeligt og ekstraordinært vedligehold aftales, herunder beredskabsplan for, hvis der opstår utætheder/tæringer etc.

J.1.11 Overvågningsplan

Der vil blive etableret en plan for overvågning af udvalgte miljøparametre i ca. 30 år. Herefter vil der blive foretaget en fornyet evaluering af overvågningsbehovet og -niveauet i de næste 30 år el. lign.

I overvågningsplanen skal indgå:

- Miljøparameteren, herunder hvorfor den bør overvåges
- Hvorledes overvågningen foretages
- Hvor hyppigt overvågningen foretages
- Hvem der er ansvarlig for overvågningen (udførende myndighed)
- Hvem der kontrollerer, at overvågningen er foretaget (tilsynsførende myndighed)
- Hvad der skal gøres, hvis overvågningen viser noget andet, end miljøvurderingen forudså, eller der optræder andre uregelmæssigheder, herunder etablering af en beredskabsplan ved uheld etc.

I beslutningsgrundlaget er anført, at *Det bør tilstræbes at deponering af radioaktivt affald ikke baseres på, at kommende generationer skal udføre sikkerhedsprocedurer og monitering.*

J.2 Økonomisk overslag

Der er i Tabel J.1 angivet omtrentlige udgifter til VVM-processen. Overslaget er blandt andet baseret på den viden, der er p.t., samt erfaringer fra andre VVM-undersøgelser, herunder fra VVM af det danske tilslutnings- og rampeanlæg for en fast Femern Bælt forbindelse.

Forslaget til feltundersøgelser er baseret på, at depoterne ikke vil blive placeret i OSD områderne, som er de mest undersøgte. Det er vurderet, at de indledende screeningsundersøgelser vil omfatte indledende geofysik, 5 boringer til relevant dybde, vandprøveudtagning og pejlinger, geokemiske undersøgelser og vurderinger, opsætning og kalibrering af en grundvandsmodel til sammenligning med forstudiets resultater samt rapportering. For én lokalitet vurderes omkostningerne til dette at ligge omkring 2,5 mio. DKK.

Når færre lokaliteter er udpeget og der skal træffes et konkret valg, skal der foretages mere detaljerede undersøgelser og sikkerhedsanalyser. Dette er vurderet at omfatte yderligere 3 boringer, tracer test, prøvepumpninger og slug test, yderligere geokemiske undersøgelser på bl.a. intakt prøver, detaljeret modellering, detaljeret sikkerhedsanalyse smat rapportering. For én lokalitet skønnes omkostnignerne hertil at ligge omkring 5 mio. DKK.

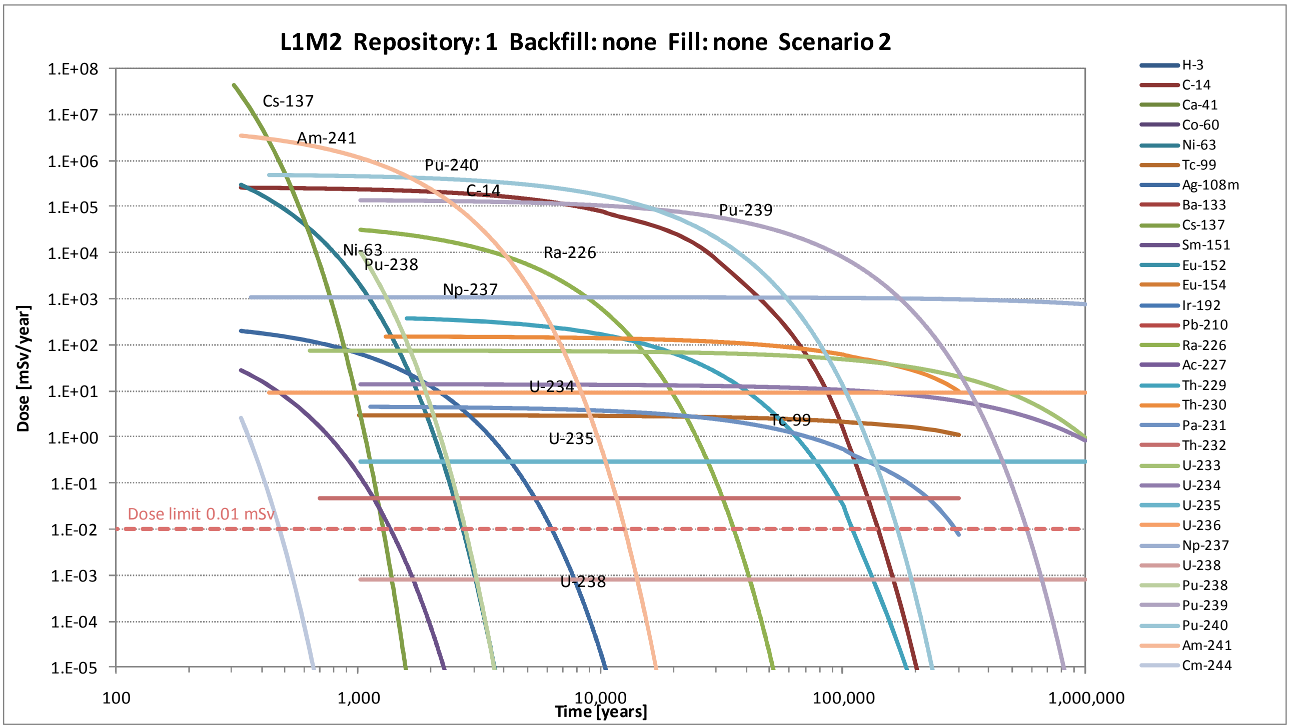
I Tabel J.1 er det forudsat, at der foretages indledende undersøgelser på 5 til 6 lokaliteter og detaljerede undersøgelser på 2 til 3 lokaliteter.

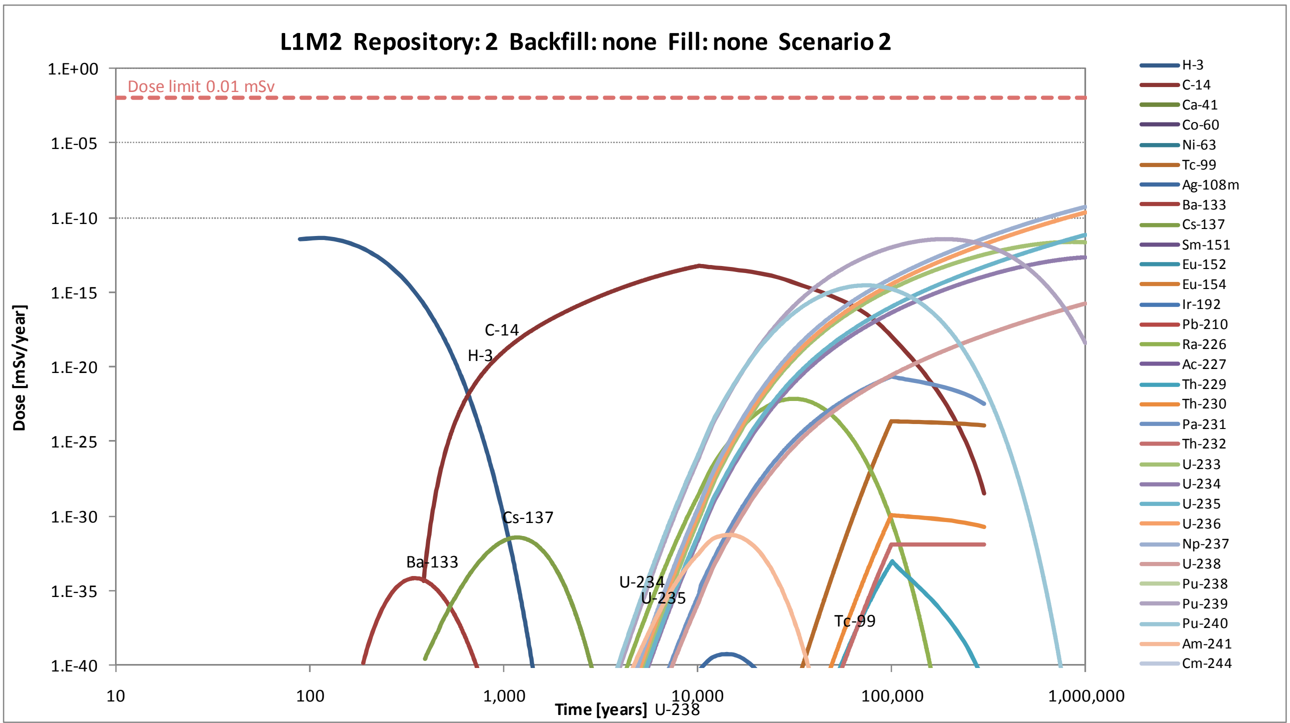
Tabel J.1 Økonomisk overslag for den skitserede aktivitetsplan frem til etablering af et depot

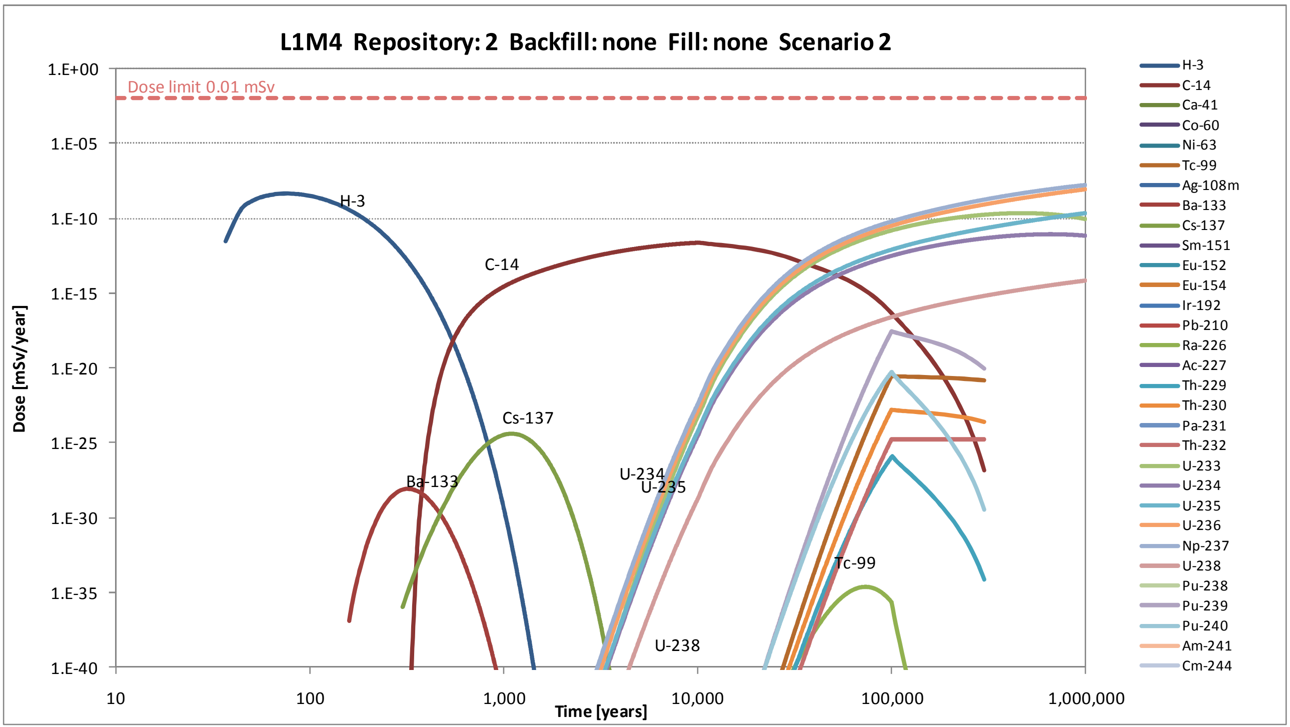
Aktivitet	**Overslag, DKK**
Planlægning af det videre forløb	300.000 - 500.000
Udpegning af mulige lokaliteter	22,5 - 30 mio.
Projekteringslov	500.000*
VVM-proces og skitseprojektering	2 - 5 mio.
Forslag til og vedtagelse af anlægslov	1 mio. *
Detailprojektering og udbud	1 - 2 mio. *
I alt	27 - 39 mio.

* Overslaget er behæftet med usikkerhed – for de med stjerne markerede felter er usikkerheden betydelig.

Appendix K: Results of dose calculations



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Dose [mSv/year]
Time [years]
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1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
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100,000
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H-3
C-14
Cs-137
Ba-133
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
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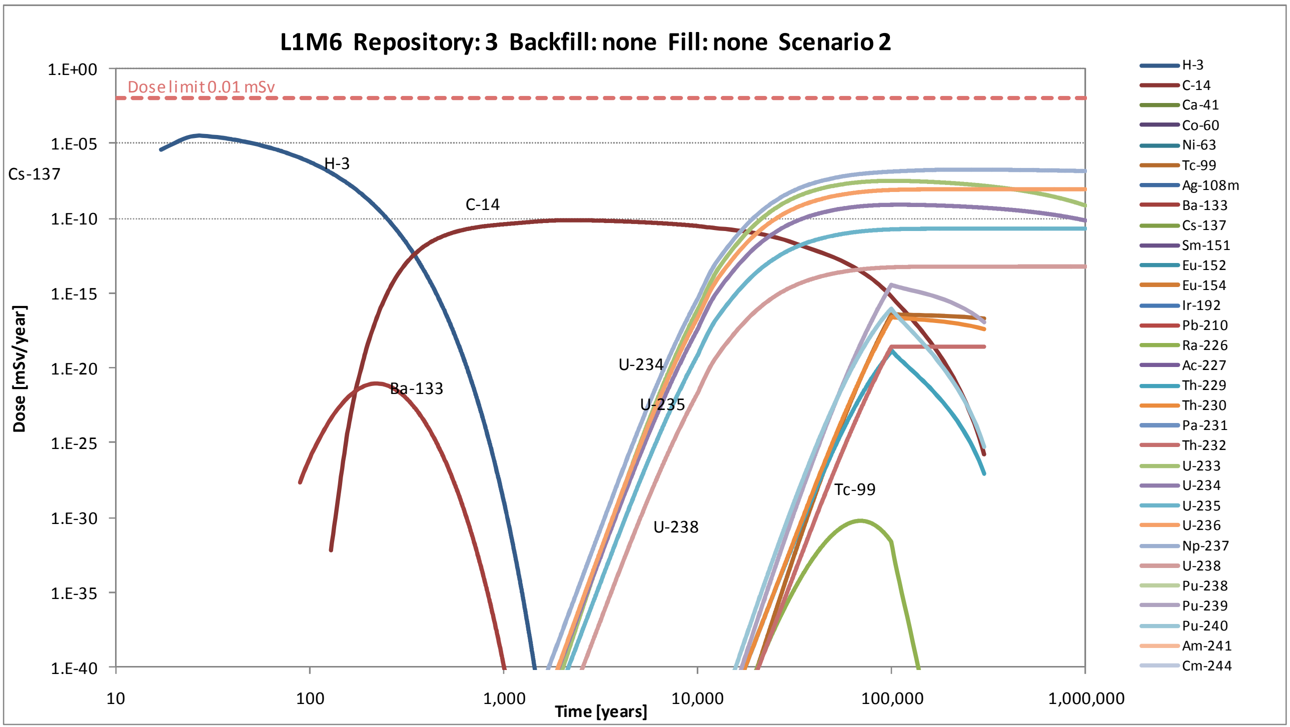
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244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Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
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Cs-137
H-3
C-14
Ba-133
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
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U-236
Np-237
U-238
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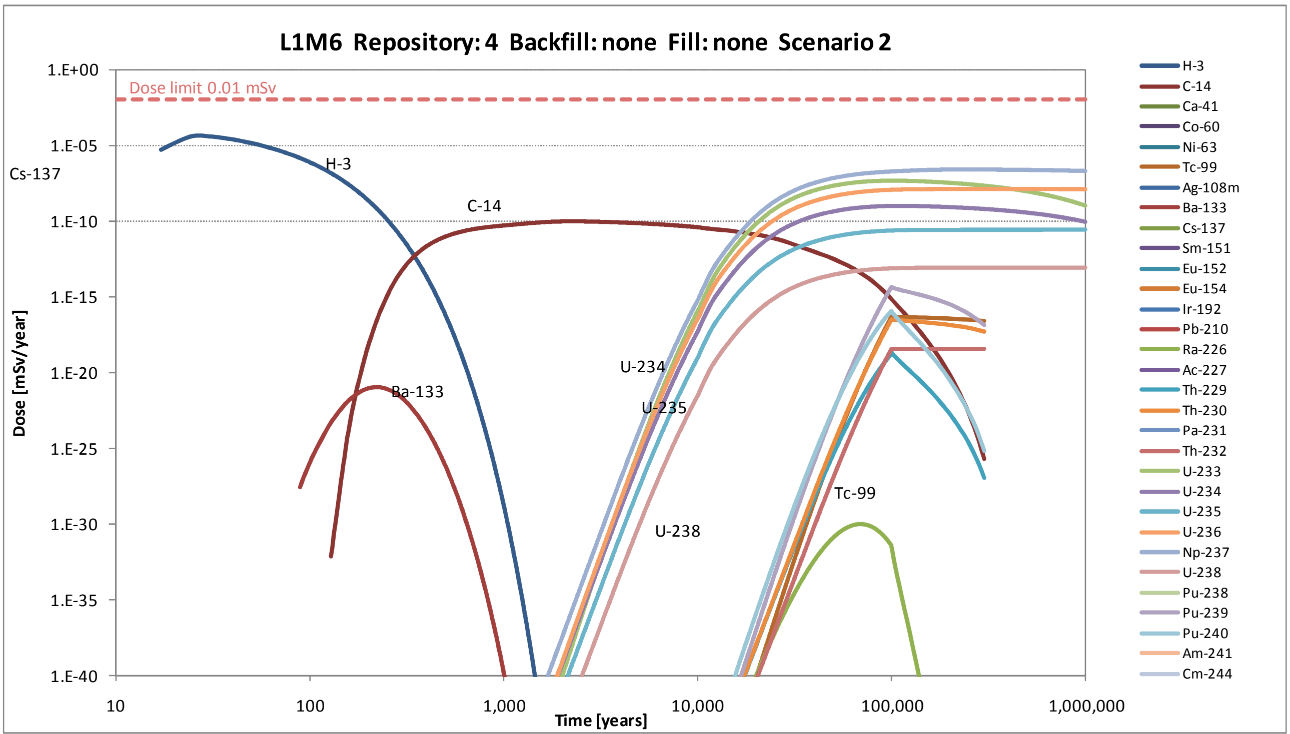
Pu-239
Pu-240
Am-241
Cm-244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Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
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Cs-137
H-3
C-14
Ba-133
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
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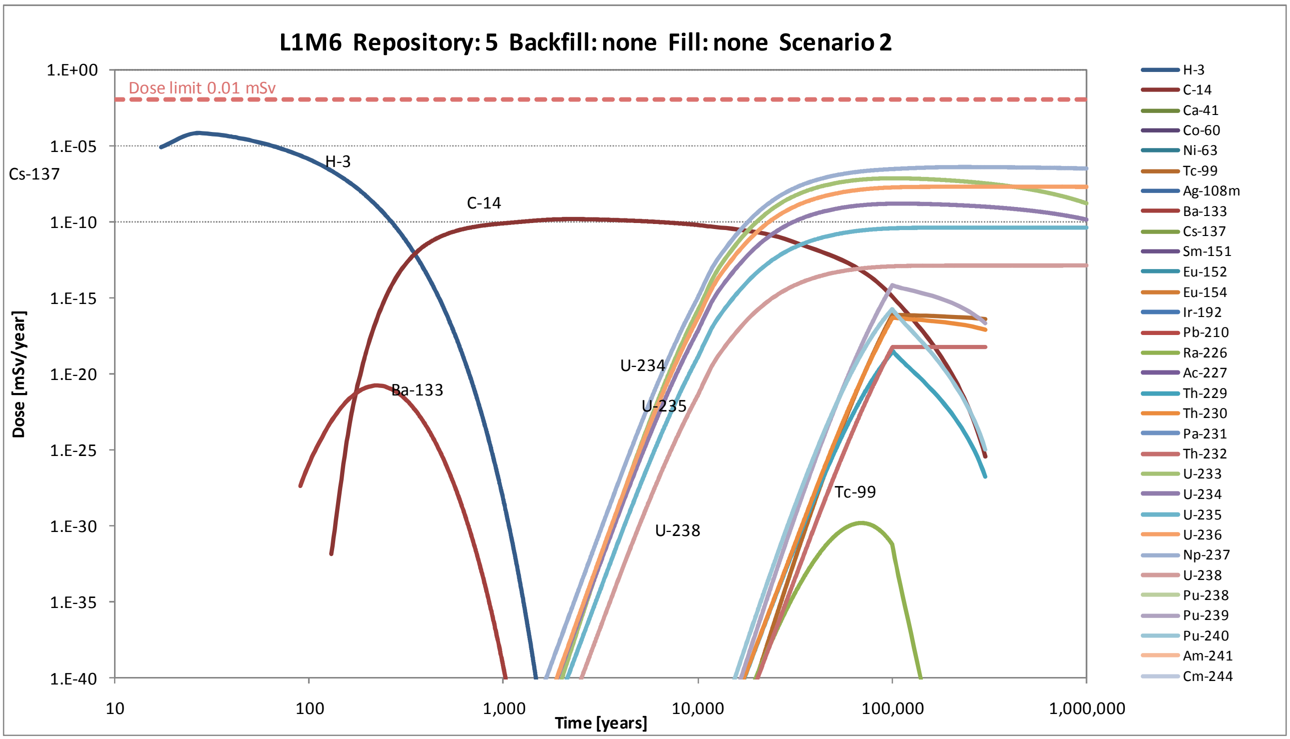
Pu-238
Pu-239
Pu-240
Am-241
Cm-244



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Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
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1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
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Cs-137
H-3
C-14
Ba-133
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
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Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
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Np-237
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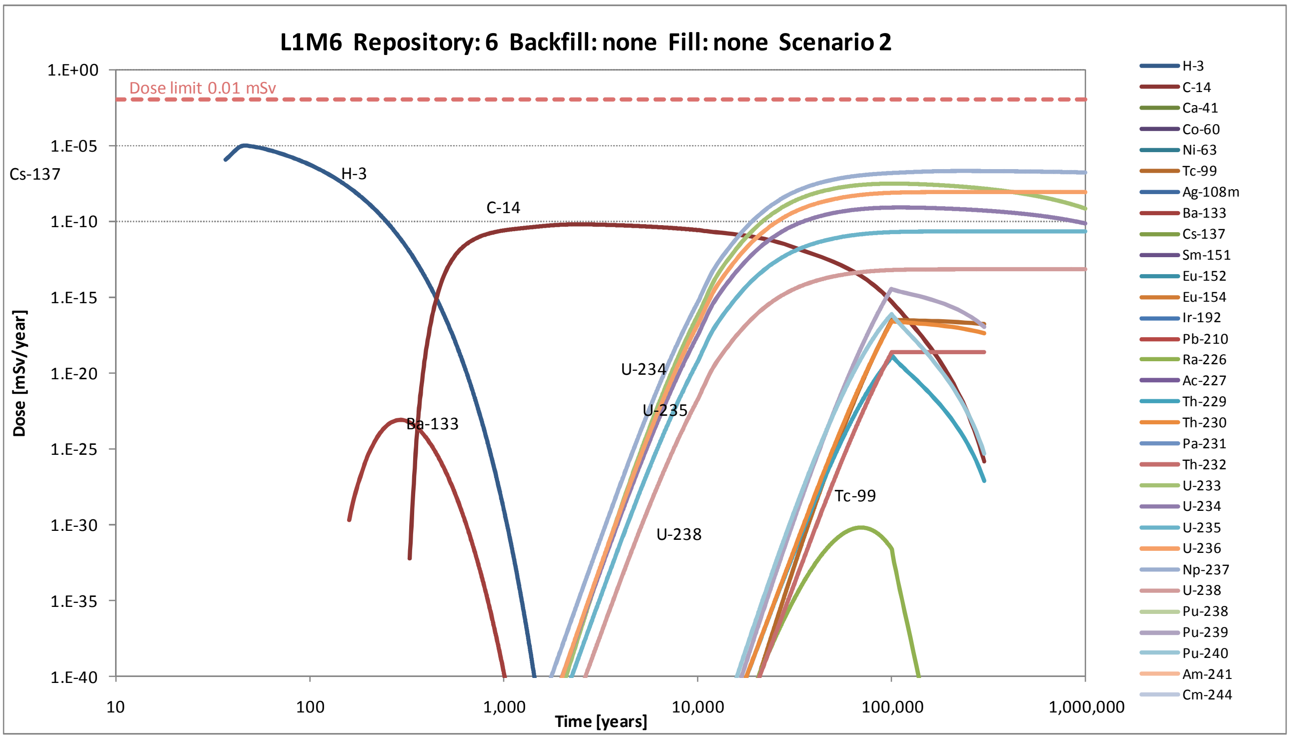
Pu-239
Pu-240
Am-241
Cm-244



