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Performance of the SKB Copper/Steel Canister

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Summary

The performance of the SKB copper/steel canister has been analyzed. The present knowledge of long-term function of the canister is summarized. Radionuclide release calculations for a reference failure scenario and the effect of some variations on release rates are shown. The Features, Events and Processes (FEPs) that are affecting the studied scenarios have been classified according to the "Rock Engineering Systems" methodology as defined by SKB for the copper/steel canister.

Radionuclide release rate is calculated for a reference failure scenario where a small hole in the weld of the outer copper overpack is assumed to exist at the time of deposition. The hole in the copper overpack is assumed to be of constant size until the inner steel canister loses its mechanical integrity. The steel is assumed to maintain mechanical stability during 5 000 years and after this time period the hole through the copper is assumed to be 0.1 m^2, which translate to insignificant transport resistance from the canister wall.

The release rates for C-14, Sr-90, I-129, Cs-137, Pu-239 and Am-241 are calculated for the reference failure scenario and for a number of variations. The variations include glaciation, only few of the Zircalloy tubes damaged, different canister filling materials, variations in sorption properties of the bentonite clay and different life-time of the inner steel canister. The performance of the canister and near-field, concerning the release rates of the studied radionuclides, is as expected, comparable to the release rates obtained in SKB 91.

Sammanfattning

Funktionen hos SKBs koppar/stålkapsel har analyserats och den nuvarande kunskapen om kapselns långtidsfunktion har sammanfattats. Beräkningar av radionuklidutsläpp för ett referensscenario för kapselgenombrott redovisas och effekterna på utsläppen av några variationer visas. De företeelser, händelser och processer (Features, Events and Processes, FEPs) som påverkar de studerade scenarierna har klassificerats enligt metodiken "Rock Engineering System" såsom den definierats av SKB för koppar/stålkapseln.

Utsläppstakten för radionuklider har beräknats för ett referensscenario i vilket ett litet hål i den yttre kopparkapselns svetsfog förutsätts finnas redan vid tidpunkten för placering av kapseln i förvaret. Hålet i kopparkapseln antages ha samma storlek så länge stålkapseln behåller sin mekaniska integritet. Stålet antages behålla sin mekaniska stabilitet i 5 000 år. Efter denna tid antages hålet genom kopparn vara 0.1 m^2, vilket innebär att transportmotståndet i kapselväggen då är obetydligt.

Utsläppstakterna för ^{14}C, ^{90}Sr, ^{129}I, ^{137}Cs, ^{239}Pu och ^{241}Am beräknas för referensscenariet för kapselgenombrott och för ett antal variationer. Variationer omfattar istider, endast några Zircalloyrör skadade, olika kapselfyllnadsmaterial, variation i sorptionsegenskaper i bentonitleran och olika livslängd för stålkapseln. Funktionen av kapsel och närzon beträffande utsläppstakter är, som förväntat, jämförbar med de utsläppstakter som erhölls i SKB 91.

Contents

1 Introduction

1.1 Background

In Sweden as in several other countries, spent nuclear fuel is considered to be disposed of in repositories in deep crystalline rock. The repository will be designed so that its safety will rely on multiple barriers. The barriers are usually defined as the engineered barriers, the waste itself, the canister, the buffer materials and the natural barriers, comprising the surrounding near-field and far-field rock. The properties of these barriers are such that a repository performing to expectations will delay the release of radionuclides for a time long enough to reduce the eventual release into the biosphere to levels well below the maximum release limits. However, in safety assessments one has to examine the consequences of less likely scenarios concerning repository performance. This is usually made by assuming that some of the canisters have an initial defect that will allow radionuclides to escape into the clay buffer around the canisters and host rock.

1.1.1 Aim of this report

The canister retaining the spent fuel is an important factor for the total safety of a repository. There have been several different canister designs and materials suggested for use in the Swedish final repository. In the KBS-3 study [*KBS, 1983*] two different canister designs were included, one Hot Isostatically Pressed (HIP) copper canister and a lead-filled copper canister. The current main alternative is the SKB copper/ steel canister constructed as a steel canister as the load-bearing part and has an outer corrosion resistant copper layer. This report describes the present knowledge concerning the performance of this canister regarding the long-term safety. The following subjects will be addressed:

- Internal and external processes that can jeopardize the integrity of the canister. For example gas production inside the steel canister, mechanical damage due to rock faulting, effects of steel corrosion products.
- Migration of gas produced by the corrosion of the steel of a damaged canister.
- Calculations of the release rate from a damaged canister into the far-field for a few scenarios and illustrations of the sensitivity for some important parameters.
- Connection between the studied cases and the process system for the copper/steel canister defined according to the "Rock Engineering Systems" methodology.

2 Near-field Description

The general layout of the near-field barriers that is the basis for the present report is the same as in the KBS-3 study [*KBS, 1983*]. The repository comprises of a horizonal tunnel system backfilled with a sand/bentonite mixture. In the bottom of the tunnels, the canisters with spent are emplaced in individual vertically drilled holes and embedded in bentonite clay, see Figure 2-1.