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Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
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1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
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Cs-137
H-3
C-14
Ba-133
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
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Th-230
Pa-231
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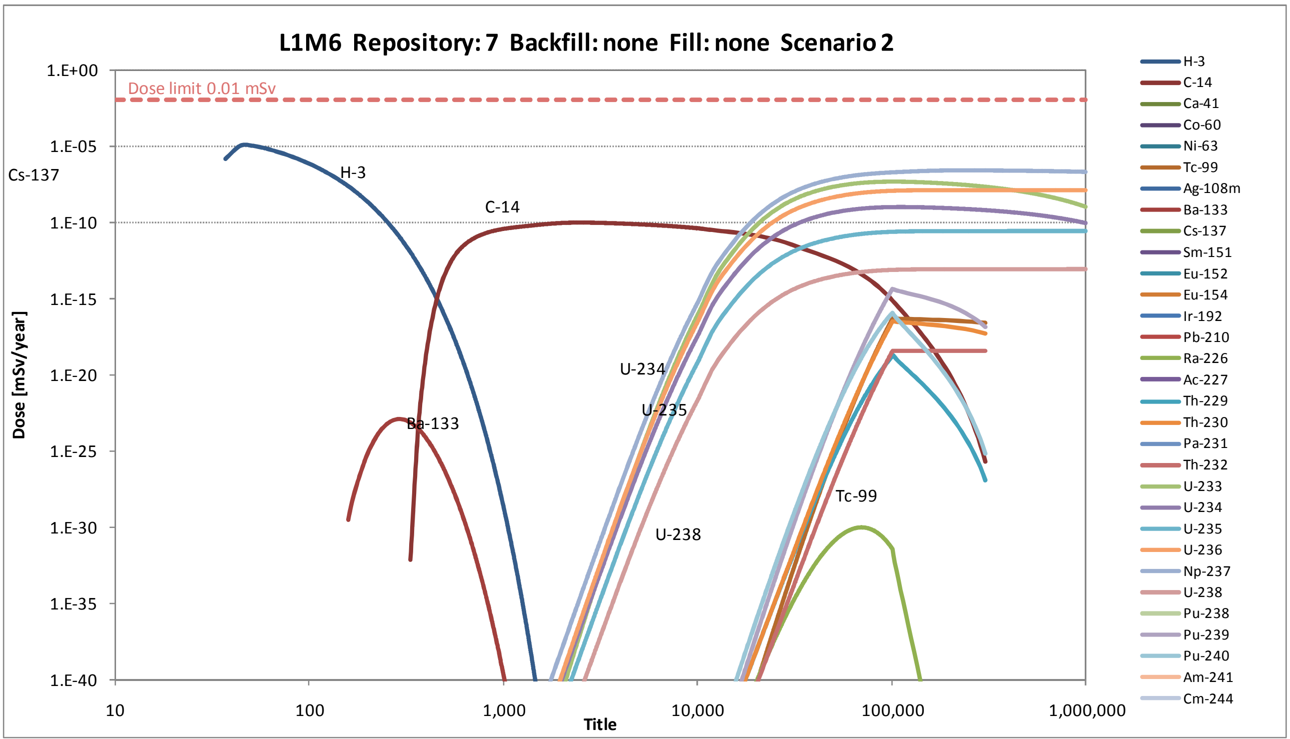
Pu-238
Pu-239
Pu-240
Am-241
Cm-244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Dose [mSv/year]
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1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
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Ba-133
U-234
U-235
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Tc-99
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C-14
Ca-41
Co-60
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Tc-99
Ag-108m
Ba-133
Cs-137
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Eu-152
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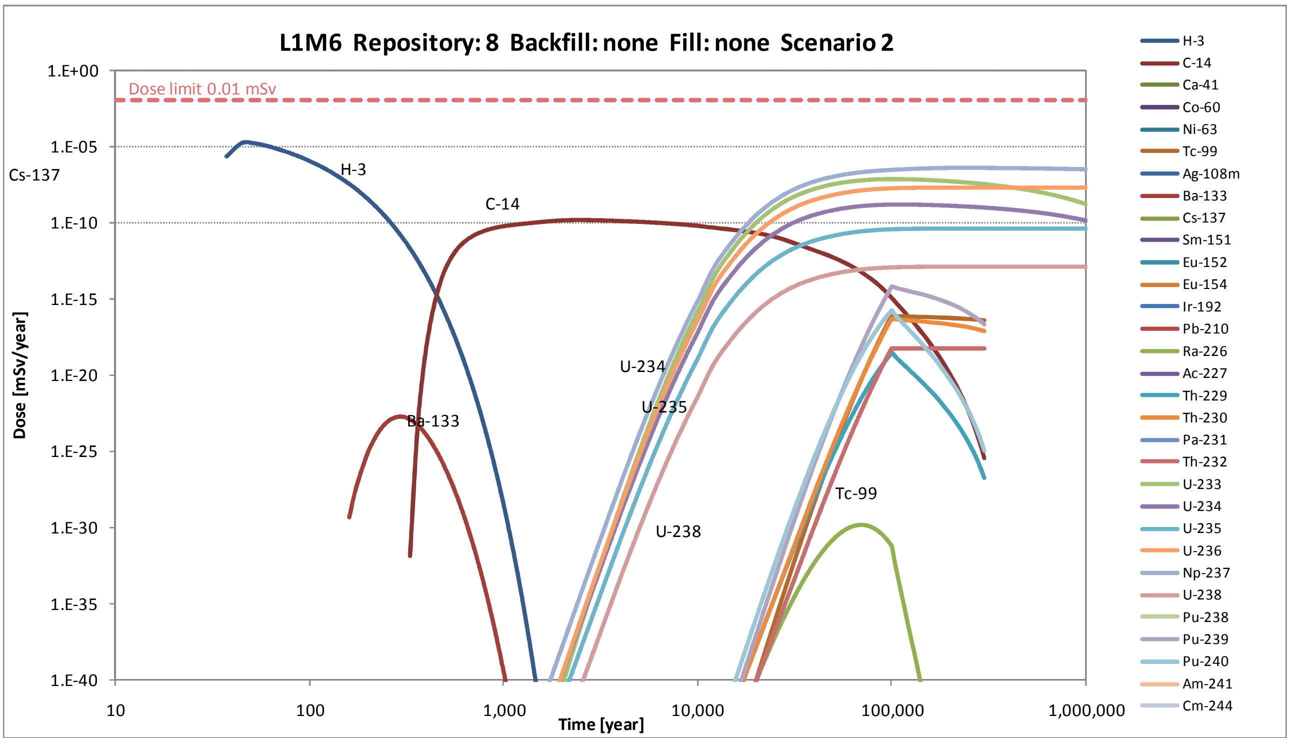
Cm-244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Dose [mSv/year]
Time [year]
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1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
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U-234
U-235
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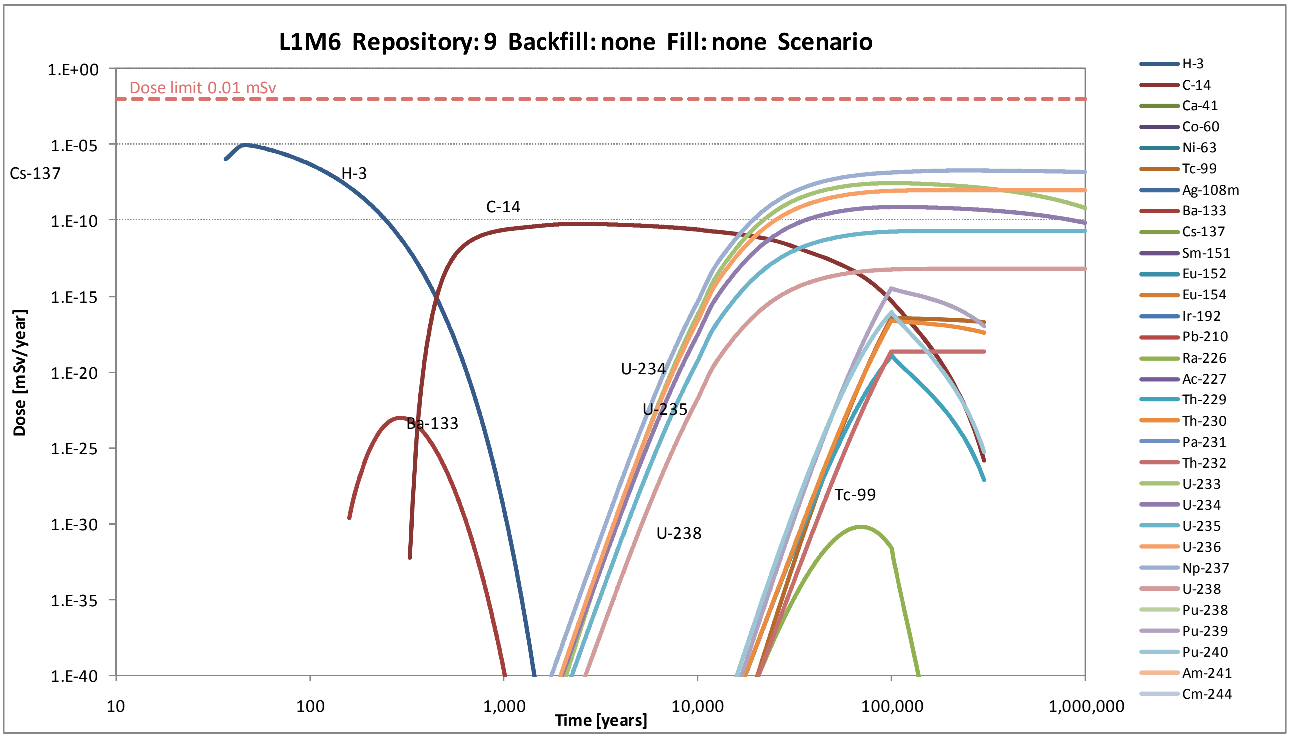
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Time [years]
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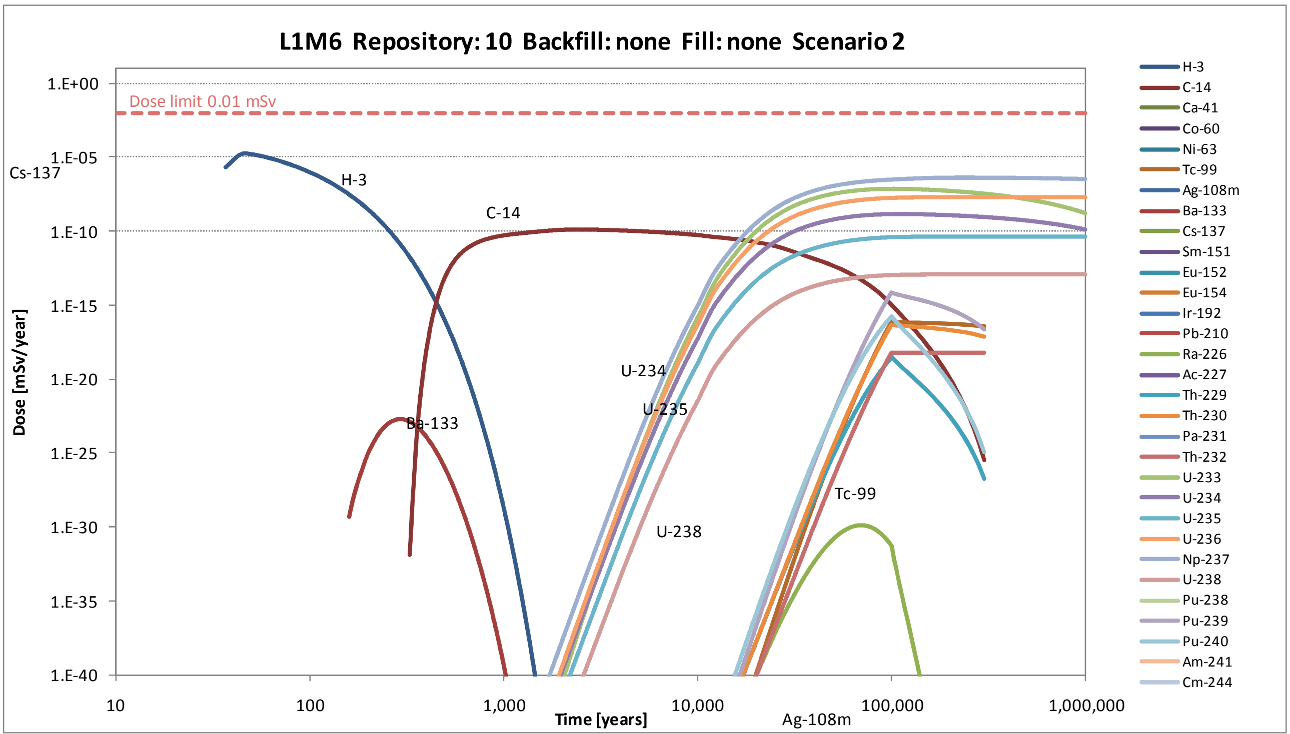
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Time [years]
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U-235
U-238
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Ag-108m
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Co-60
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Tc-99
Ag-108m
Ba-133
Cs-137
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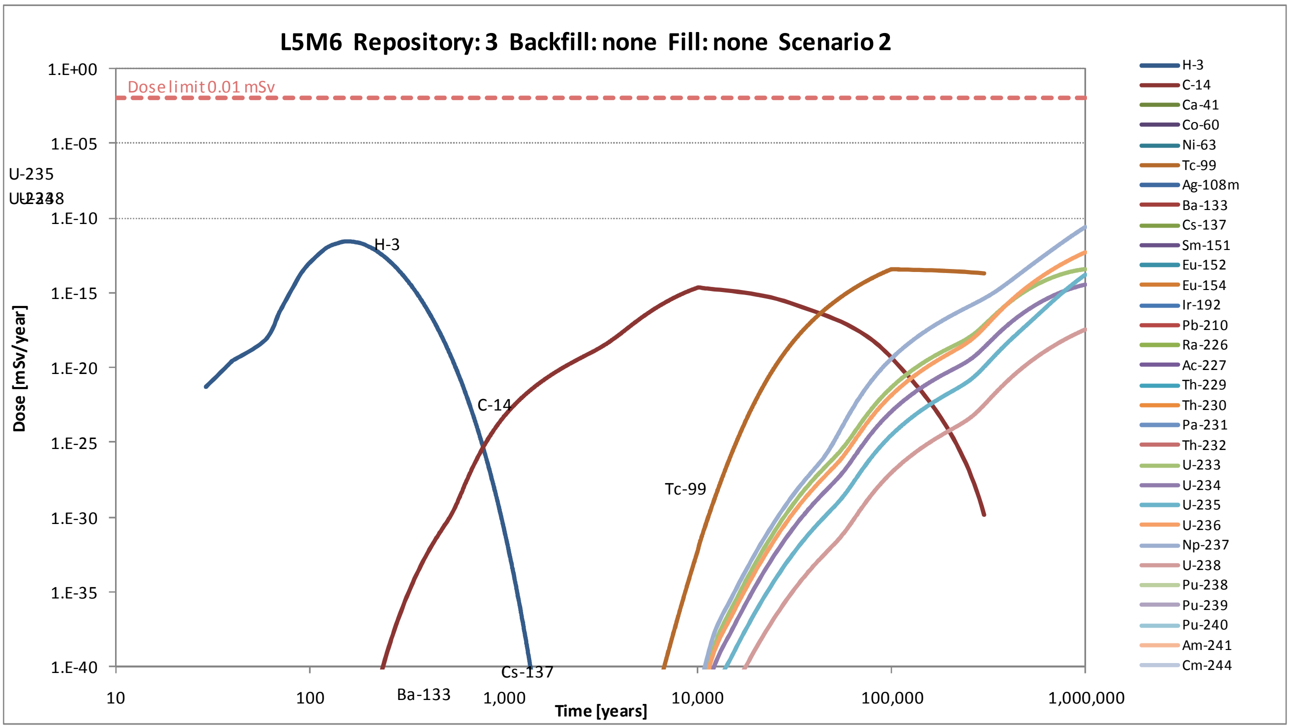
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Time [years]
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1.E-35
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Ca-41
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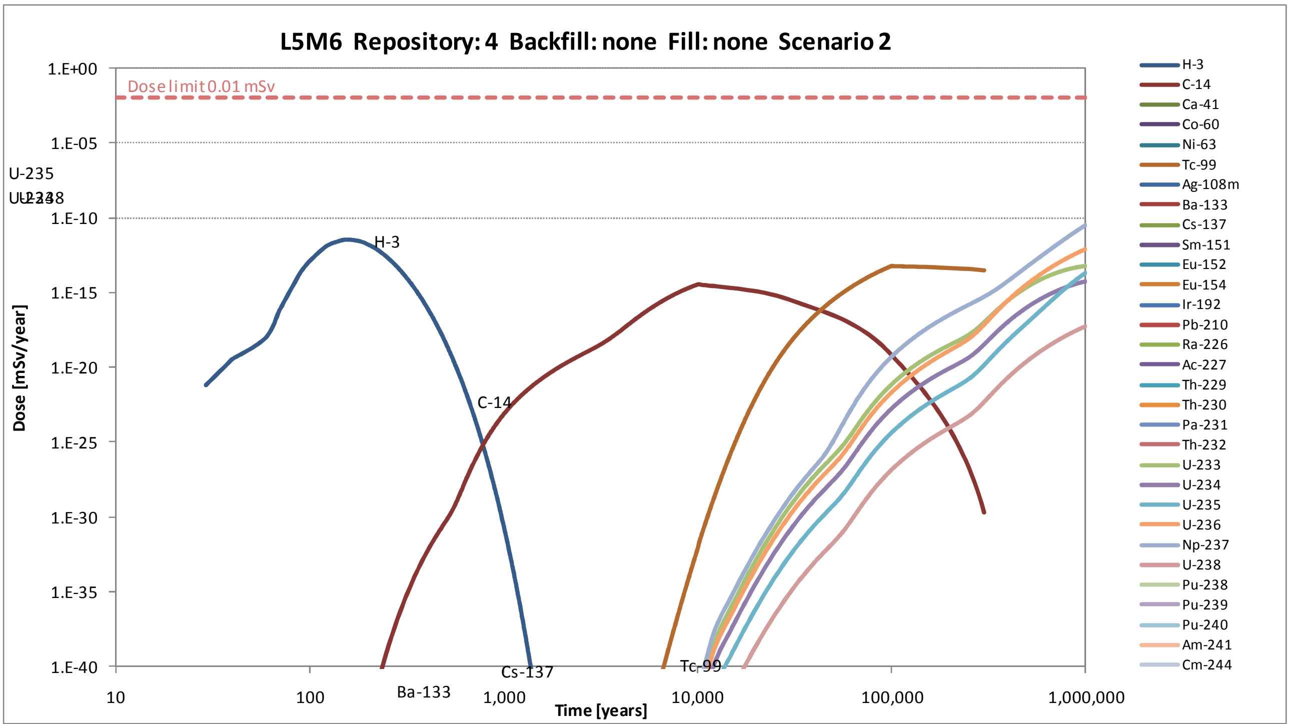
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Dose [mSv/year]
Time [years]
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1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
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C-14
Tc-99
Cs-137
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Co-60
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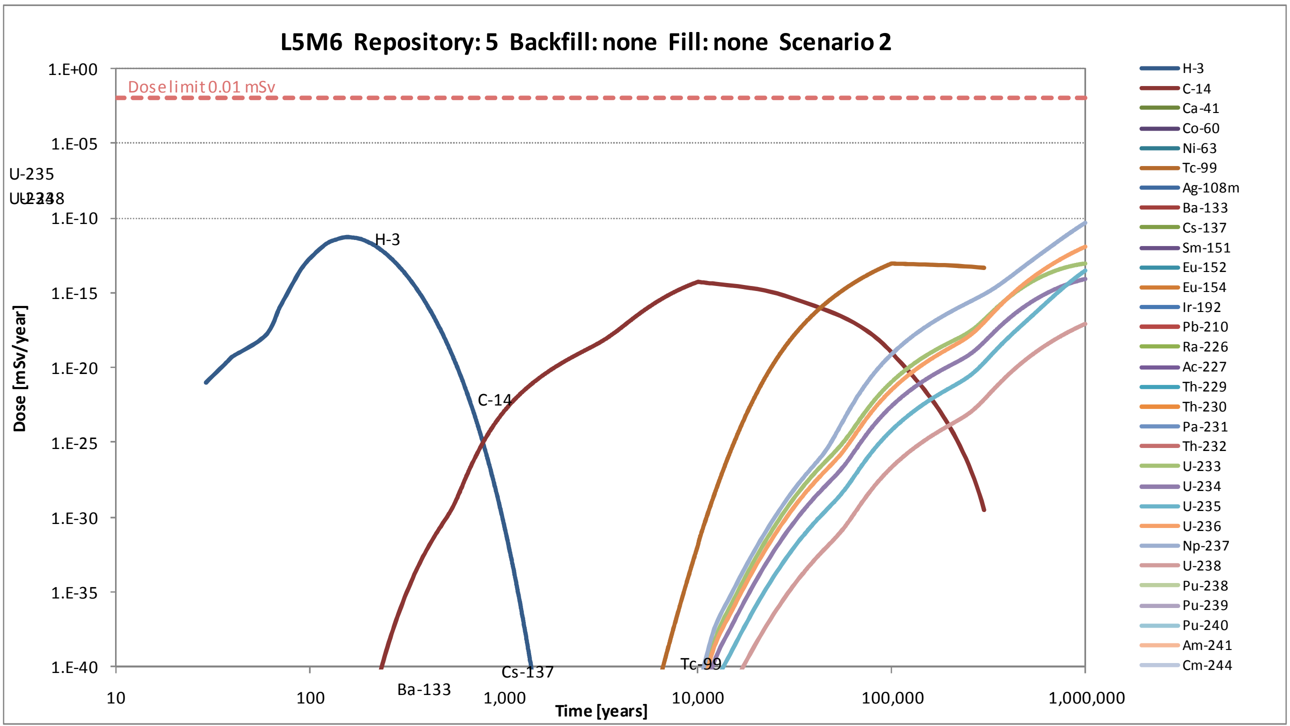
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Dose [mSv/year]
Time [years]
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1.E-15
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Cs-137
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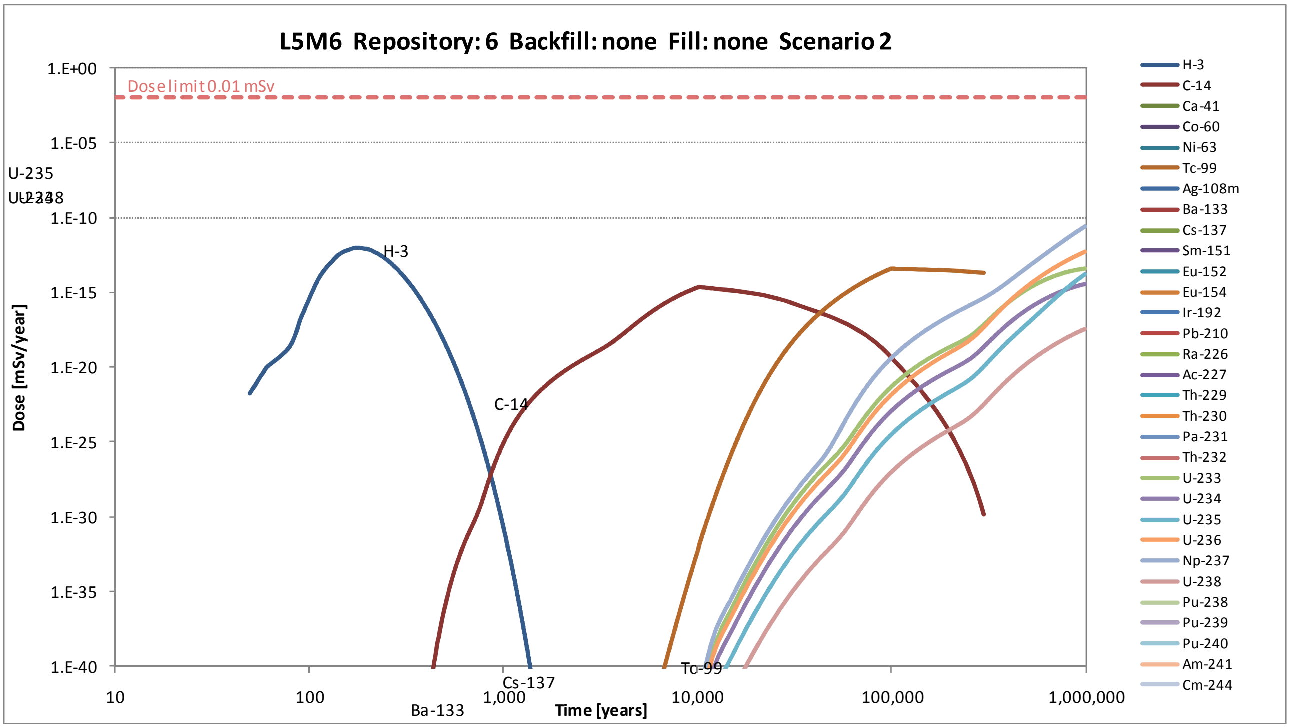
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Dose [mSv/year]
Time [years]
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1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
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H-3
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Cs-137
Ba-133
Tc-99
U-235
U-238
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
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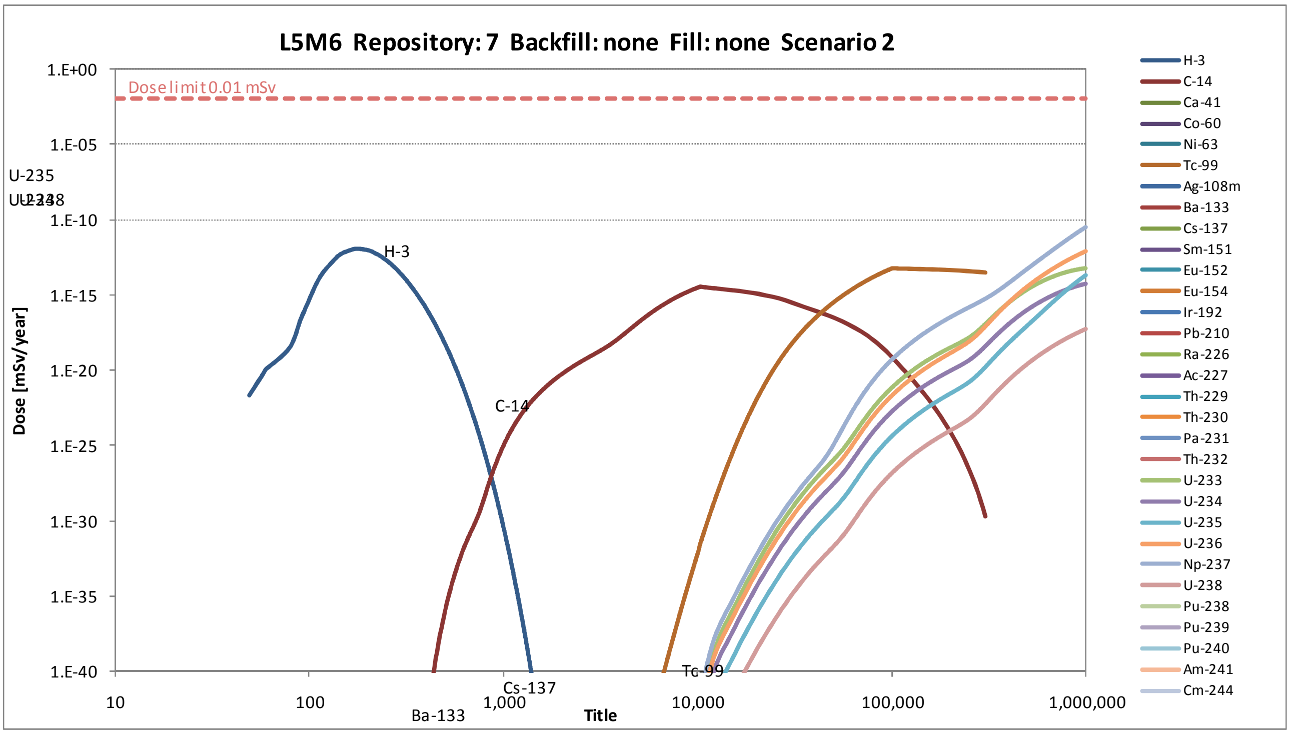
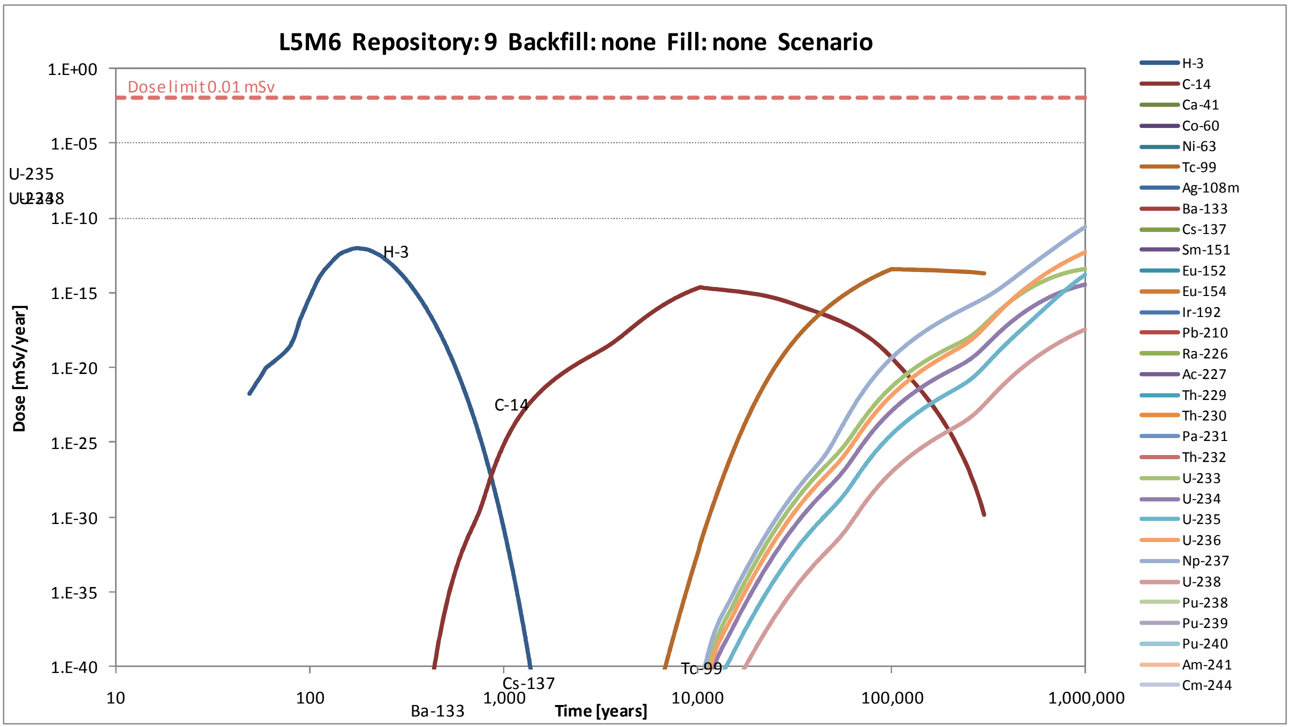
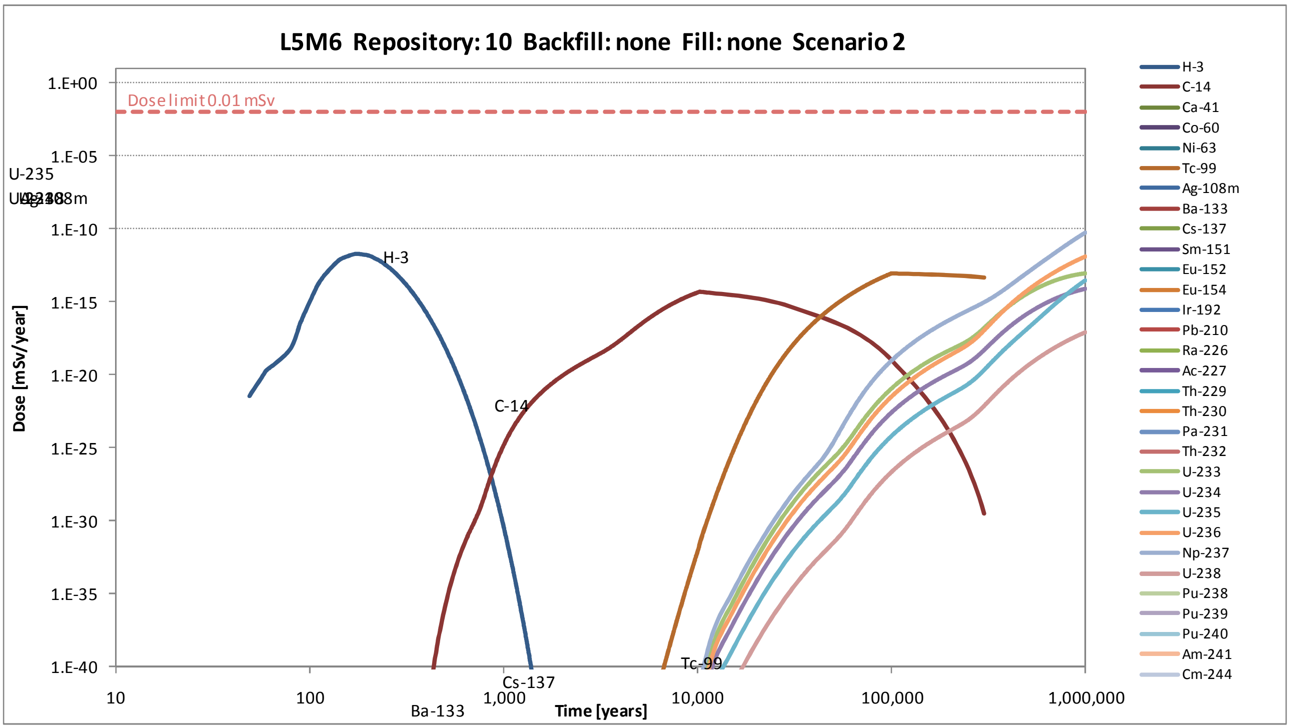
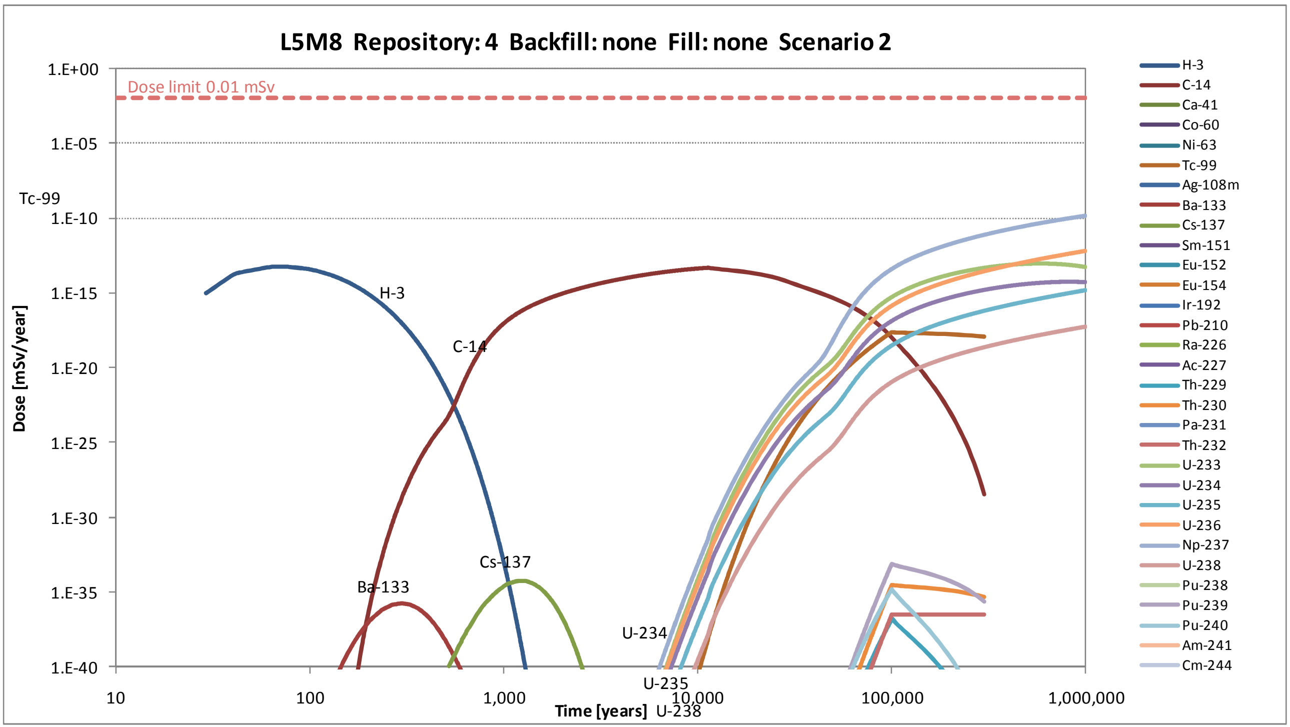
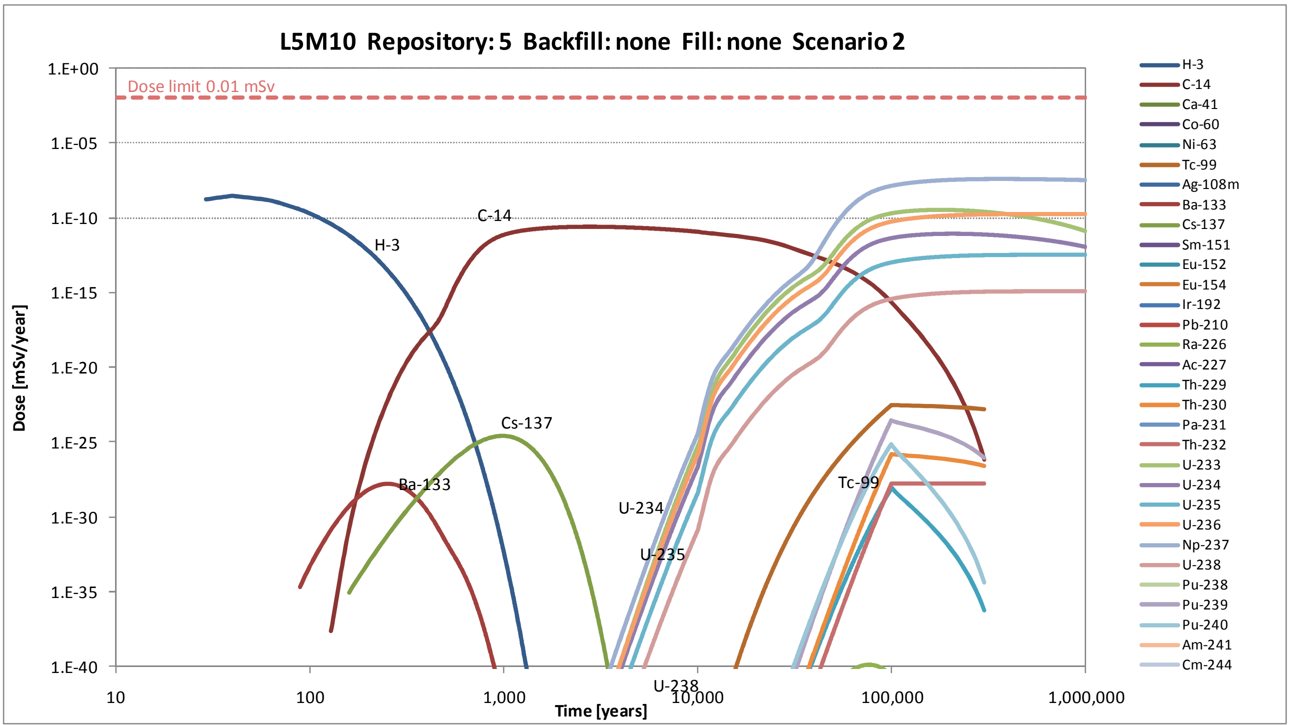
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

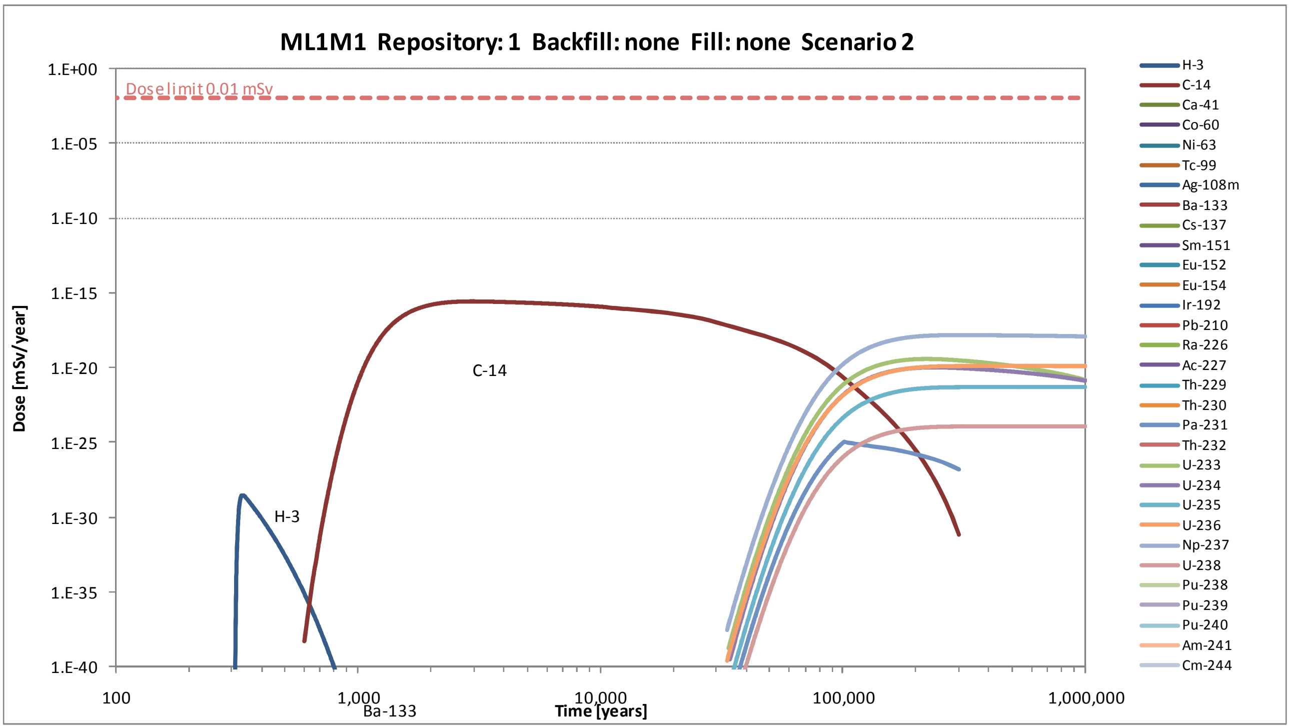

L5M6 Repository: 7 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
Title
H-3
C-14
Tc-99
Cs-137
Ba-133
U-235
U-238
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

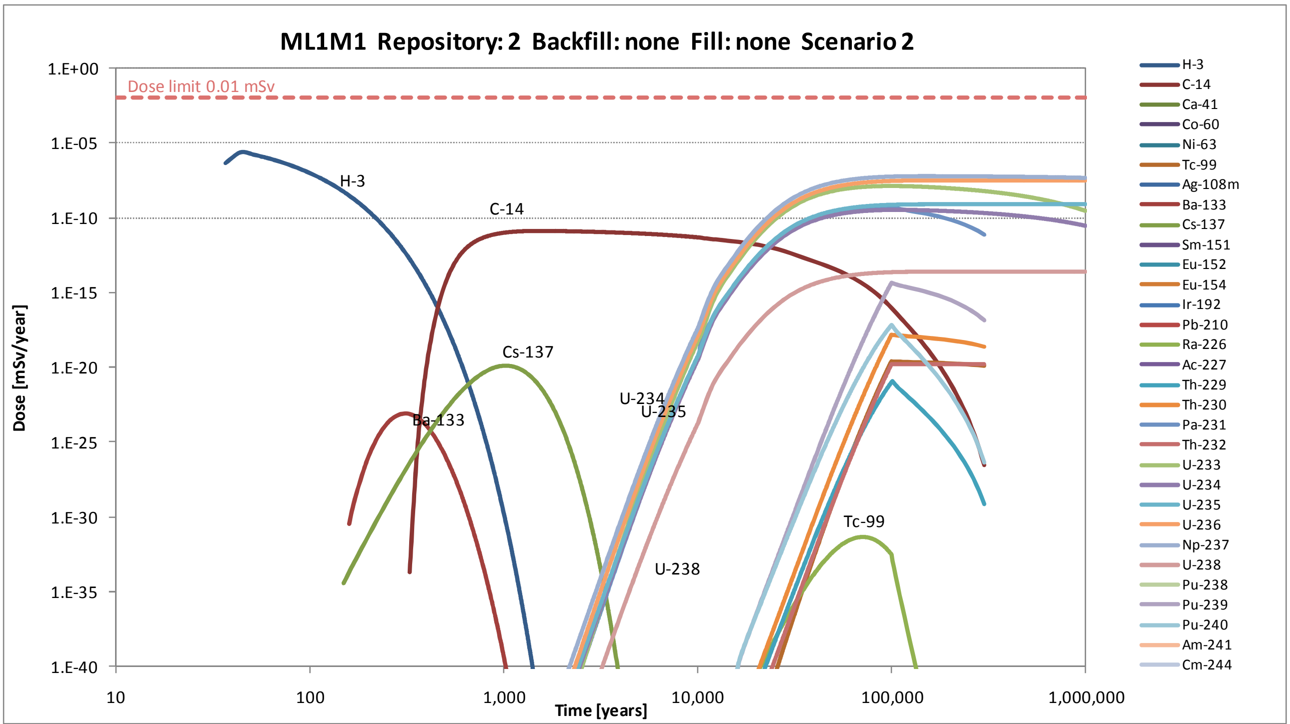

L5M6 Repository: 9 Backfill: none Fill: none Scenario
Dose limit 0.01 mSv
Dose [mSv/year]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
Time [years]
H-3
C-14
Tc-99
Cs-137
Ba-133
U-235
U-238
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

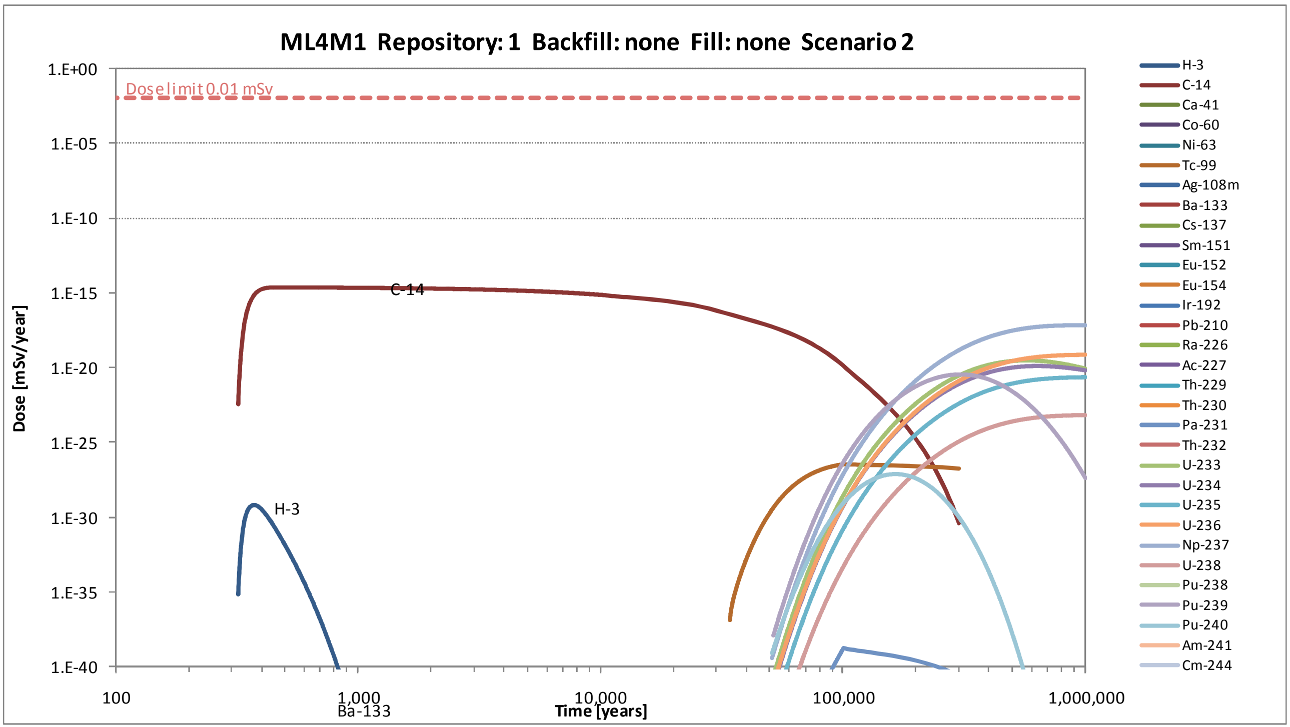

L5M6 Repository: 10 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Tc-99
Cs-137
Ba-133
U-235
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

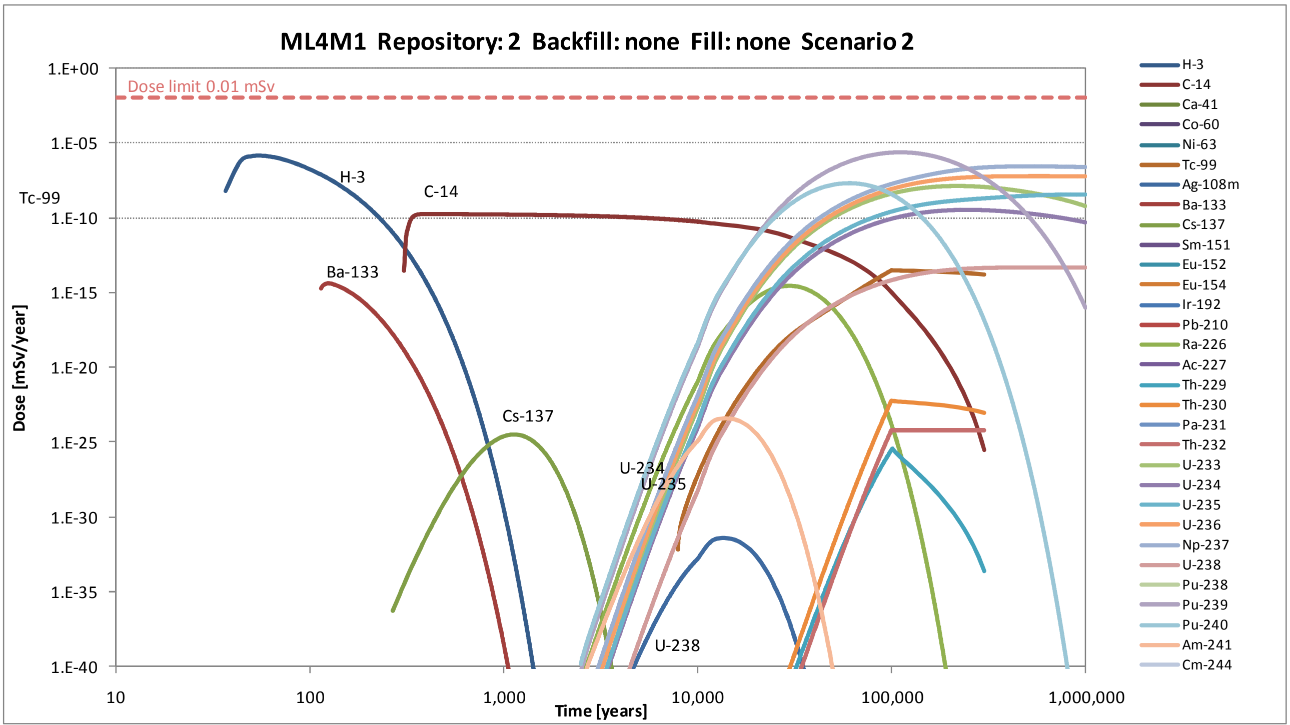

L5M8 Repository: 4 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
Tc-99
H-3
C-14
Cs-137
Ba-133
U-234
U-235
U-238
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

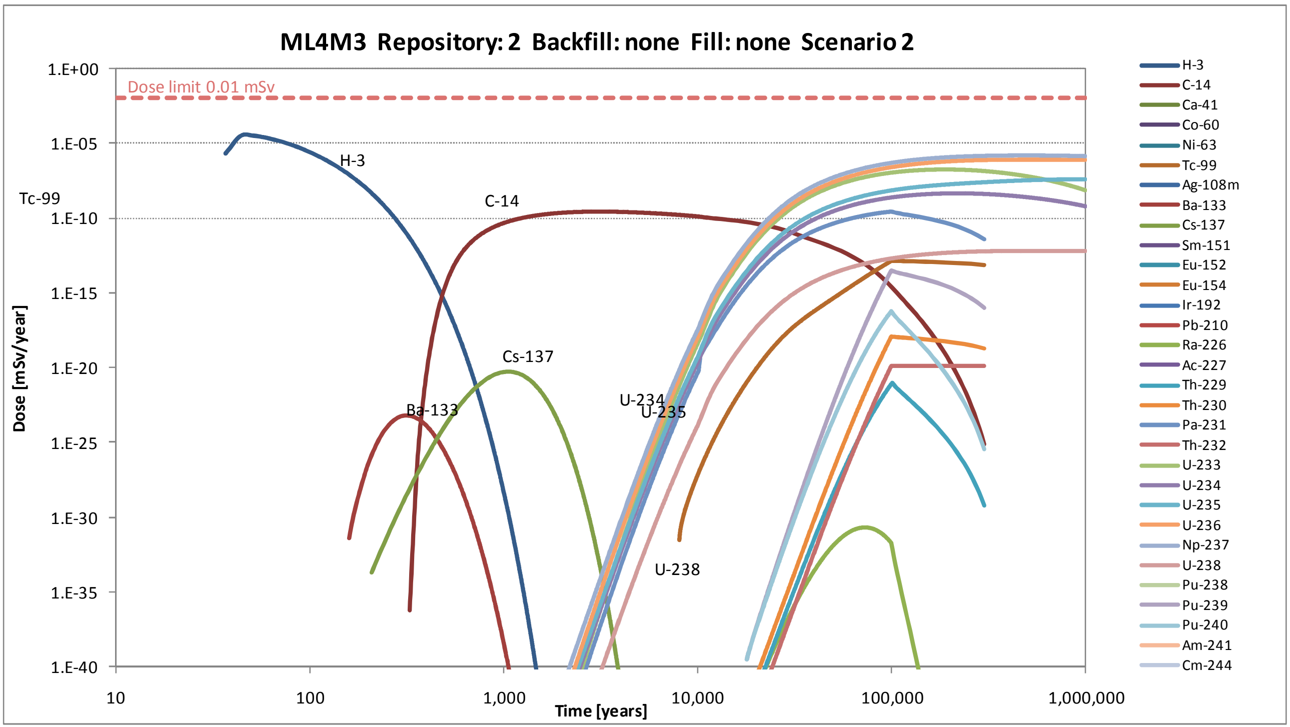

L5M10 Repository: 5 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
C-14
H-3
Cs-137
Ba-133
U-234
U-235
Tc-99
U-238
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

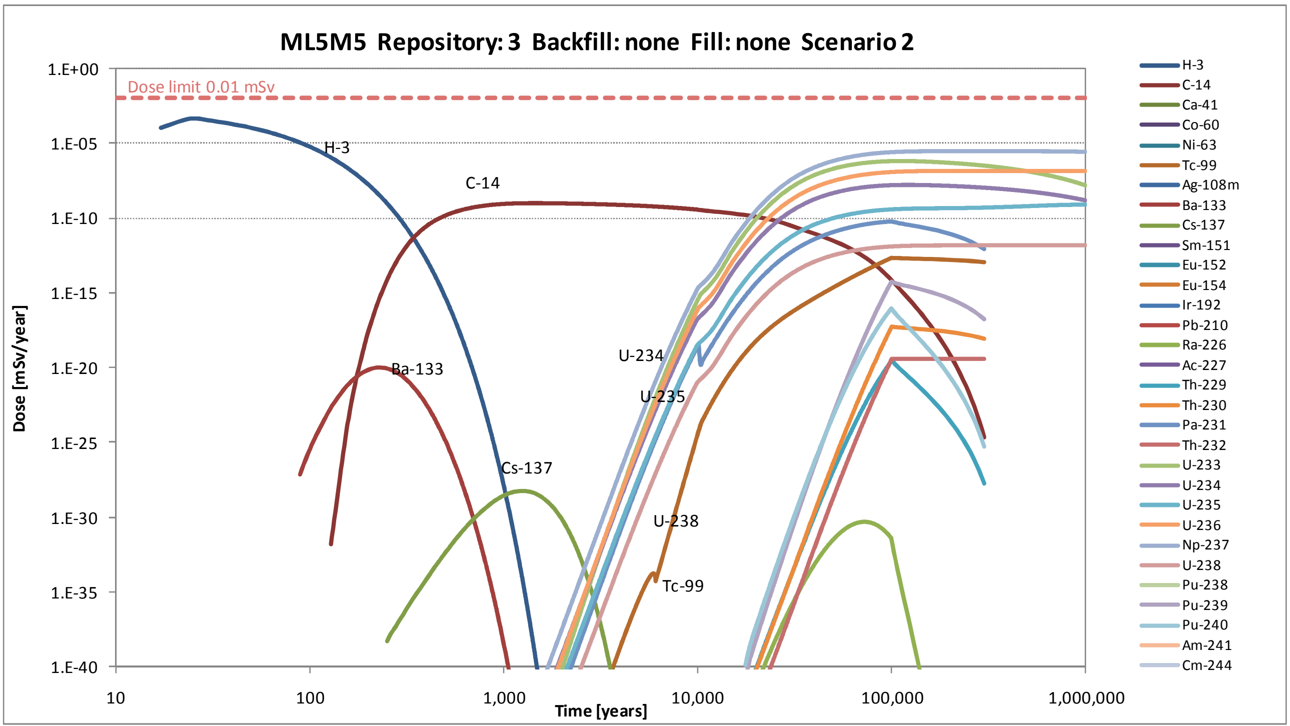

ML1M1 Repository: 1 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
C-14
H-3
Ba-133
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

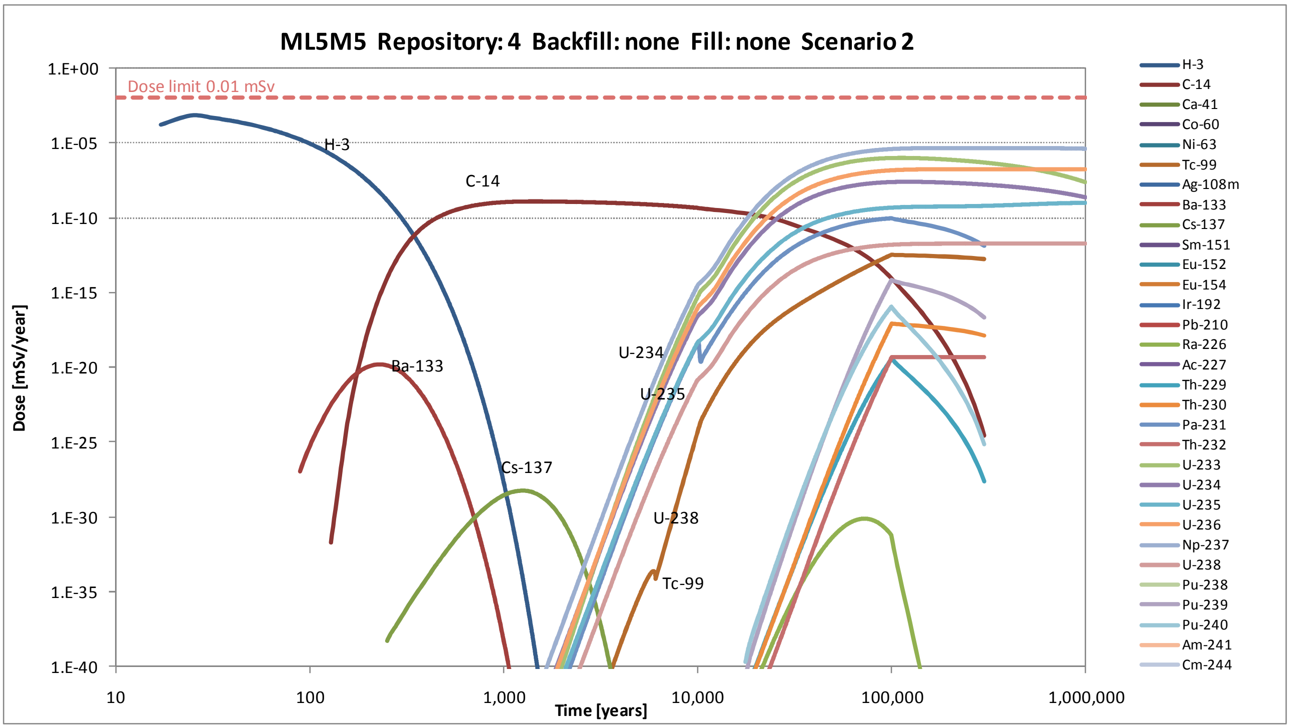

ML1M1 Repository: 2 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
H-3
C-14
Cs-137
Ba-133
U-234
U-235
U-238
Tc-99
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

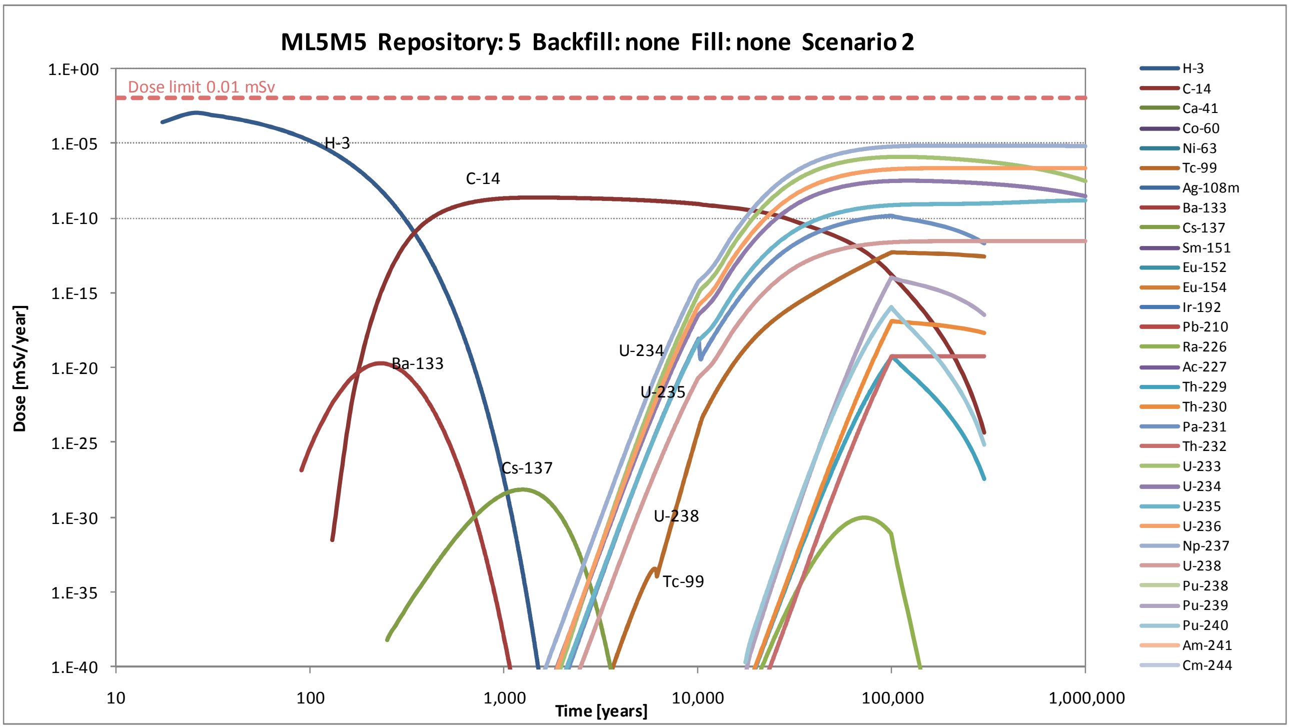

ML4M1 Repository: 1 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
C-14
H-3
Ba-133
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

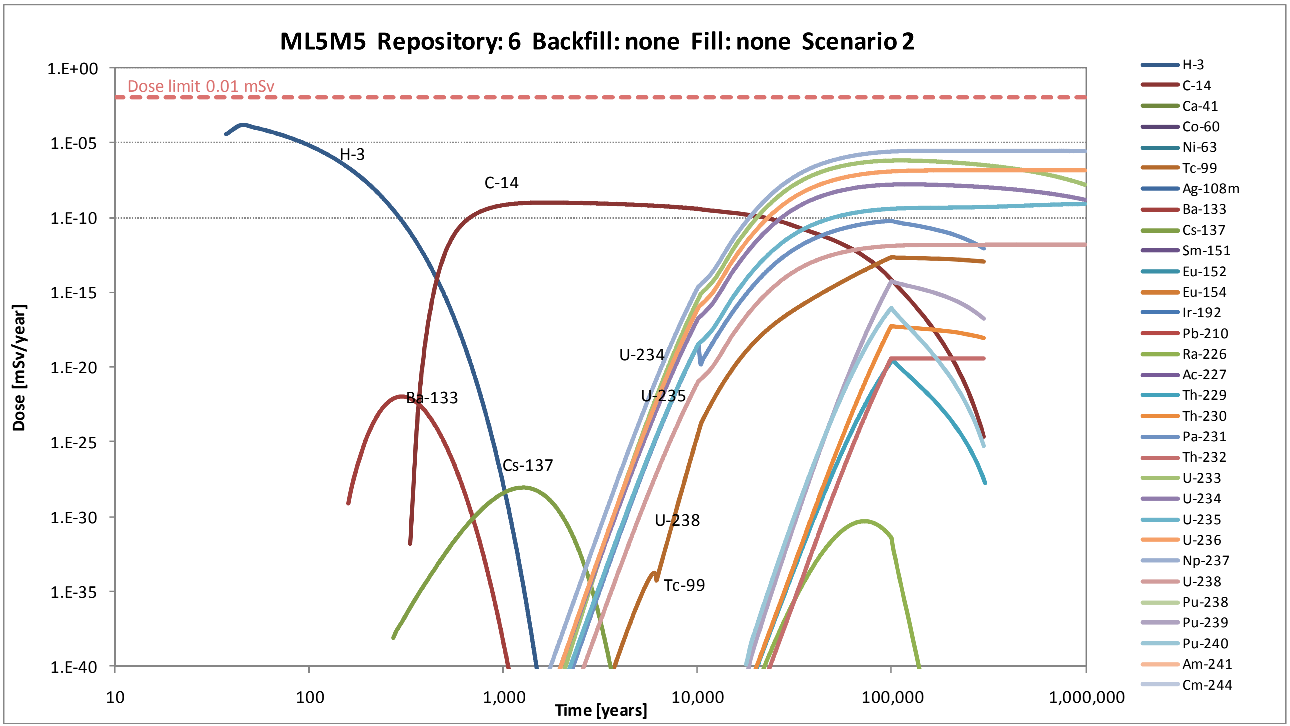

ML4M1 Repository: 2 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
Tc-99
H-3
C-14
Ba-133
Cs-137
U-234
U-235
U-238
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

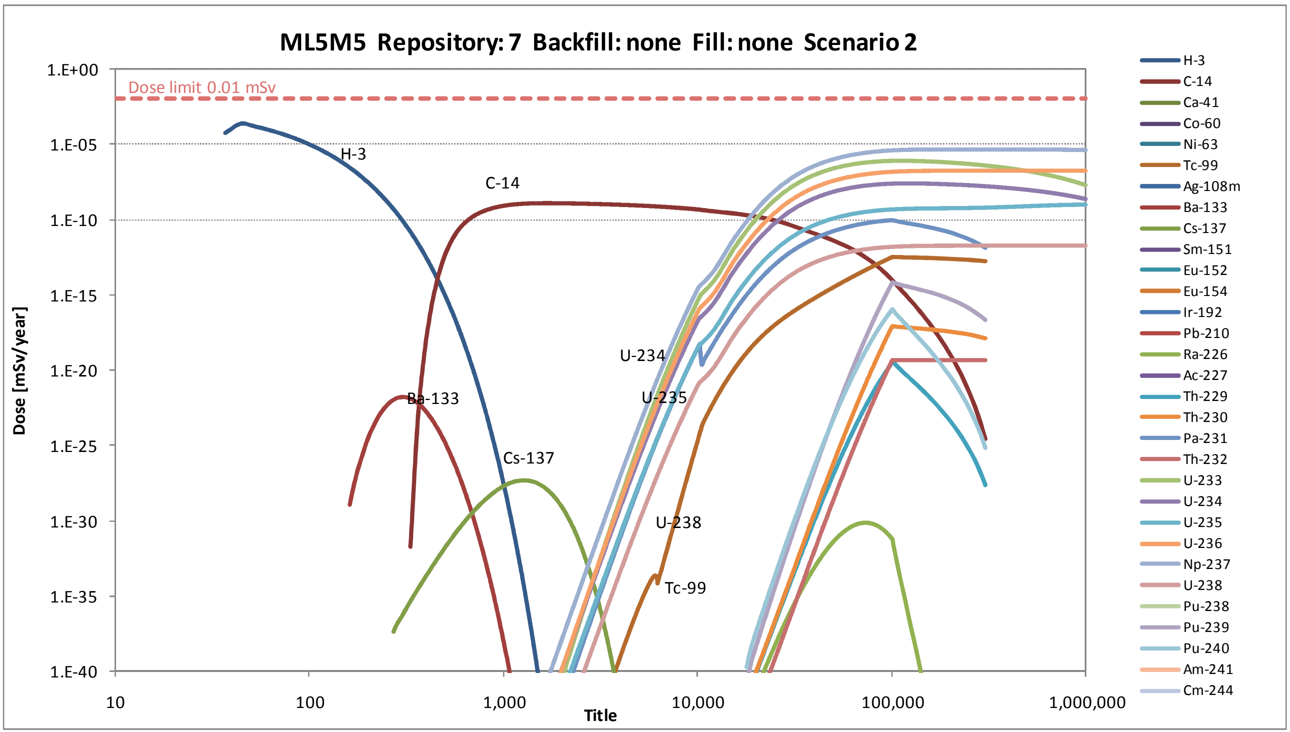

ML4M3 Repository: 2 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Tc-99
Cs-137
Ba-133
U-234
U-235
U-238
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

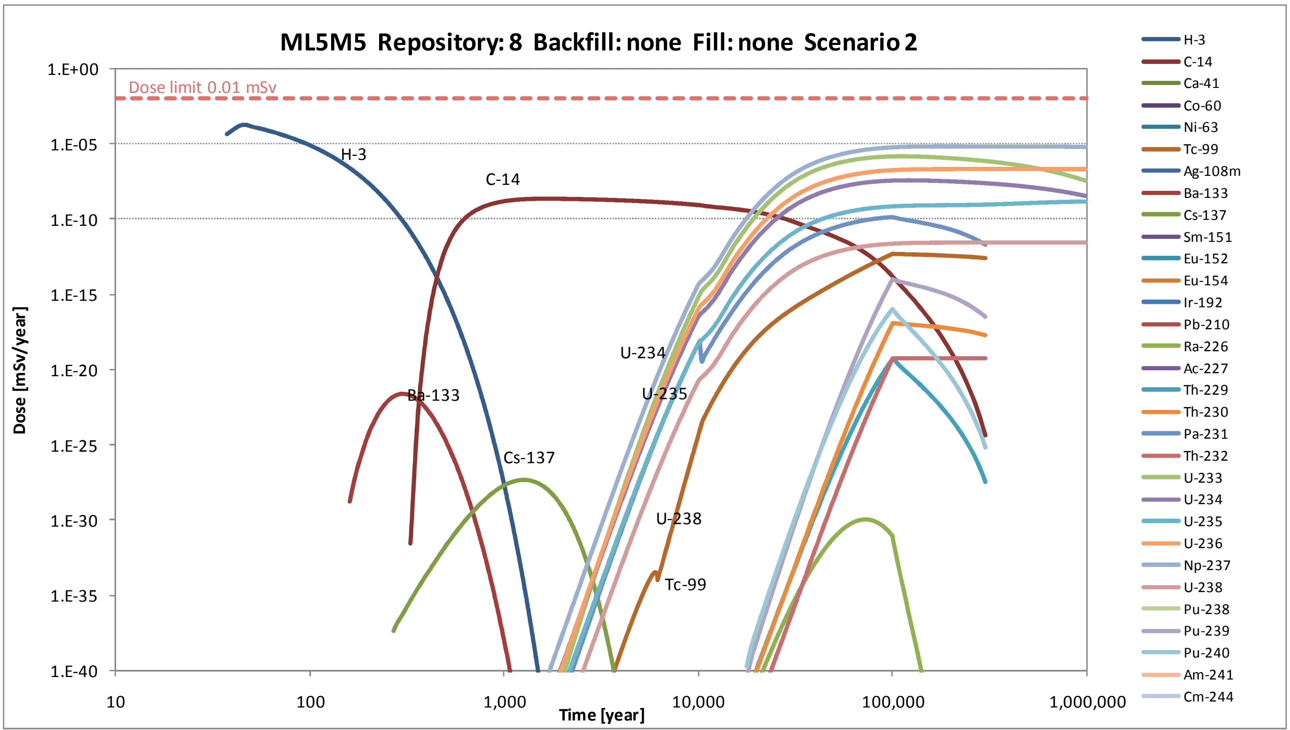

ML5M5 Repository: 3 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ba-133
Cs-137
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

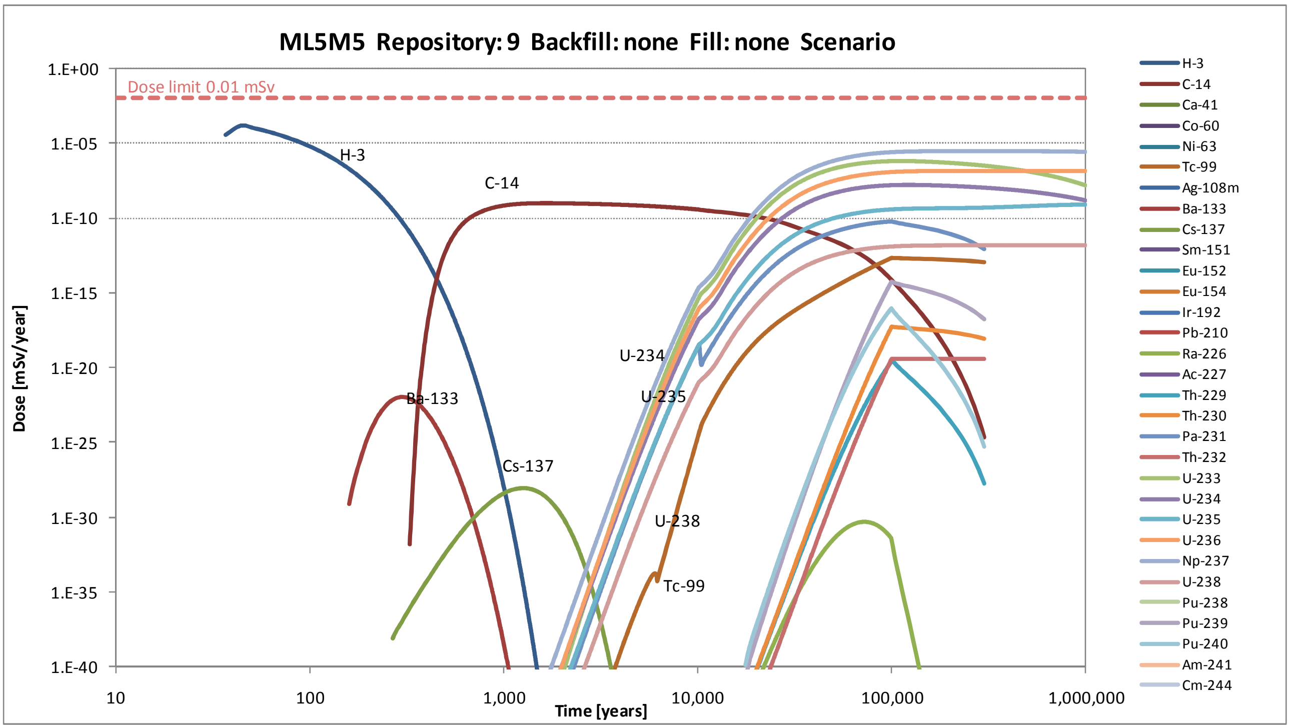

ML5M5 Repository: 4 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ba-133
Cs-137
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

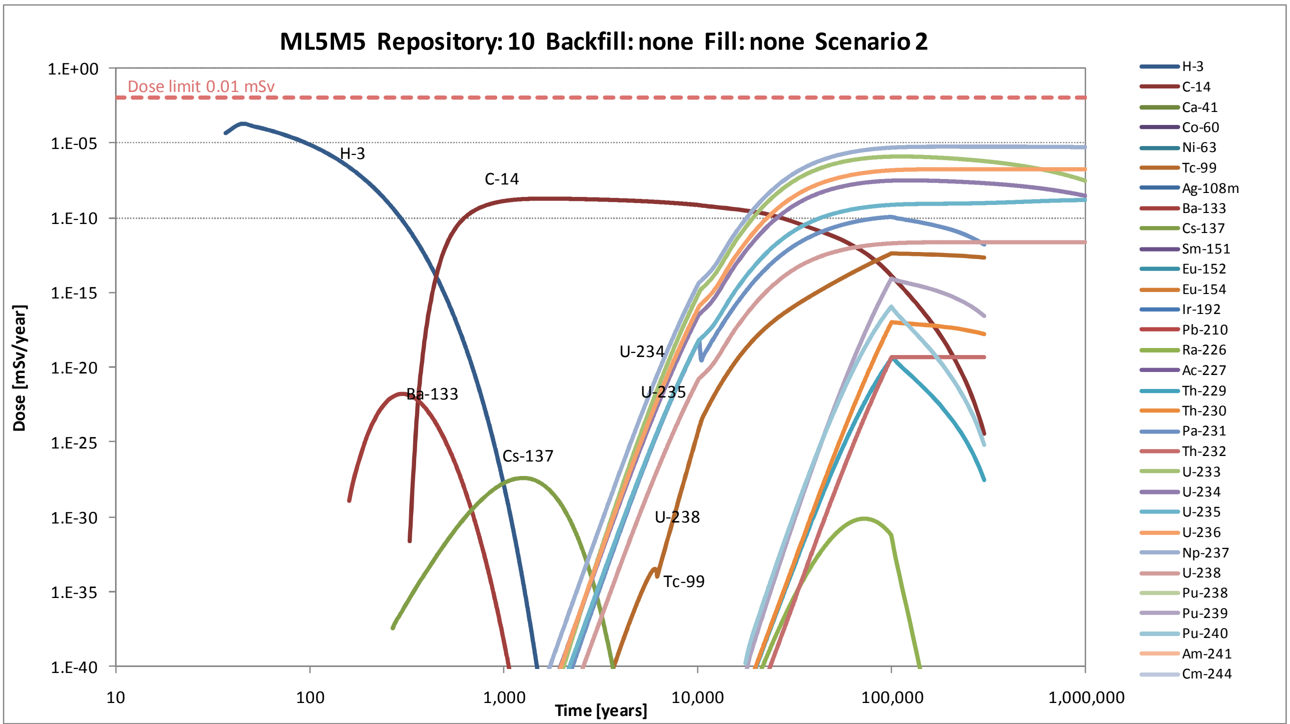

ML5M5 Repository: 5 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ba-133
Cs-137
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

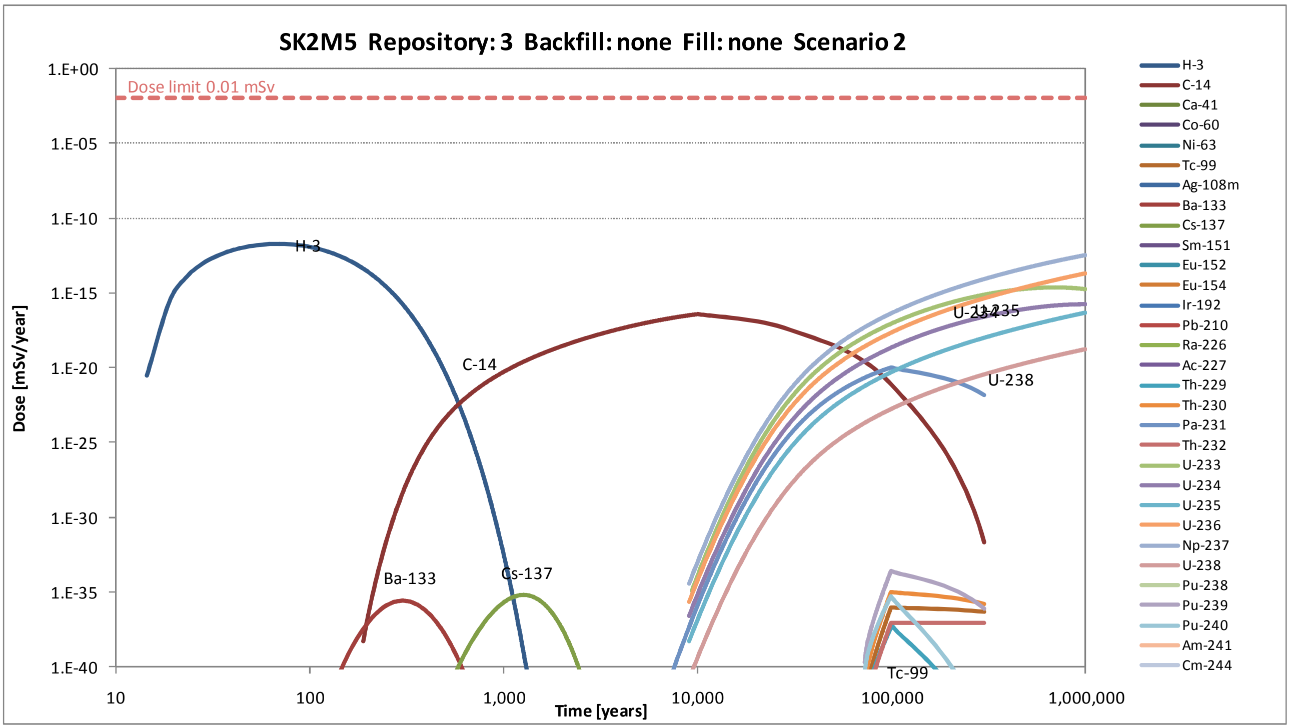

ML5M5 Repository: 6 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ba-133
Cs-137
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

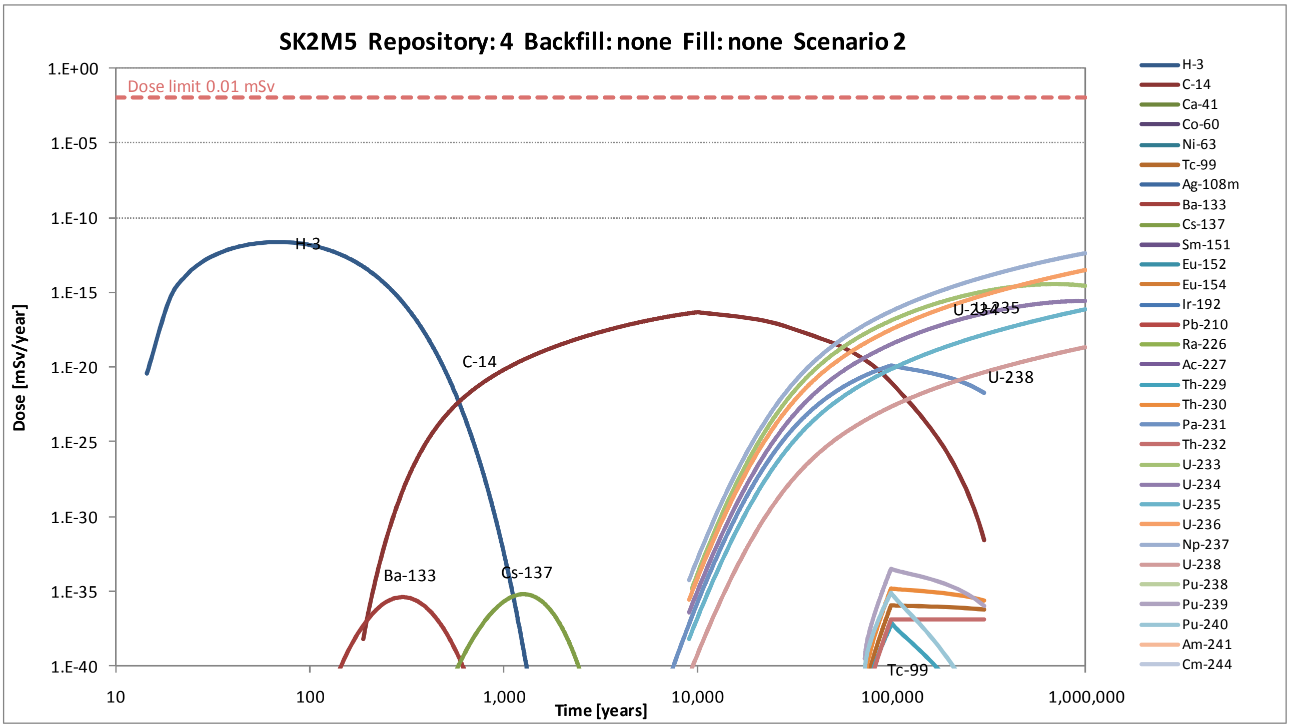

ML5M5 Repository: 7 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Title
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ba-133
Cs-137
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244


ML5M5 Repository: 8 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [year]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ba-133
Cs-137
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244


ML5M5 Repository: 9 Backfill: none Fill: none Scenario
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ba-133
Cs-137
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244


ML5M5 Repository: 10 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ba-133
Cs-137
U-234
U-235
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244


SK2M5 Repository: 3 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
H-3
C-14
Ba-133
Cs-137
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244


SK2M5 Repository: 4 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
H-3
C-14
Ba-133
Cs-137
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
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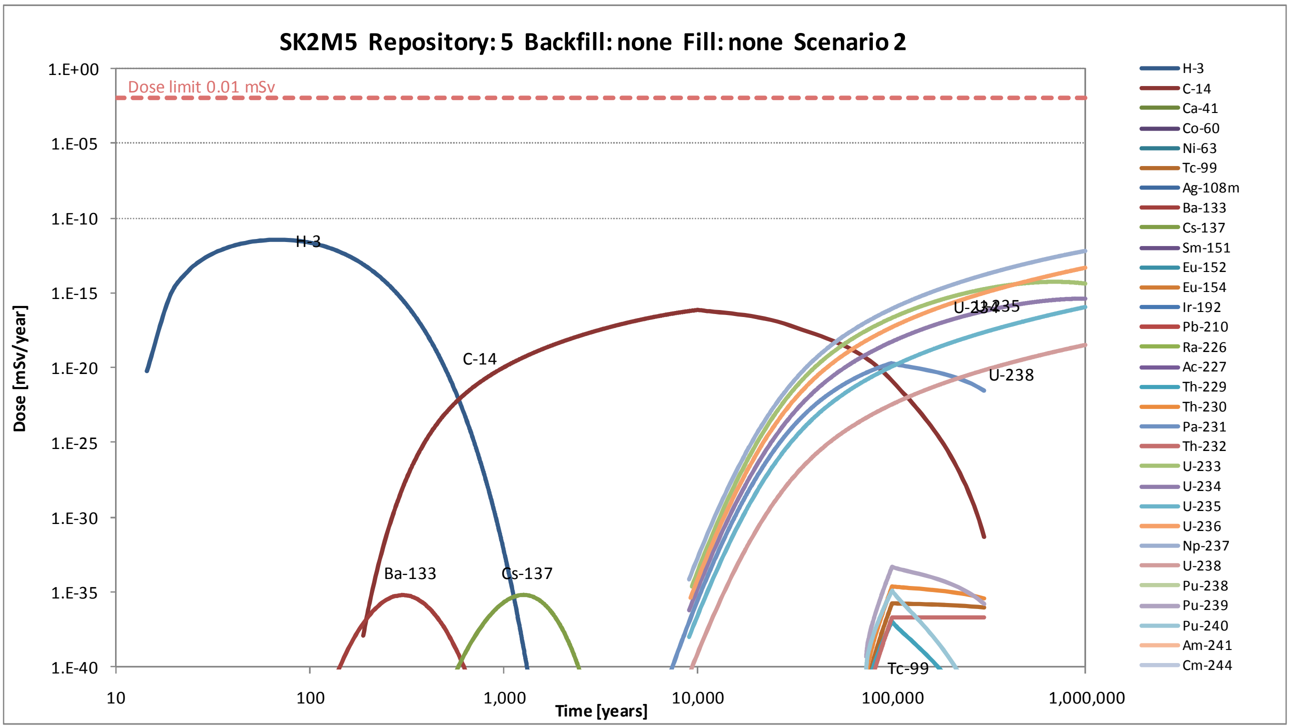
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244


SK2M5 Repository: 5 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ba-133
Cs-137
Tc-99
U-238
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
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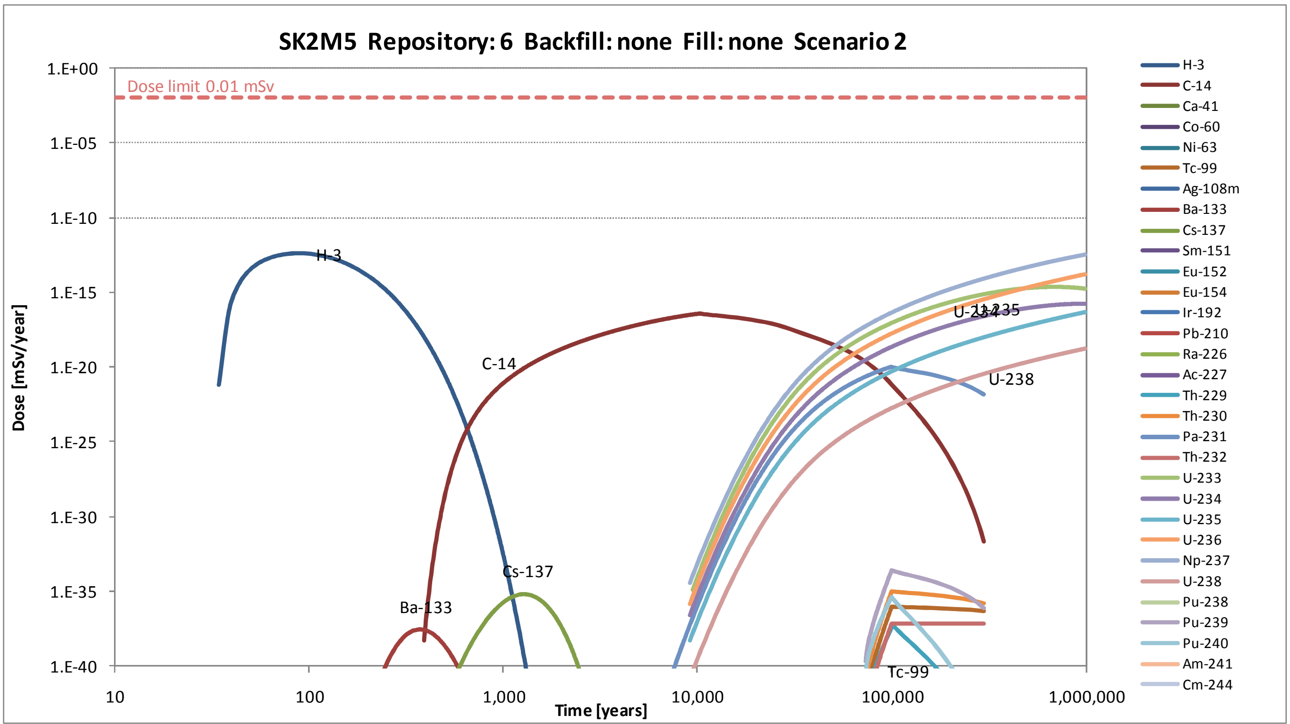
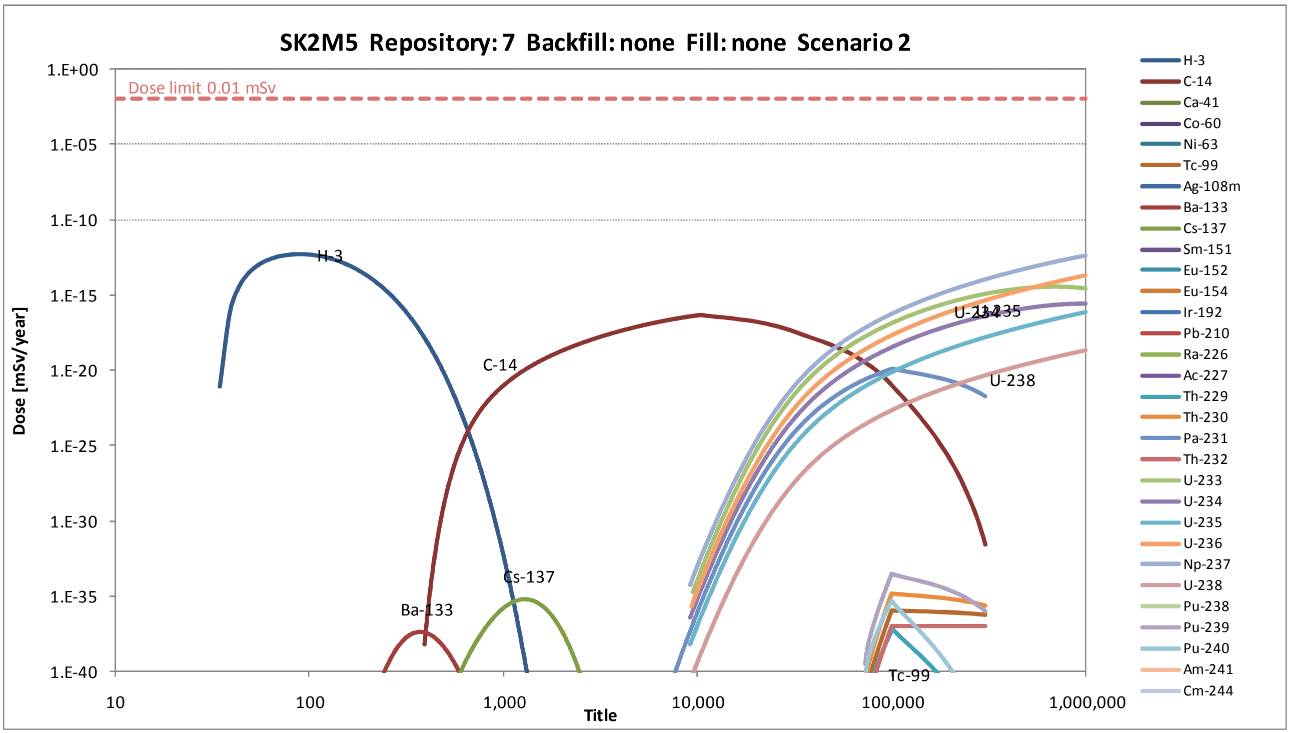
Pu-239
Pu-240
Am-241
Cm-244


SK2M5 Repository: 6 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ba-133
Cs-137
Tc-99
U-238
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244



SK2M5 Repository: 7 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Title
H-3
C-14
Ba-133
Cs-137
U-238
Tc-99
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
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1,000
10,000
100,000
1,000,000
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
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Pu-238
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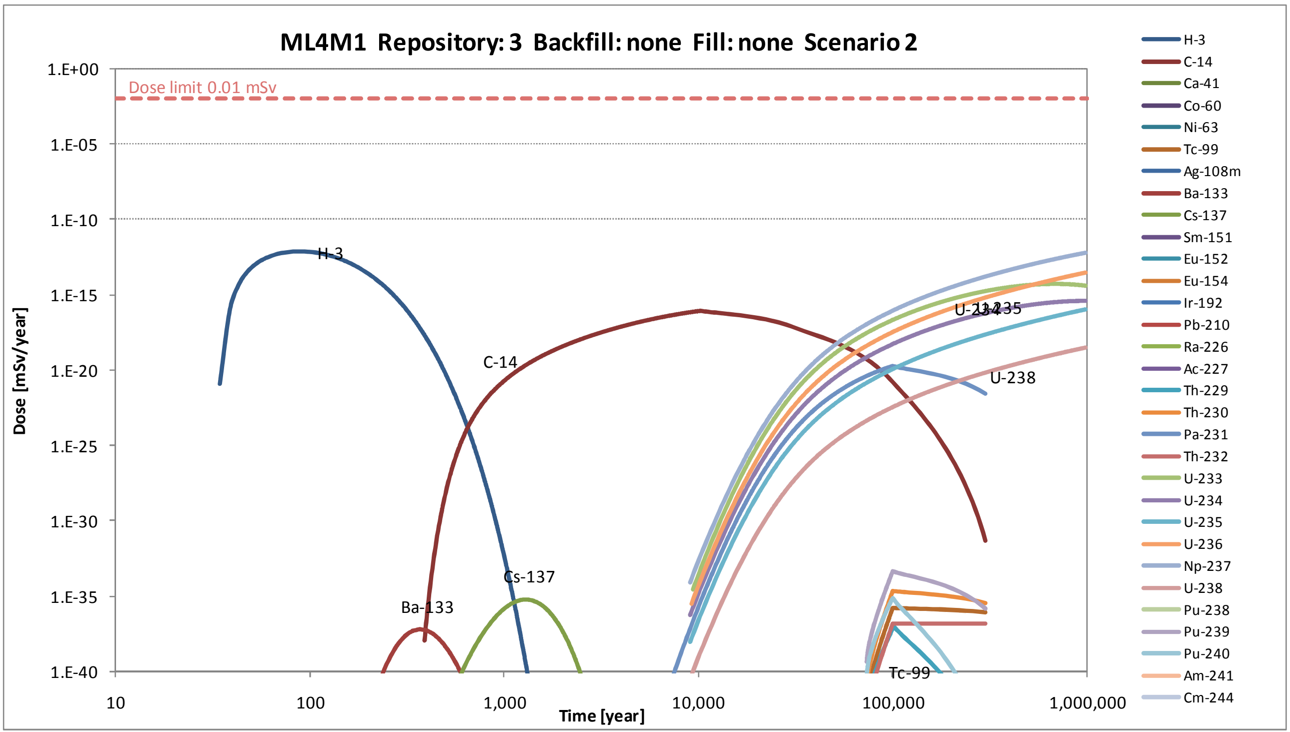
Pu-240
Am-241
Cm-244



ML4M1 Repository: 3 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [year]
H-3
C-14
Ba-133
Cs-137
U-238
Tc-99
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
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U-235
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Np-237
U-238
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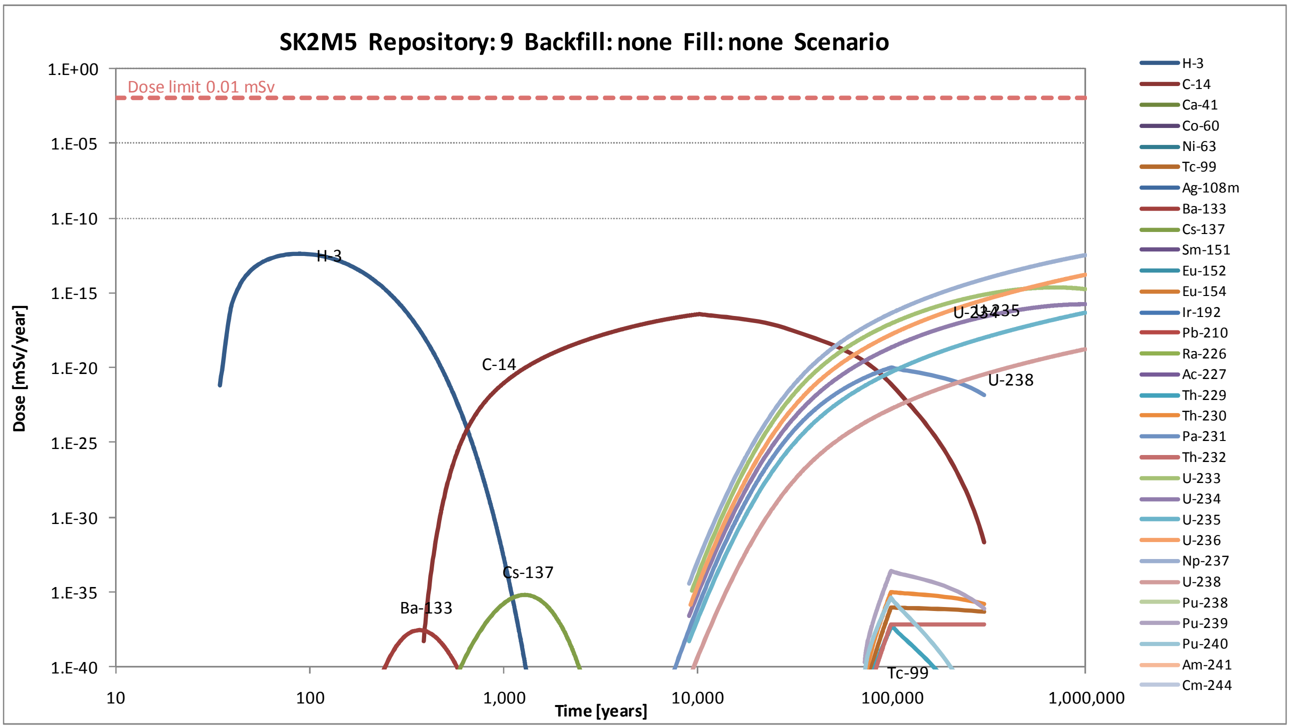
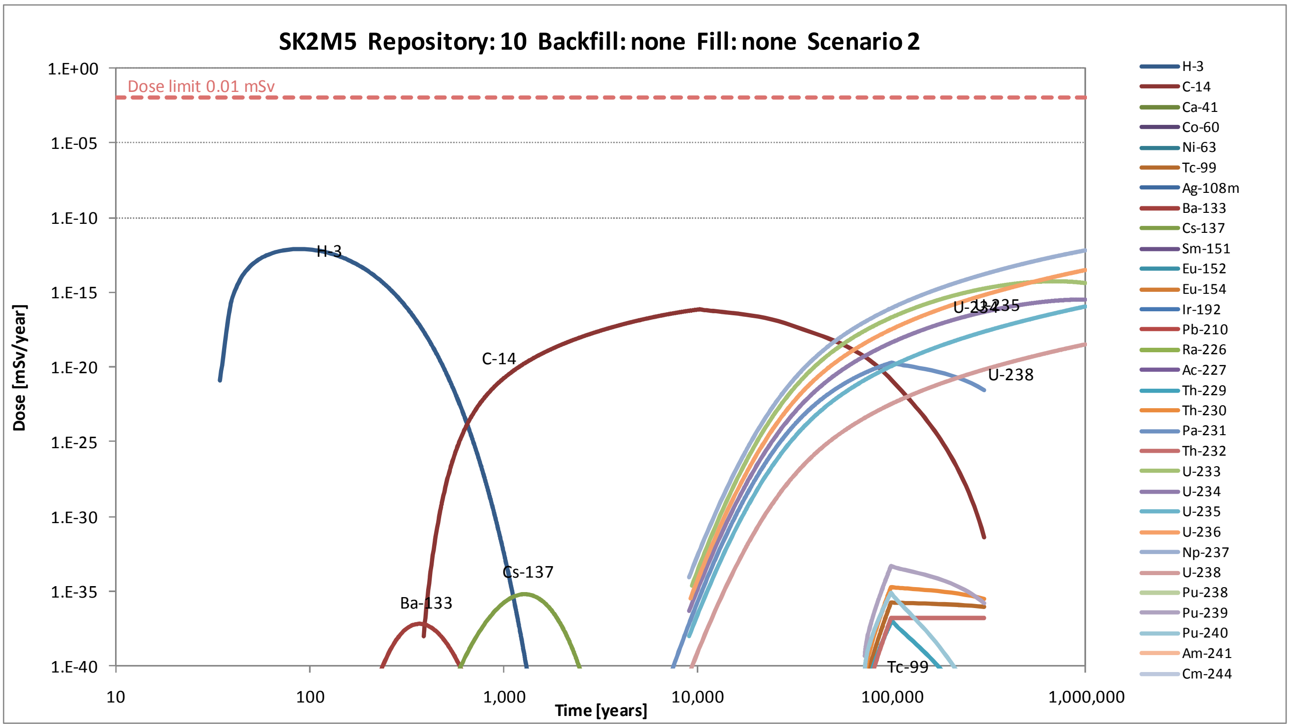
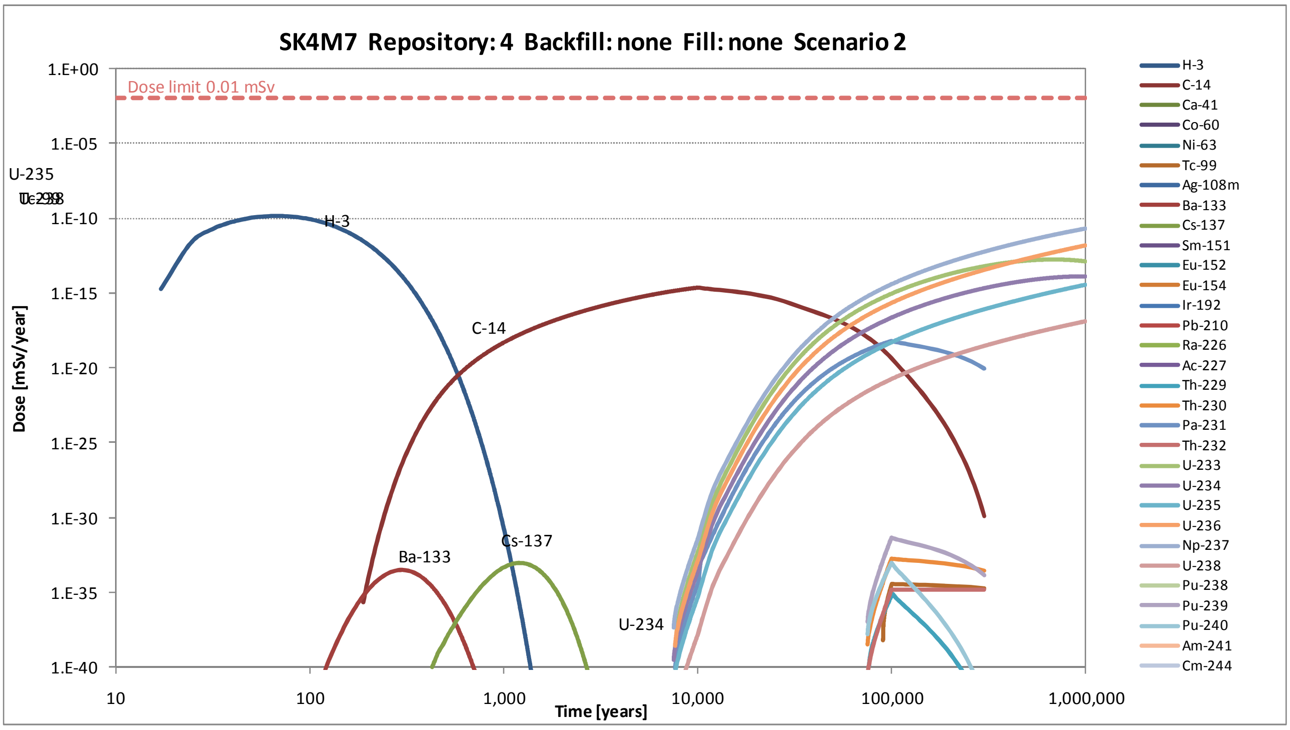
Pu-240
Am-241
Cm-244


SK2M5 Repository: 9 Backfill: none Fill: none Scenario
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
H-3
C-14
Ba-133
Cs-137
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244


SK2M5 Repository: 10 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
H-3
C-14
Ba-133
Cs-137
U-238
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244


SK4M7 Repository: 4 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
H-3
C-14
Ba-133
Cs-137
U-234
U-235
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
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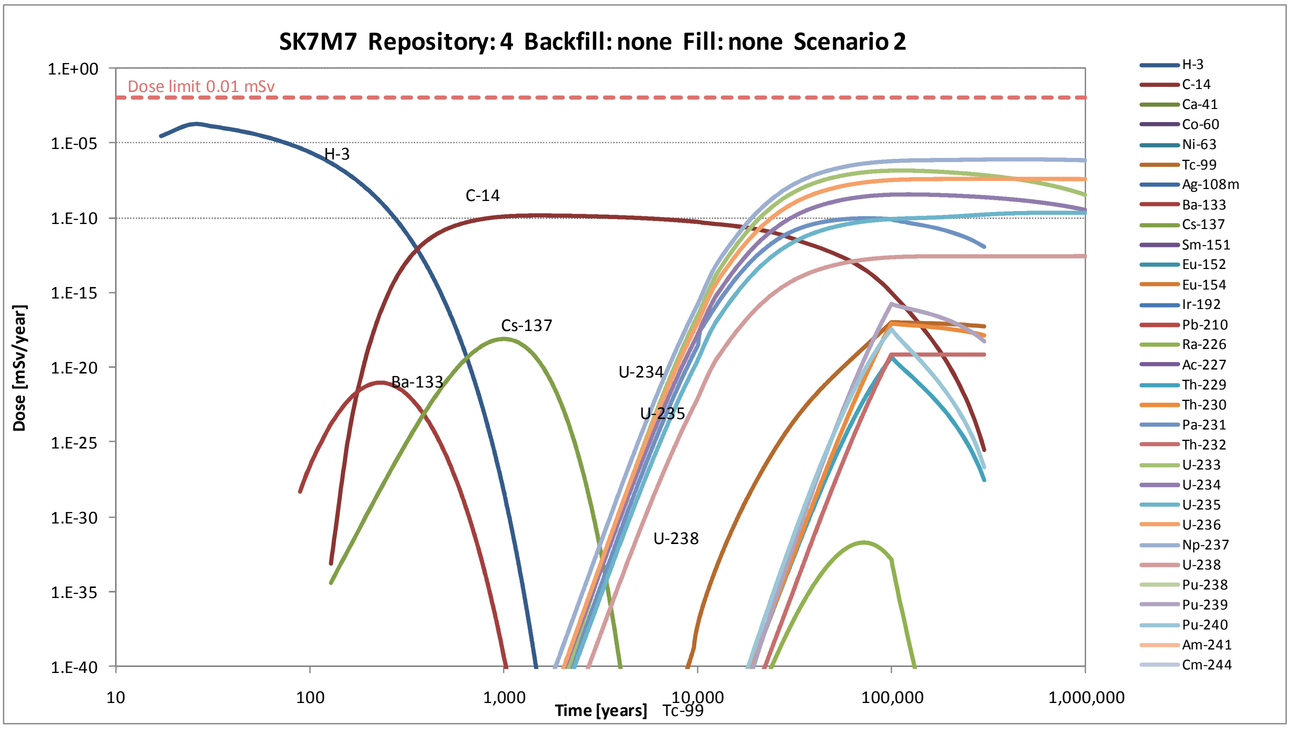
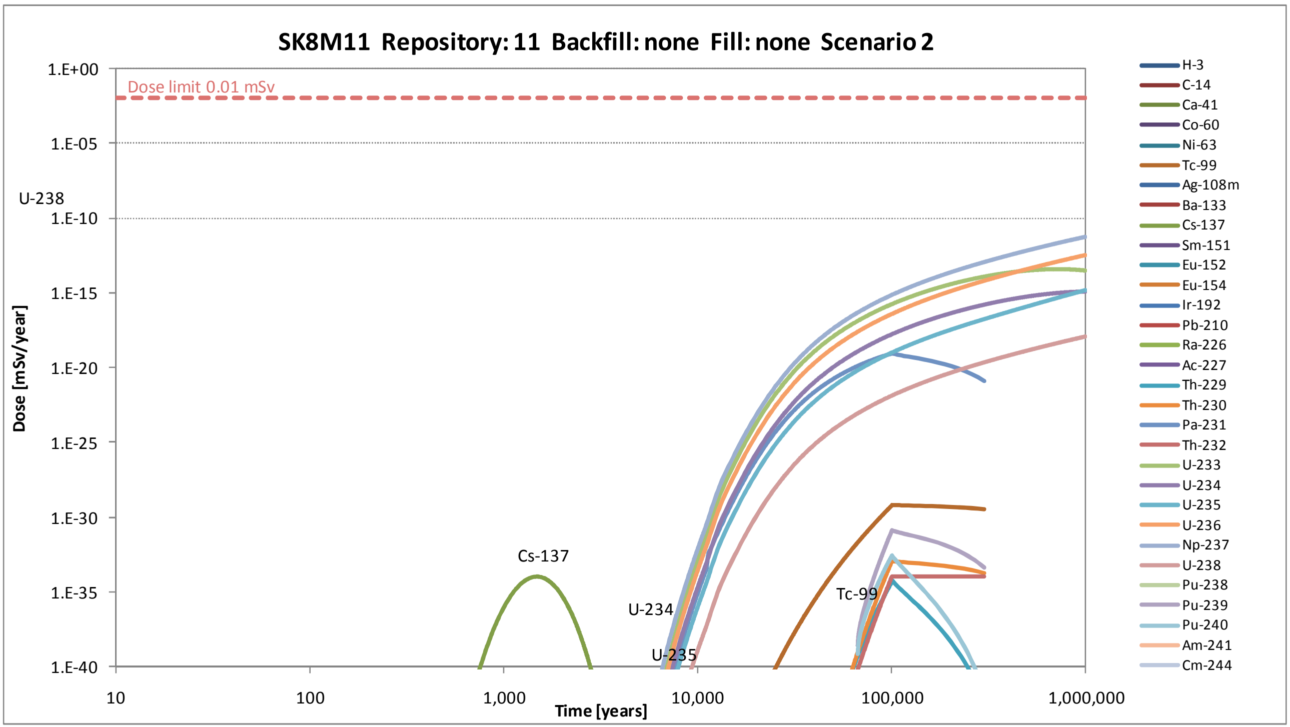
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244


SK7M7 Repository: 4 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
H-3
C-14
Ba-133
Cs-137
U-234
U-235
U-238
Tc-99
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244


SK8M11 Repository: 11 Backfill: none Fill: none Scenario 2
Dose limit 0.01 mSv
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
U-238
Cs-137
U-234
U-235
Tc-99
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
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U-234
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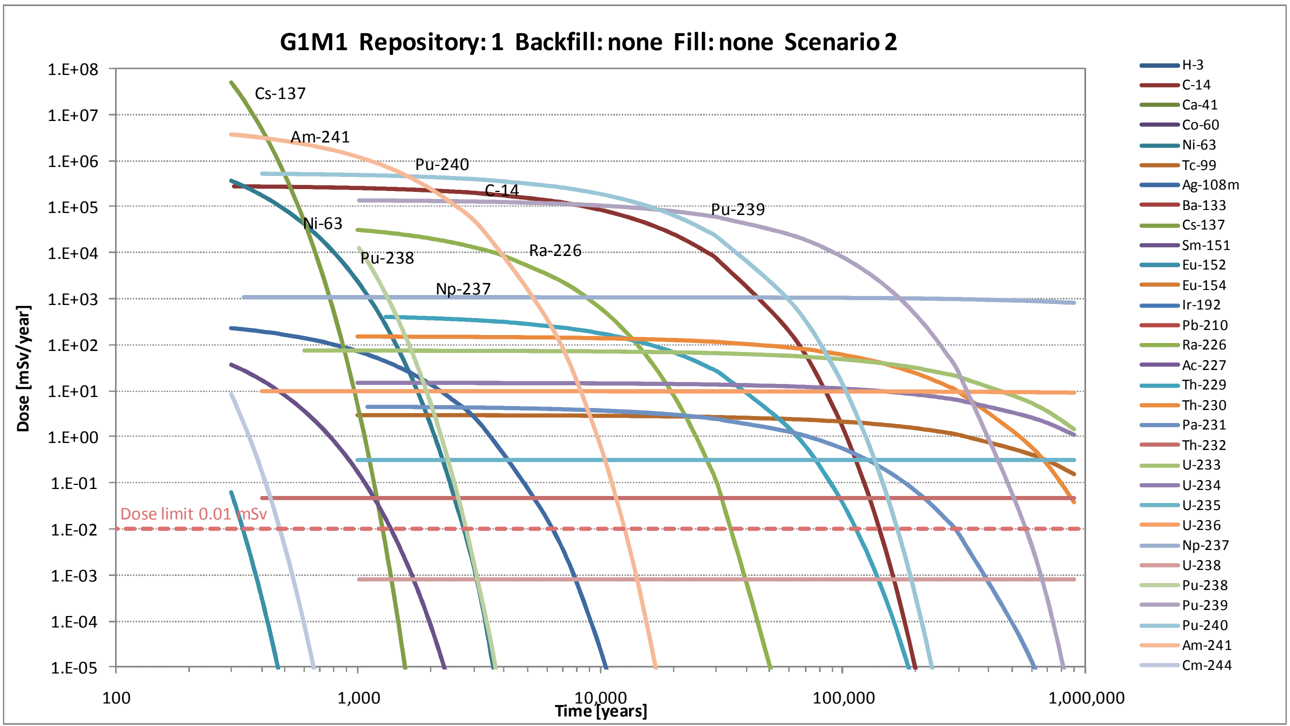
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244


G1M1 Repository: 1 Backfill: none Fill: none Scenario 2
Dose [mSv/year]
Time [years]
1.E+08
1.E+07
1.E+06
1.E+05
1.E+04
1.E+03
1.E+02
1.E+01
1.E+00
1.E-01
1.E-02
1.E-03
1.E-04
1.E-05
100
1,000
10,000
100,000
1,000,000
Cs-137
Am-241
Pu-240
C-14
Pu-239
Ni-63
Ra-226
Pu-238
Np-237
Dose limit 0.01 mSv
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
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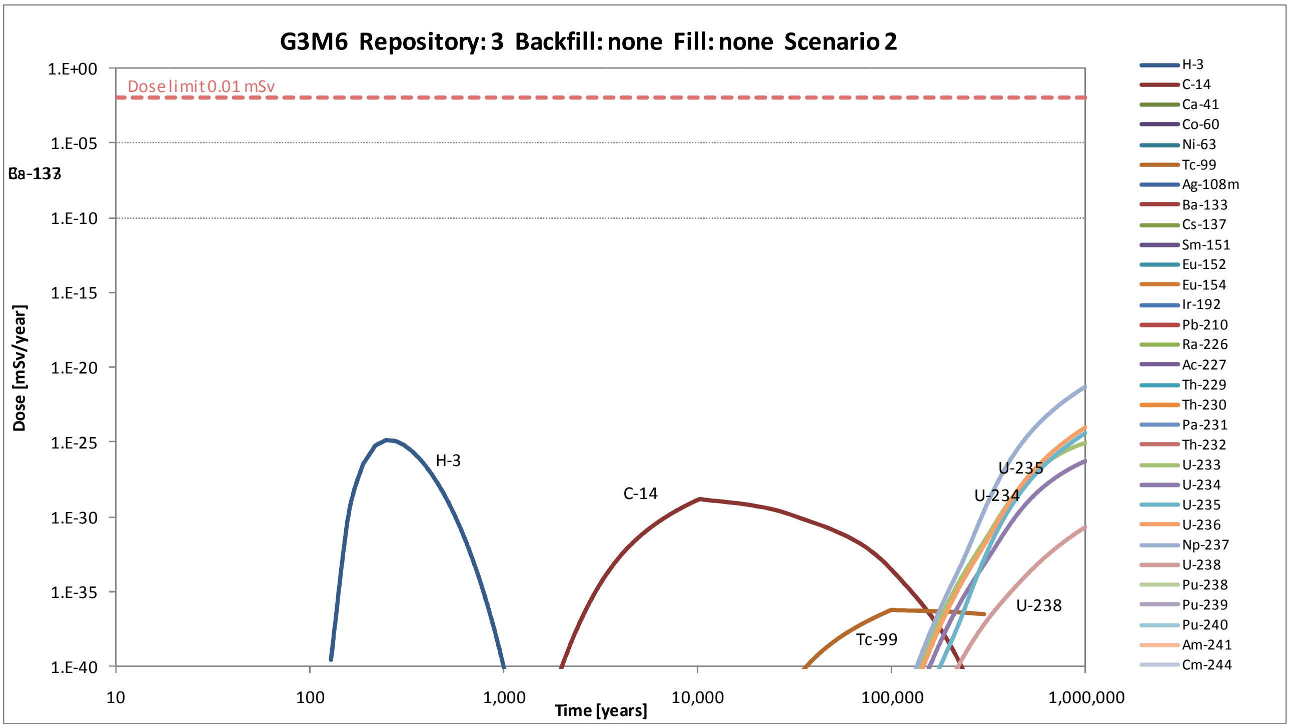
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244


G3M6 Repository: 3 Backfill: none Fill: none Scenario 2
Dose [mSv/year]
Time [years]
1.E+00
1.E-05
1.E-10
1.E-15
1.E-20
1.E-25
1.E-30
1.E-35
1.E-40
10
100
1,000
10,000
100,000
1,000,000
Dose limit 0.01 mSv
Ba-133
H-3
C-14
Tc-99
U-235
U-234
U-238
H-3
C-14
Ca-41
Co-60
Ni-63
Tc-99
Ag-108m
Ba-133
Cs-137
Sm-151
Eu-152
Eu-154
Ir-192
Pb-210
Ra-226
Ac-227
Th-229
Th-230
Pa-231
Th-232
U-233
U-234
U-235
U-236
Np-237
U-238
Pu-238
Pu-239
Pu-240
Am-241
Cm-244

Appendix L: Visual appearance of the repository

Pre-feasibility Study for Final Disposal of Radioactive Waste - Working Report 13
Visual Appearance of the Repository in the Landscape - Drafts
Appendix

Hasløv & Kjærsgaard January 2011

Working Report 13 Pre-feasibility Study for Final Disposal of Radioactive Waste
Visual Appearance in the Landscape
Appendix

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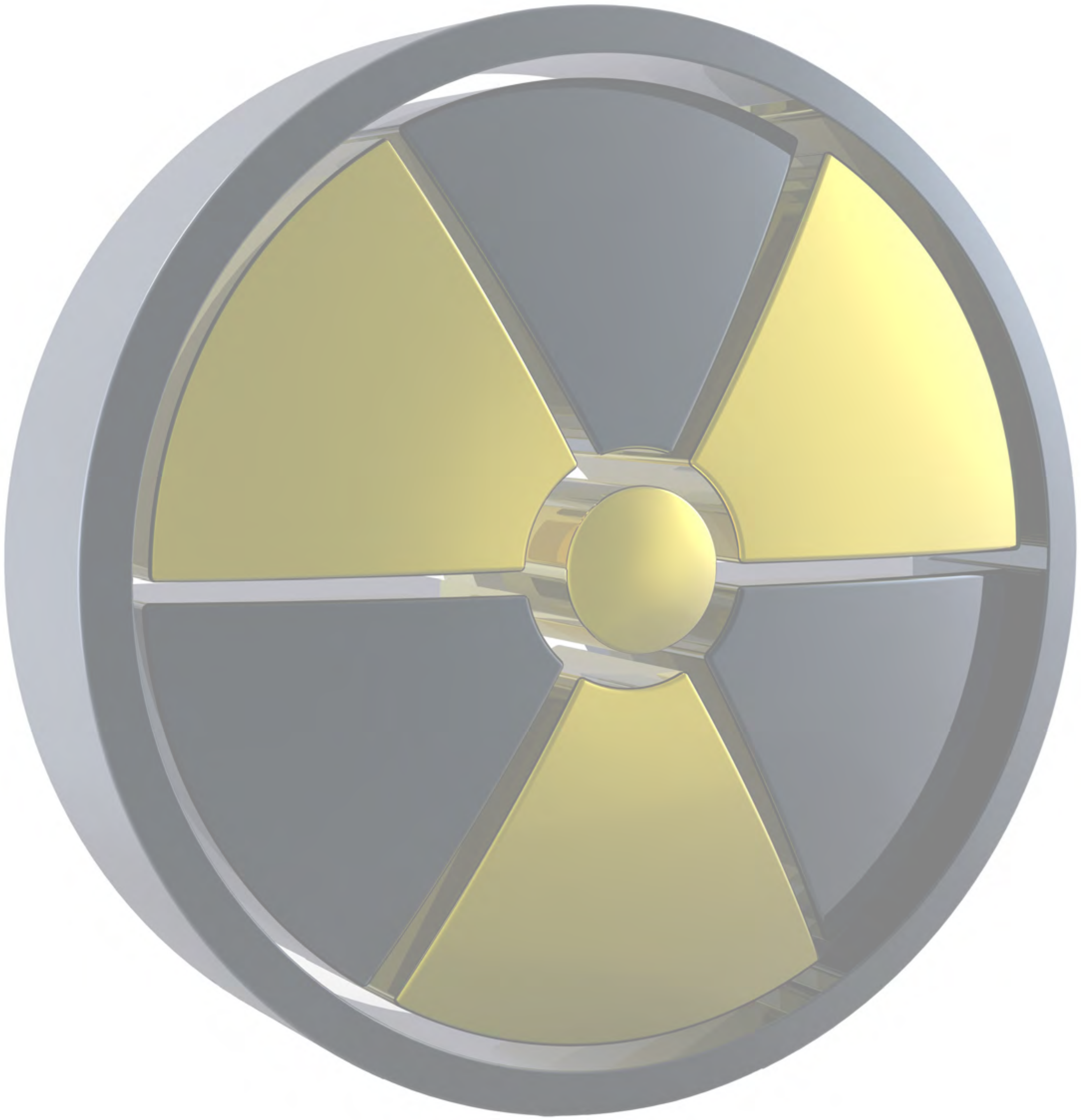


References and Intentions

The visual appearance project relies heavily on the report WR7, Danish Decommissioning (2010): Working Report 7, Conceptual design of different repository types, report prepared by COWI.
This report outlines the basic parameters for the physical appearance of the repository.

The inspirations and design principles are described in the WR13, Danish Decomissioning (2011):
Working Report 13, Visual Appearance of the Repository in the Landscape, report prepared by Hasløv & Kjærsgaard, to witch this present publication is an appendix.

The mode of operation, layout and size is determined in the above mentioned reports. This has led to the drafts for the design presented in this publication. It must be stressed that the design is initial because of the fact that the final location of the repository is yet to be determined.

Prerequisites:

• The repository has to be placed on a site equal to or less than 25.000 m².

• The repository has to be an integrated part of the landscape, and therefore limited in height, estimated to less than 10 m. This fact limits the visual impact in relation to common surrounding features.

• The repository has to be recognizable as a special plant, with a specific function.

• The visual appearance is to be inviting and visually pleasing and at the same time maintain strict security measures.

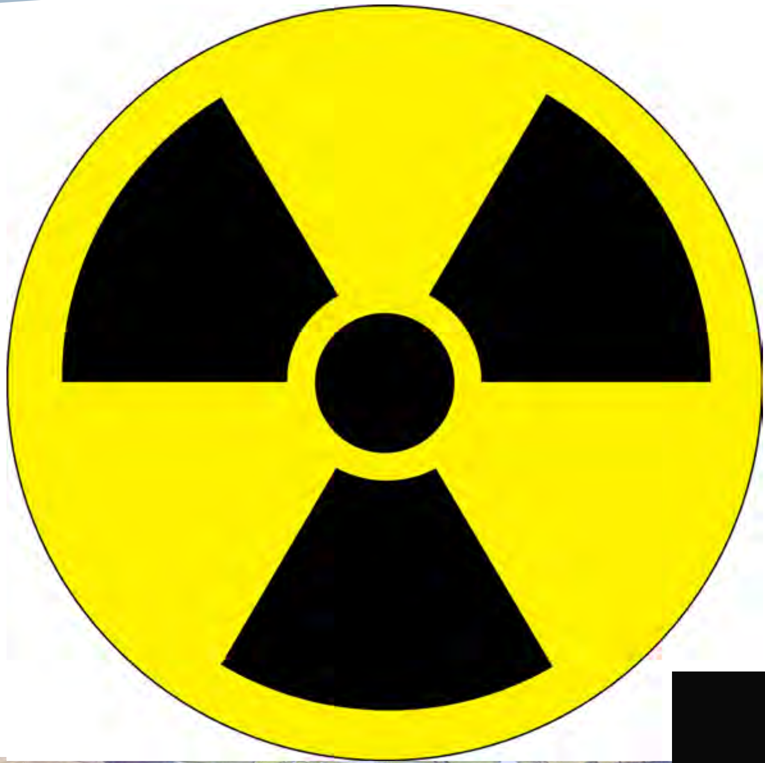
• The facility has to ensure good working conditions and easy maintenance.

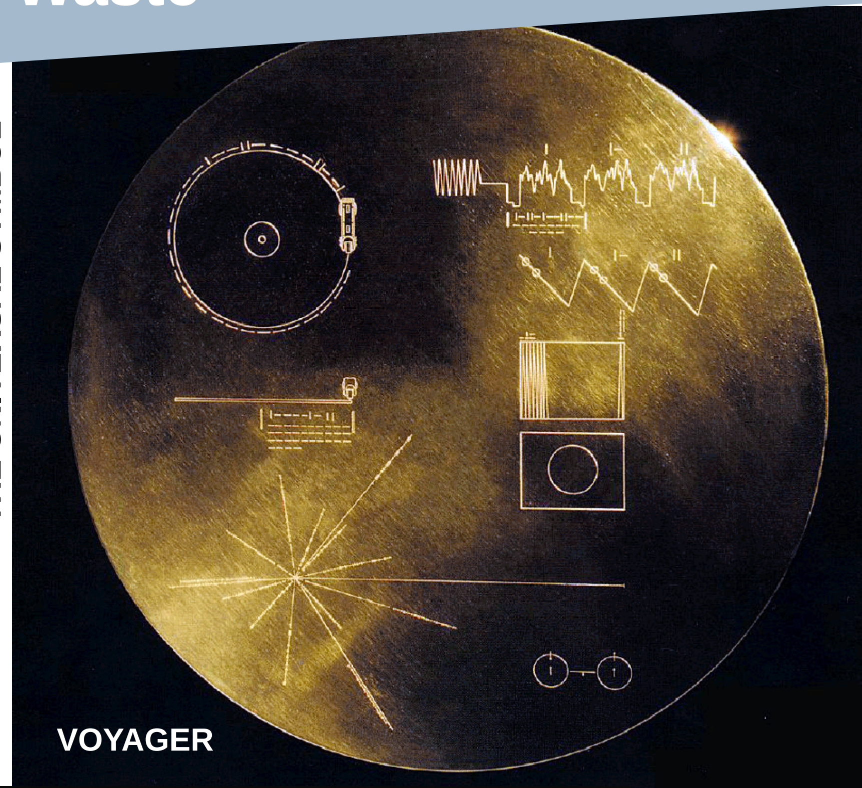
• It also has to be as environmentally sustainable as possible.

All the above has led to the present design that tries to integrate as many of the above prerequisites as possible, leading to a facility where the sum of the individual parts is even more pleasing than the individual parts.

INSPIRATIONS


AEREAL ARCHEOLOGY
TRANSLATION TRANSCENDING TIME
THE ROSETTA STONE


THE UNIVERSAL SYMBOL


VOYAGER

PRESERVATION THROUGH DESIGN


VIKING FORTRESSES
TRELLEBORG

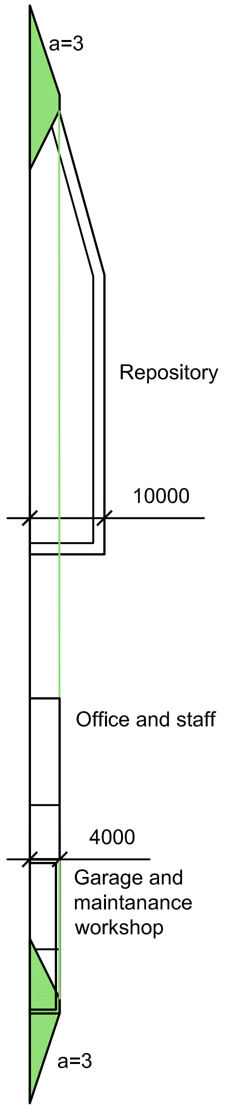

AGGERSBORG

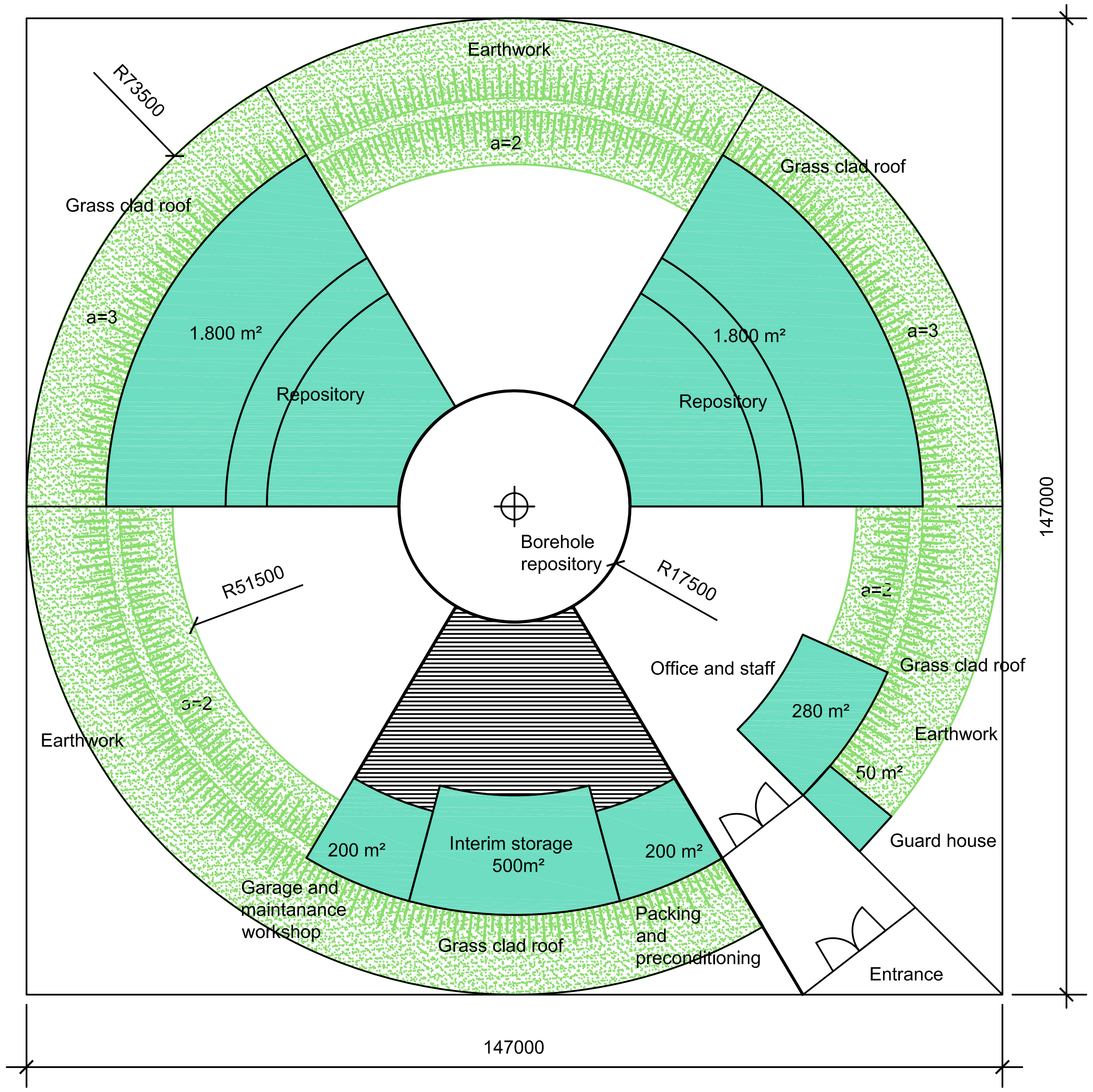
Visualization of the repository in a “typical” Danish landscape.

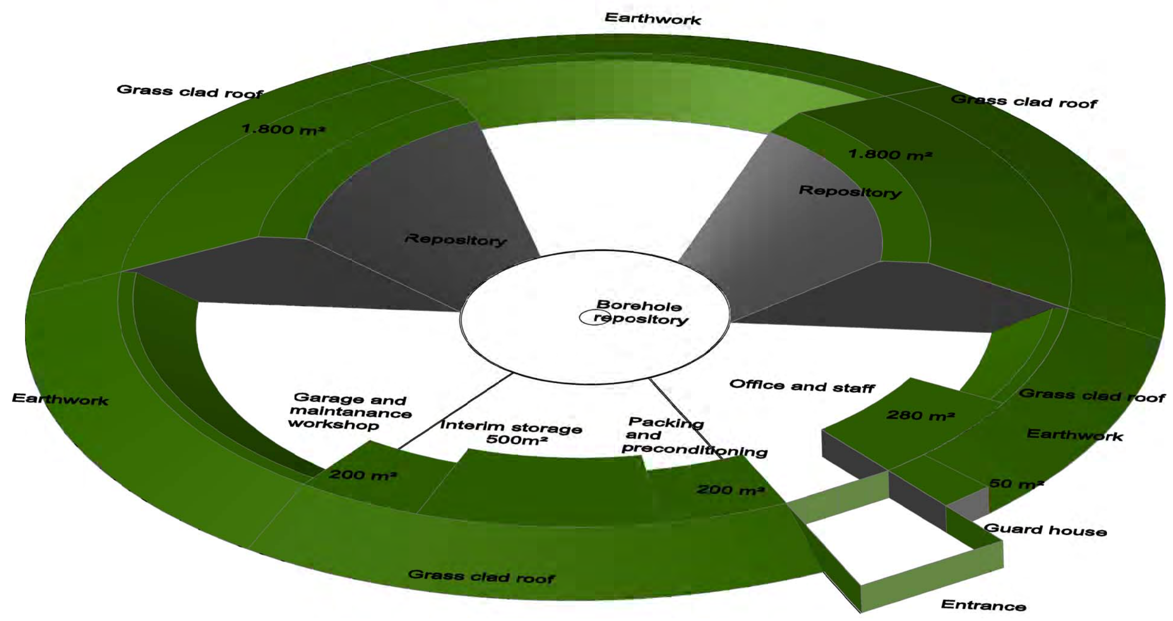
This visualization is based upon the fact that no actual site for the repository has been chosen yet. Therefore an arche-typical danish landscape has been used as representative for the placement of the facility.

The repository is shown without the visitors centre and treatment plant. The latter could be placed near to the main entrance or a bit further away, according to the analysis of the actual site (See page 11)

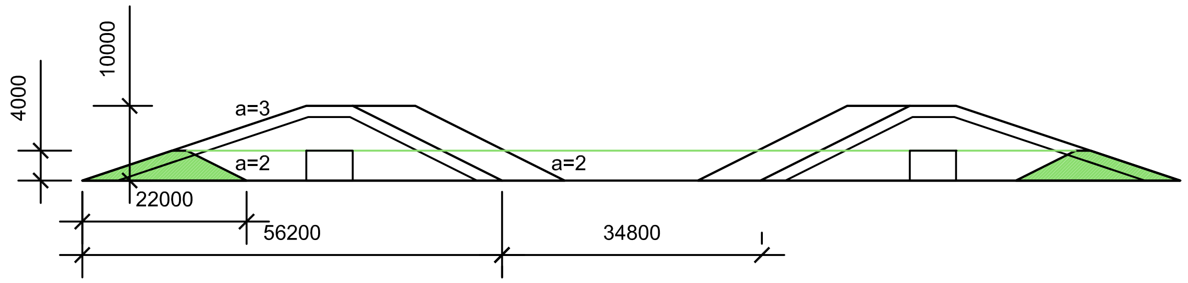
Working Report 13 Pre-feasibility Study for Final Disposal of Radioactive Waste
Visual Appearance in the Landscape
Appendix



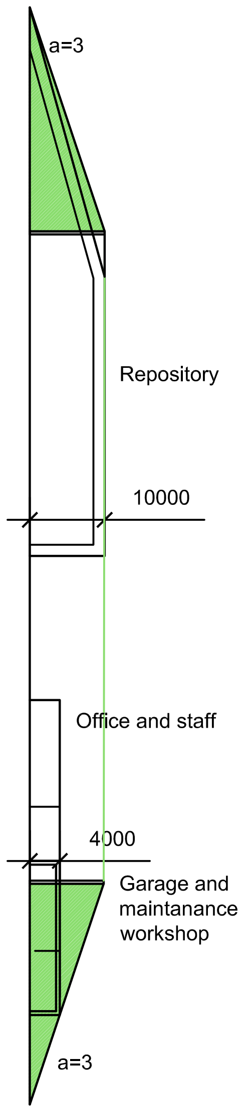


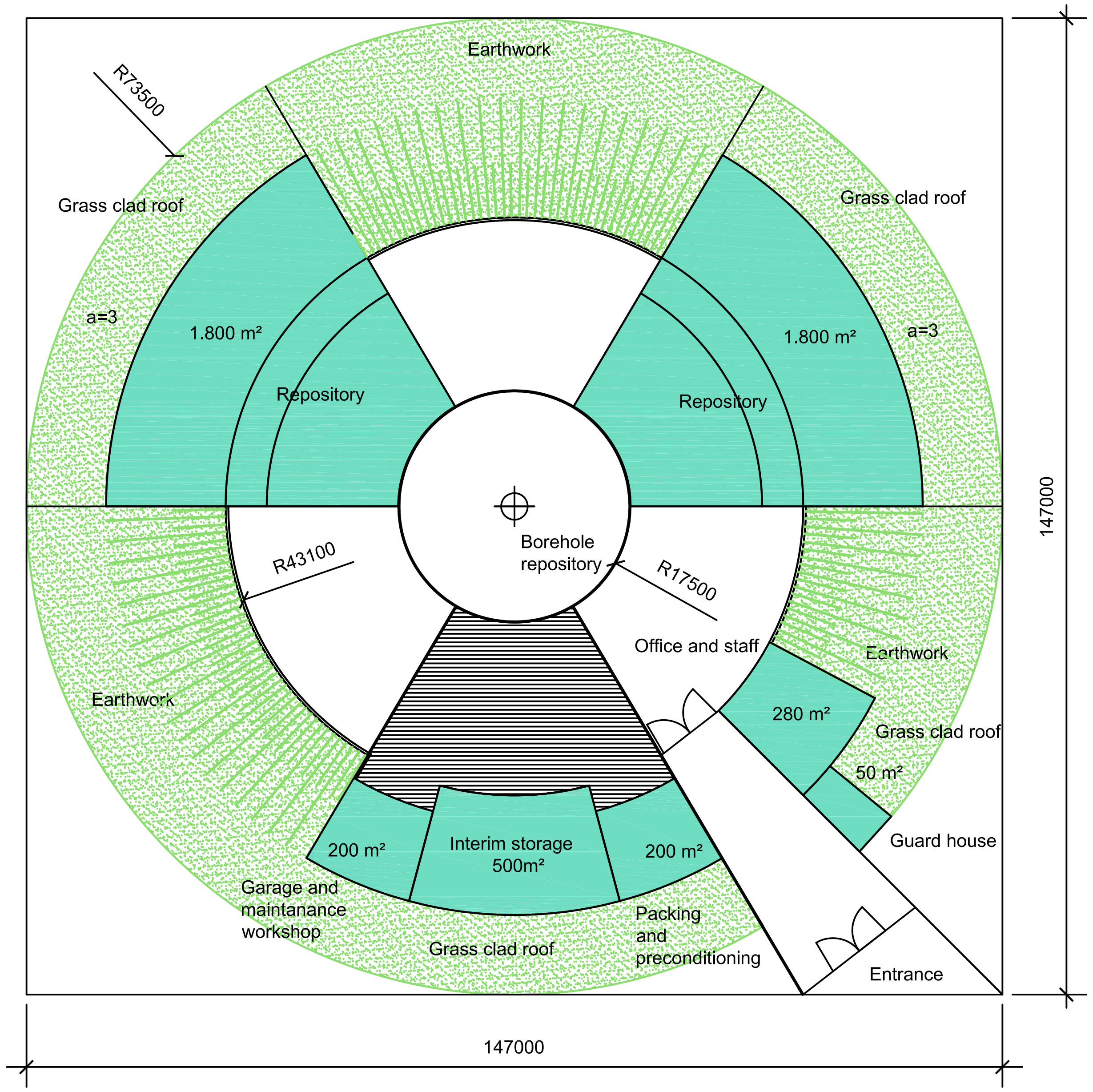


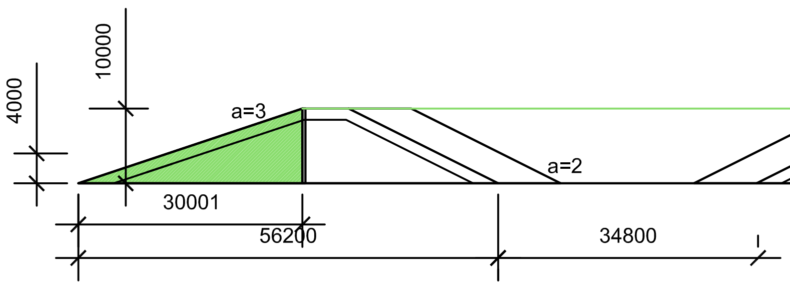
This draft constitutes the basic model for the near surface repository. All building volumes are integrated, as part of the earthworks and the roofs are clad in the same material as the earthworks e.g. grass.
The repository itself is divided into two parts in order to fit into the trefoil theme of the symbol.

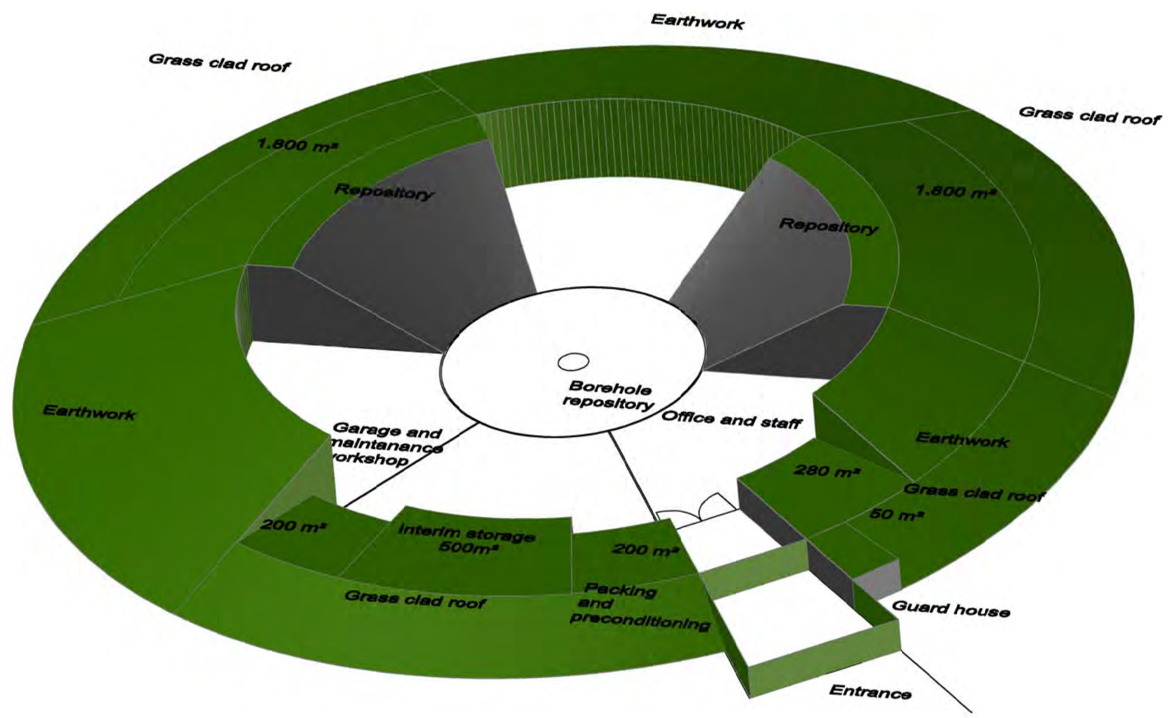


Visual appearance of the repository in the landscape
Draft 01
Plan, cross section and isometric illustration
Date: 28.12.2010
Hasløv & Kjærsgaard / BB / BUL



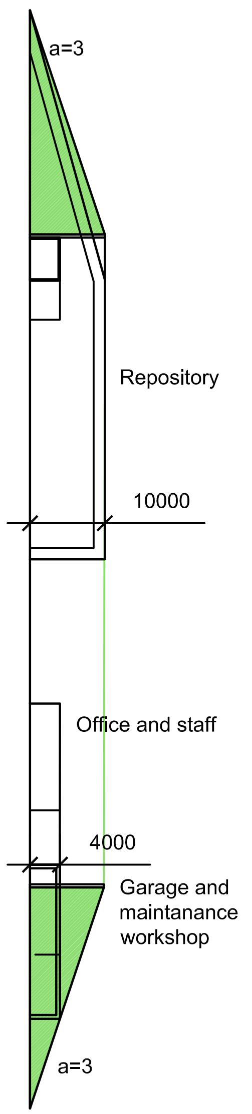


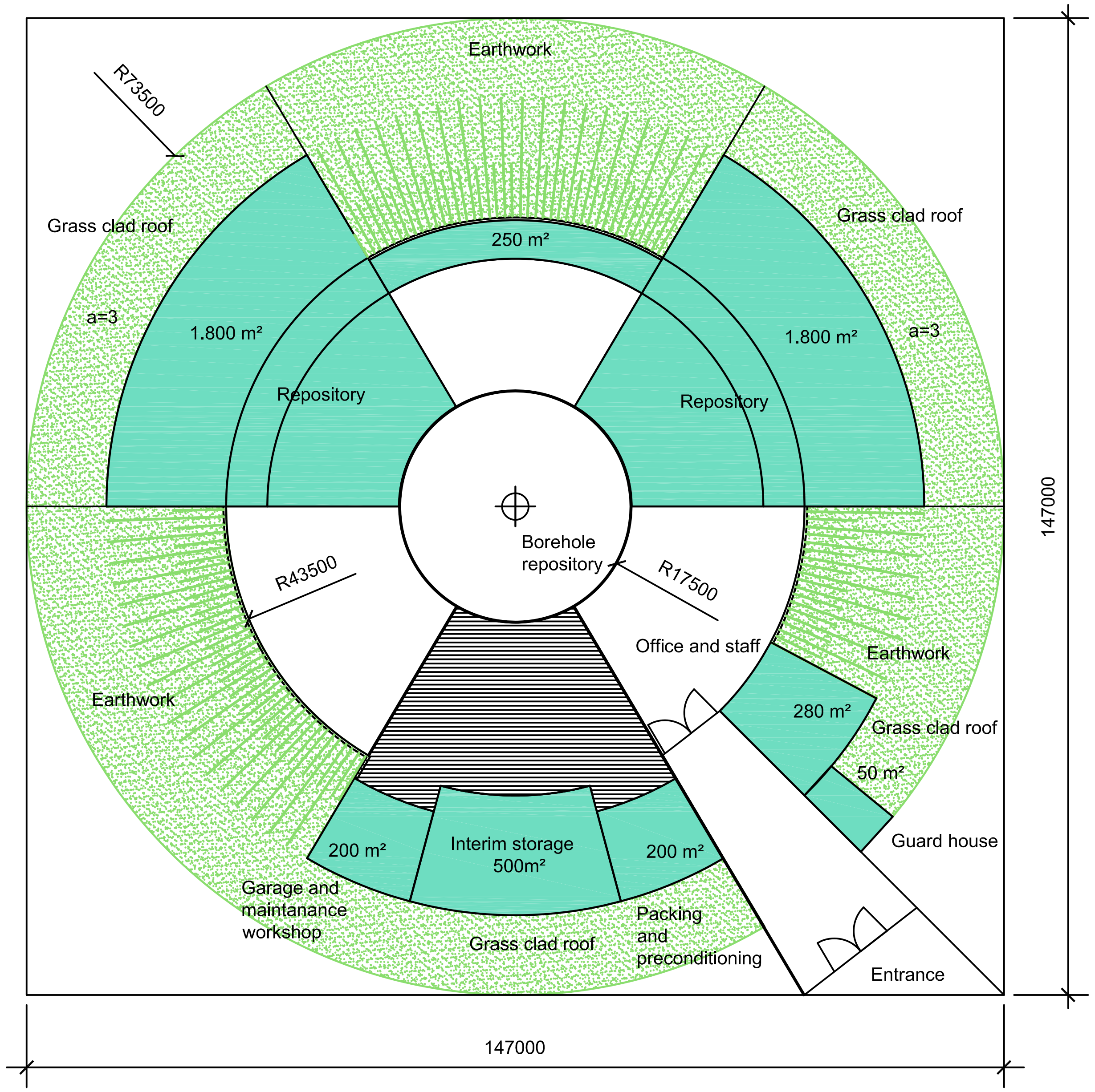


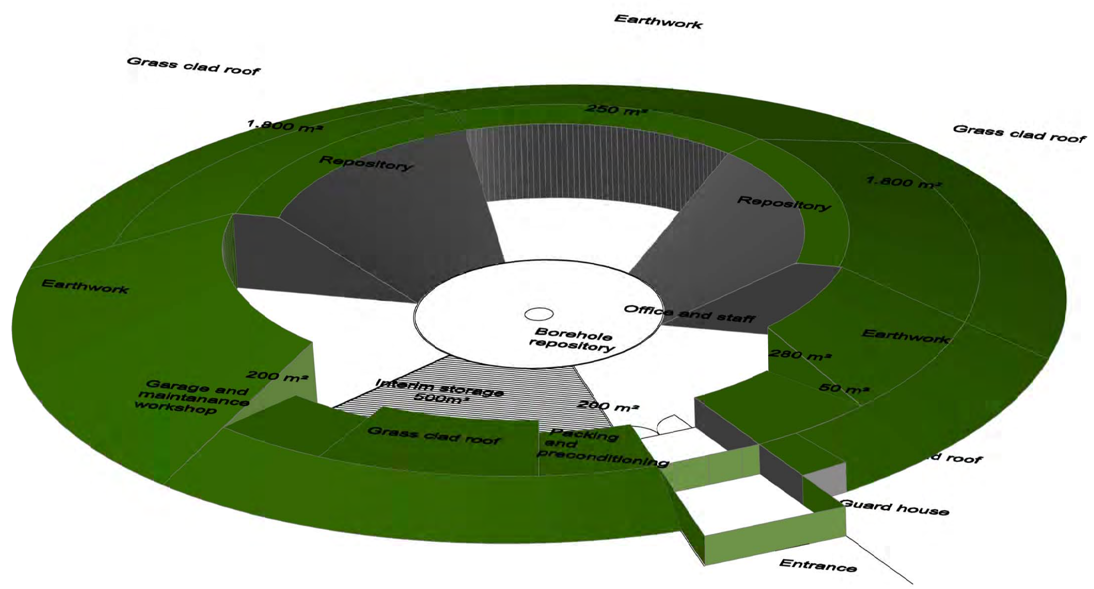


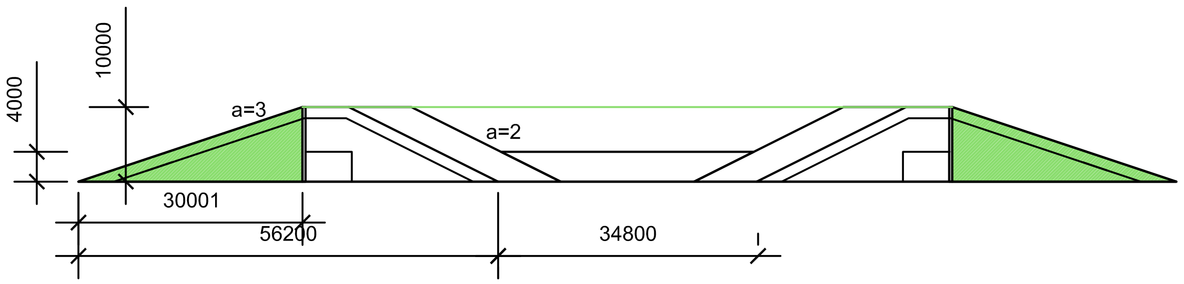
This draft is a variation of the basic model for the near surface repository. All building volumes are integrated, as part of the earthworks. The repository itself is divided into two parts in order to fit into the trefoil theme of the symbol.
The earthworks are enhanced in order to hide more of the facility and the inner courtyard is surrounded by a wall. This solution may be preferable in some types of landscapes.

Visual appearance of the repository in the landscape
Draft 02
Plan, cross section and isometric illustration
Date: 28.12.2010
Hasløv & Kjærsgaard / BB / BUL



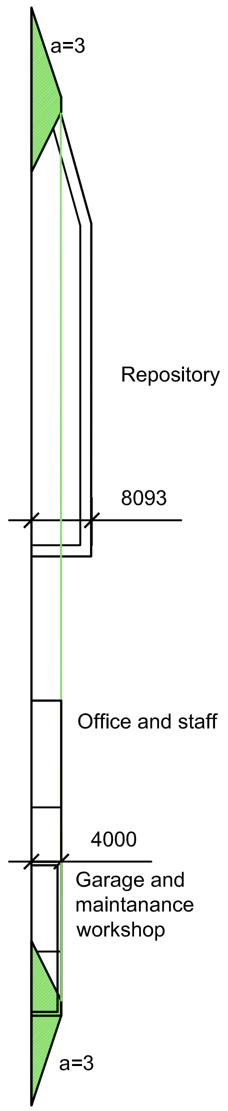


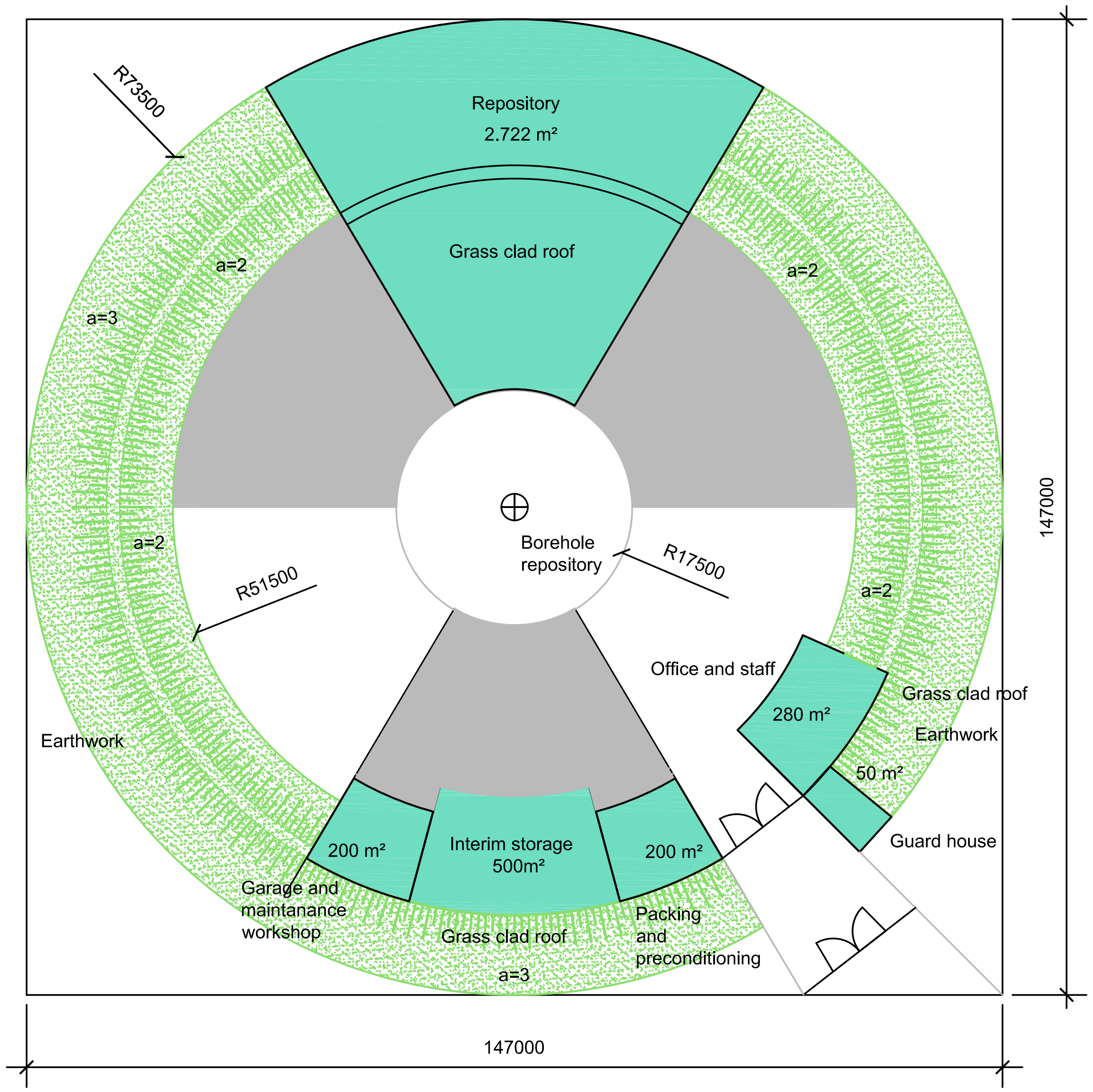


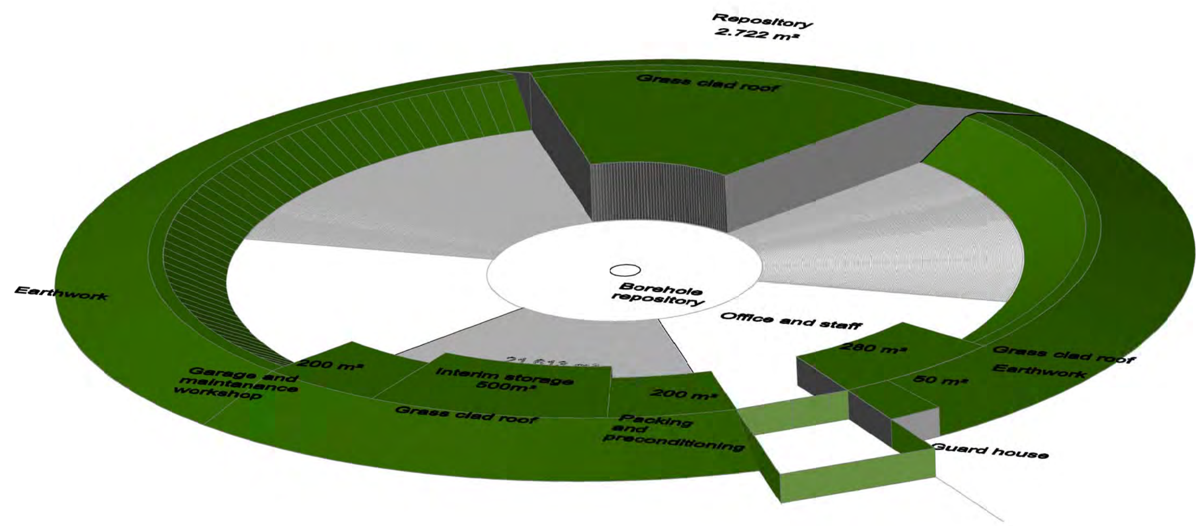


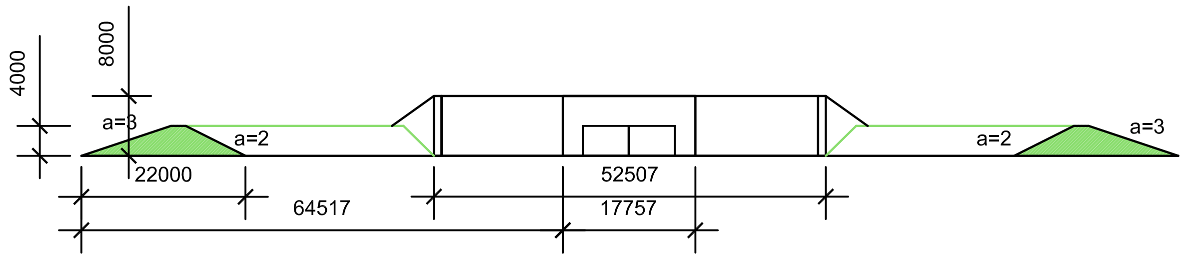
This draft is a variation of the basic model for the near surface repository. All building volumes are integrated, as part of the earthworks. The repository itself is divided into two parts in order to fit into the trefoil theme of the symbol.
The earthworks are enhanced in order to hide more of the facility and the inner courtyard is surrounded by a wall.
Furthermore the two parts of the repository is connected by a hallway.

Visual appearance of the repository in the landscape
Draft 03
Plan, cross section and isometric illustration
Date: 28.12.2010
Hasløv & Kjærsgaard / BB / BUL



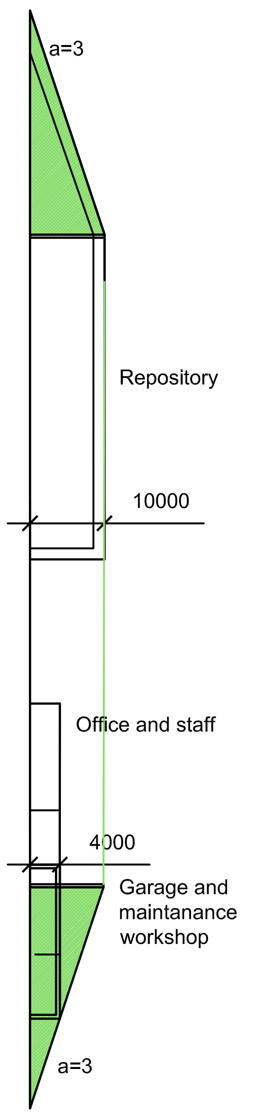


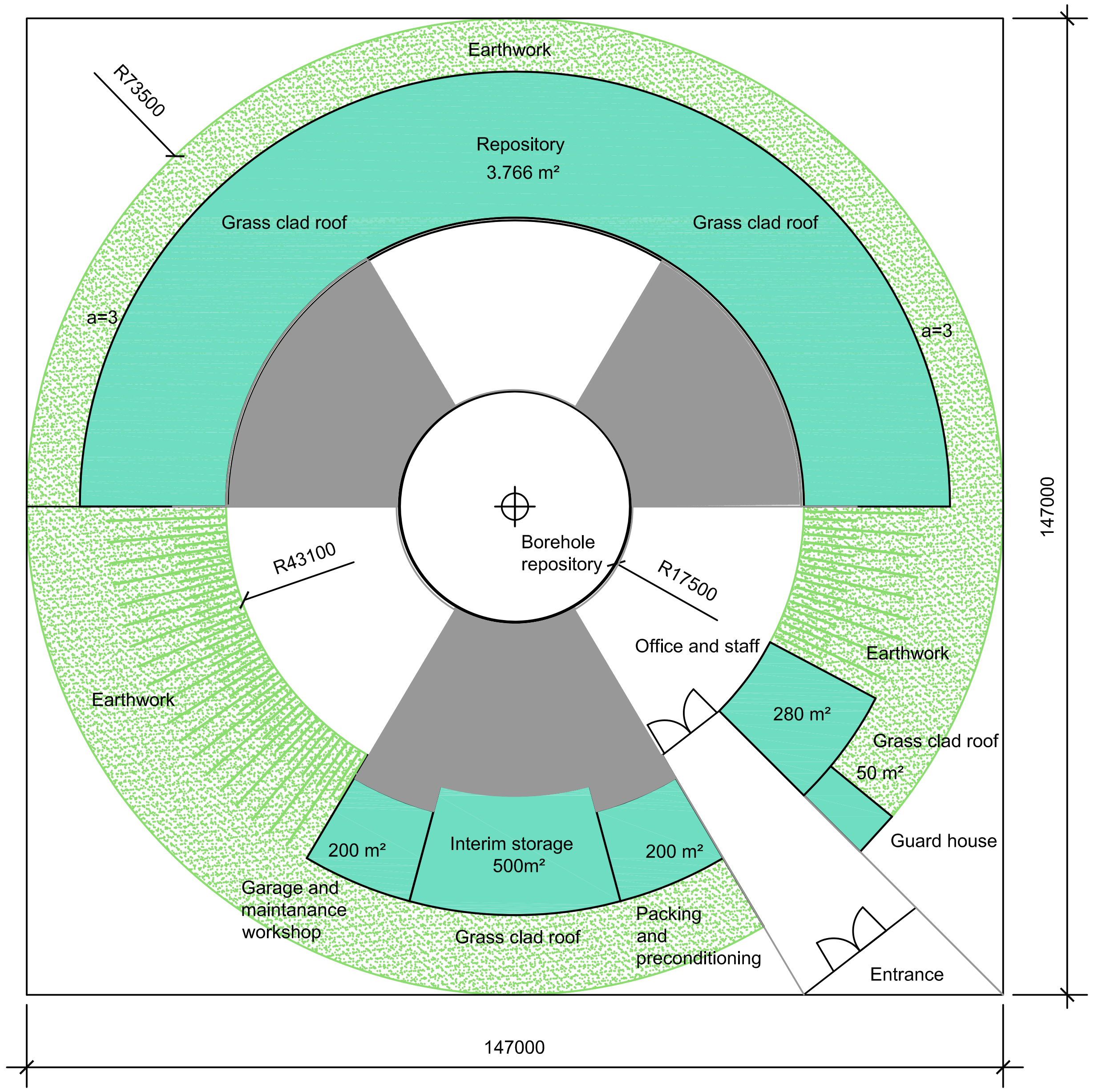


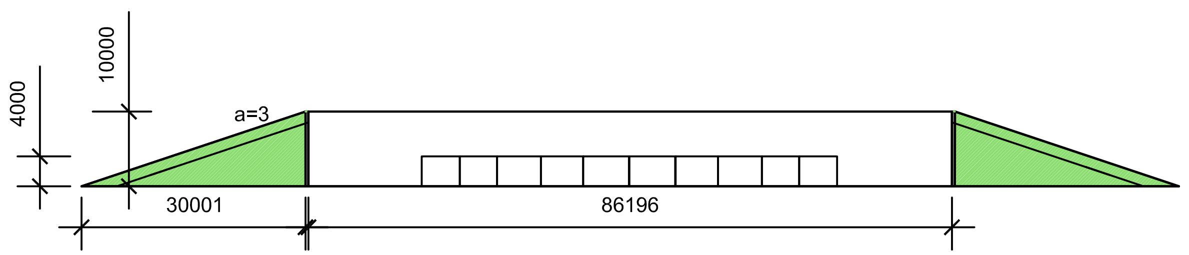


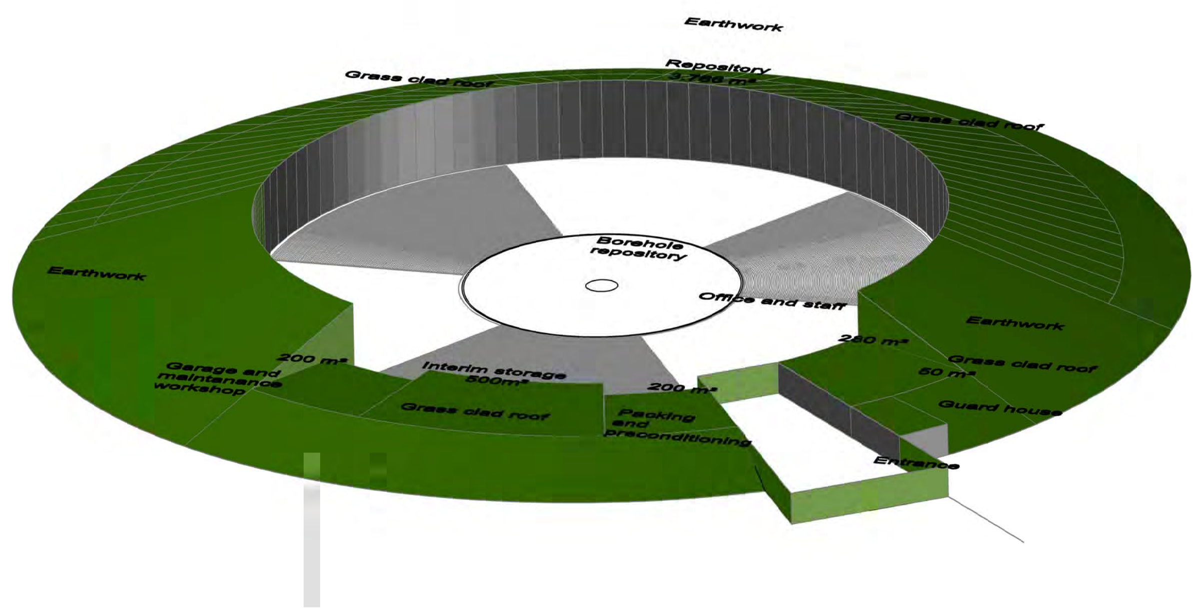
This draft is a variation of the basic model for the near surface repository. All building volumes are integrated, as part of the earthworks. The repository itself is united into one building volume. In order to fit into the trefoil theme of the symbol it fills one of the voids between the “petals”.
This also means that the area of the inner courtyard is larger and more versatile.
Placement of the borehole is not critical to the design.

Visual appearance of the repository in the landscape
Draft 04
Plan, cross section and isometric illustration
Date: 28.12.2010
Hasløv & Kjærsgaard / BB / BUL









In this draft all building volumes are totally integrated, as part of the earthworks itself. The repository is elongated and could consist of a large number of small compartments that could be sealed off one compartment at the time.
The earthworks are enhanced in order to hide more of the facility and the inner courtyard is surrounded by a wall.
This solution frees up the entire courtyard.

Visual appearance of the repository in the landscape
Draft 05
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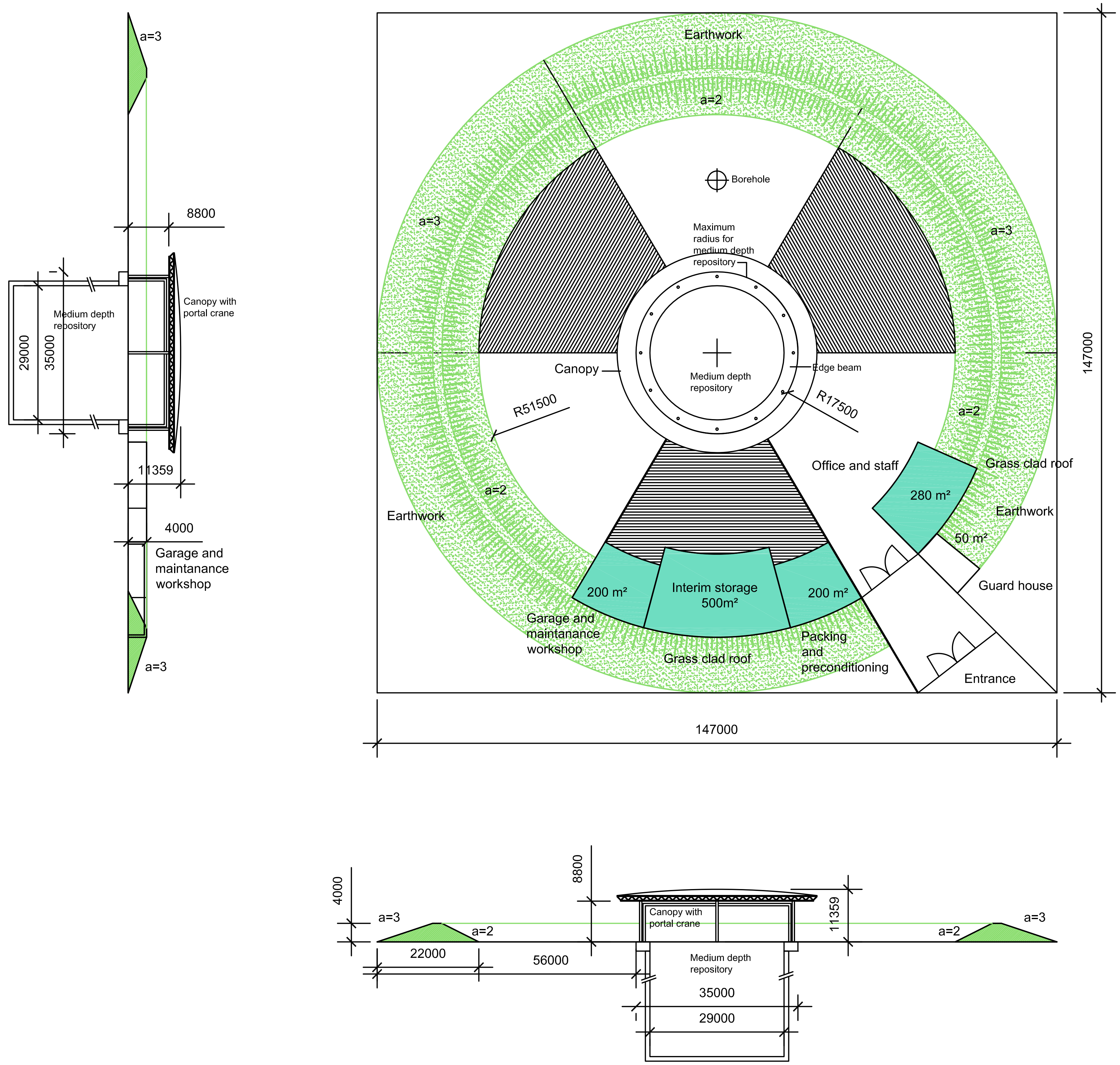

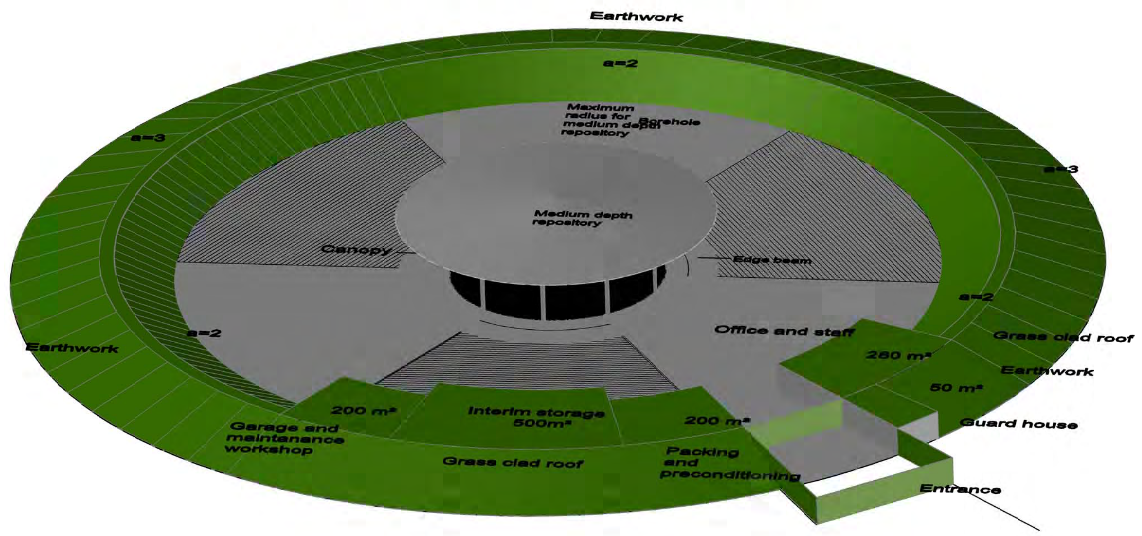

The medium depth repository requires a canopy or building volume over the pit, here placed formally as centre of the symbolic plan. All the smaller building volumes are integrated, as part of the earthworks.
The size of the repository itself is not determined, but all proposed, possible sizes can be integrated into this draft.

Visual appearance of the repository in the landscape
Draft 06
Plan, cross section and isometric illustration
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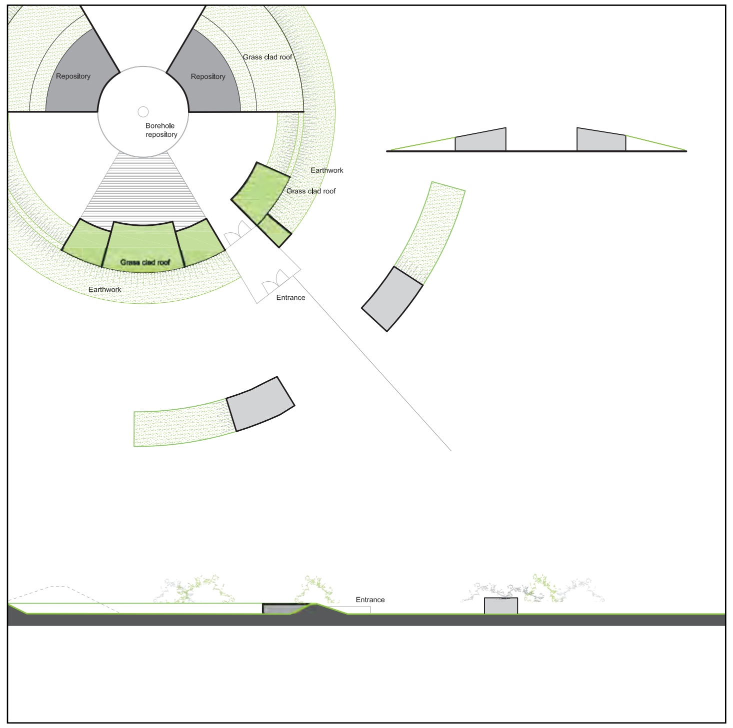

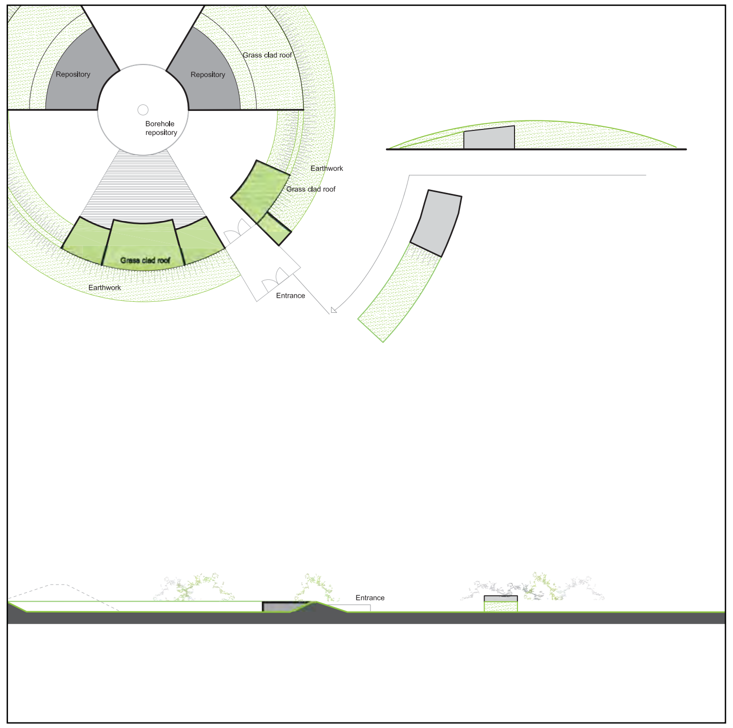

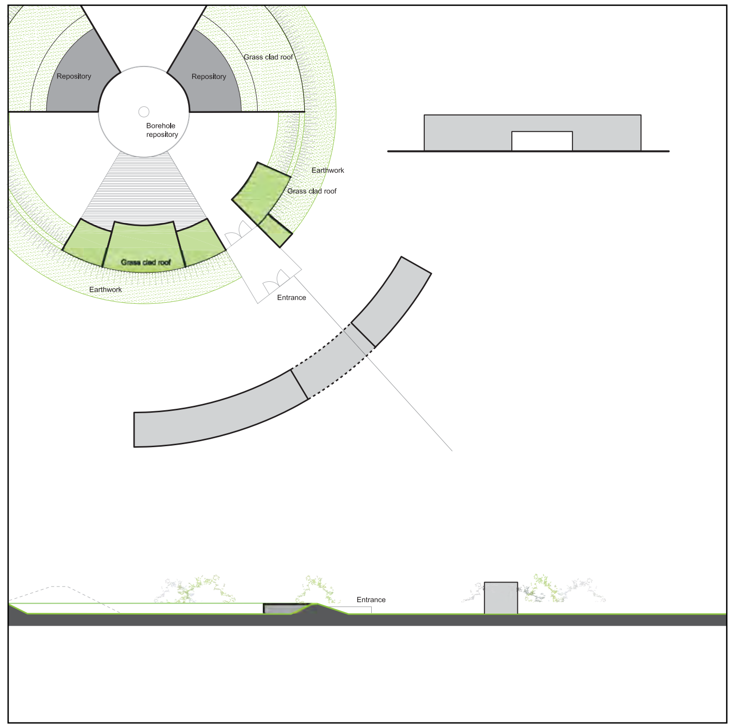

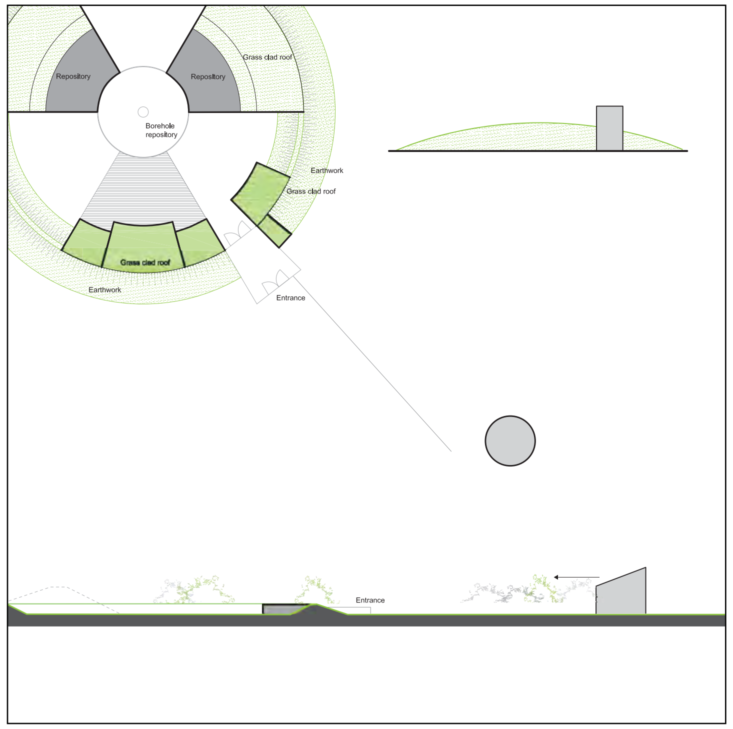

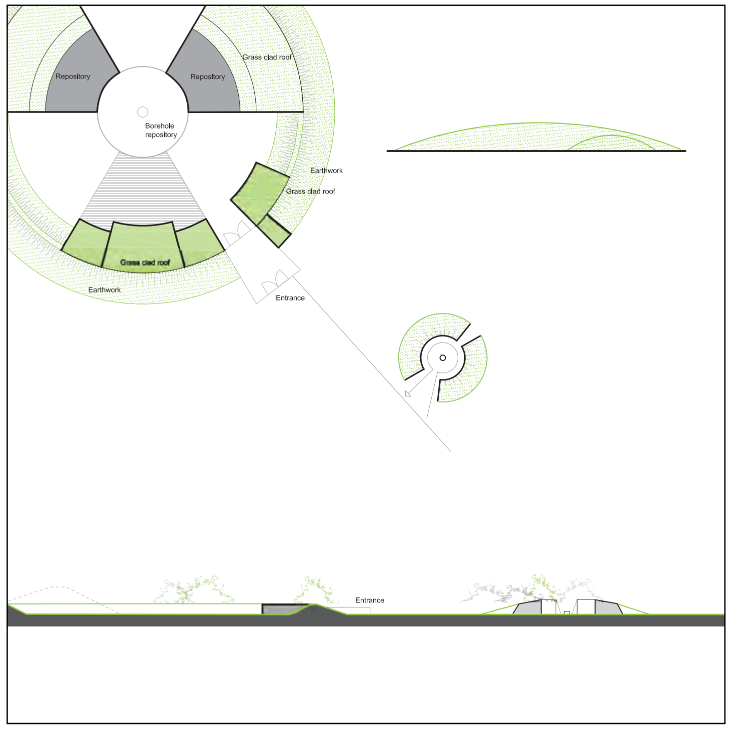

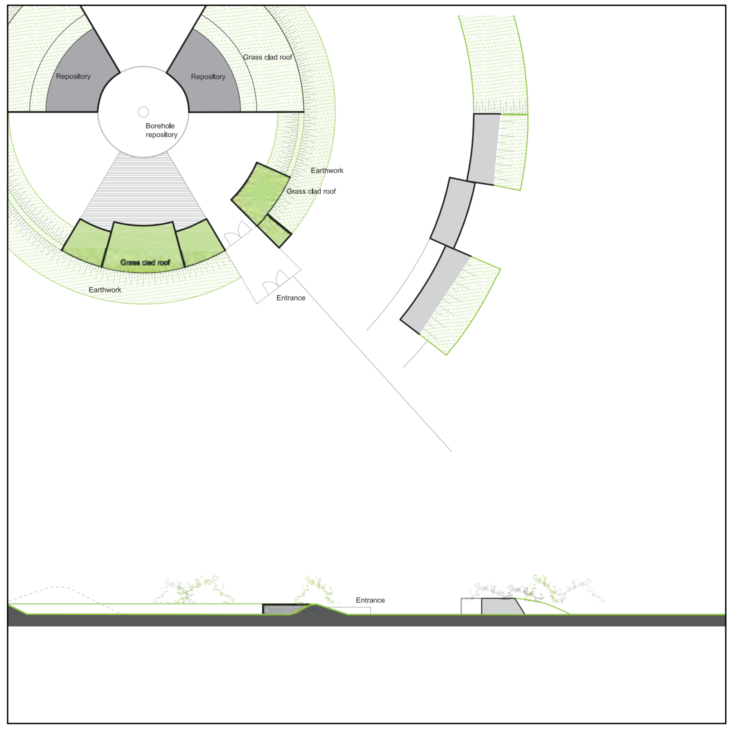

Visitors Centre and Treatment Plant

As a supplement to the repository it may be desirable to build a visitors centre with audio-visual capabilities for e.g. visiting school classes.
The building is to be of limited size, but big enough to facilitate the needs of up to around 25 persons.

Furthermore a treatment plant is needed. The spatial requirement is 2-300 m². The building encompasses facilities for solidifying, offices etc. Height is to be no less than 5 m for the workshop. Offices could be lower.

These facilities, and the repository itself needs parking space for the employees, delivery trucks and visitors.

It is proposed that all the above forms a supplementary part of the entire layout, outside the perimeter of the repository itself. It has to be integrated into the landscape witch will dictate the final design of the repository and the supplementary buildings.

Preferably the supplementary buildings forms a marked entrance, to some degree a portal or landmark, again dictated by the surrounding landscape and the preferred appearance to the outside environment.

To the left a number of suggestions of how the supplementary buildings can be added to the repository.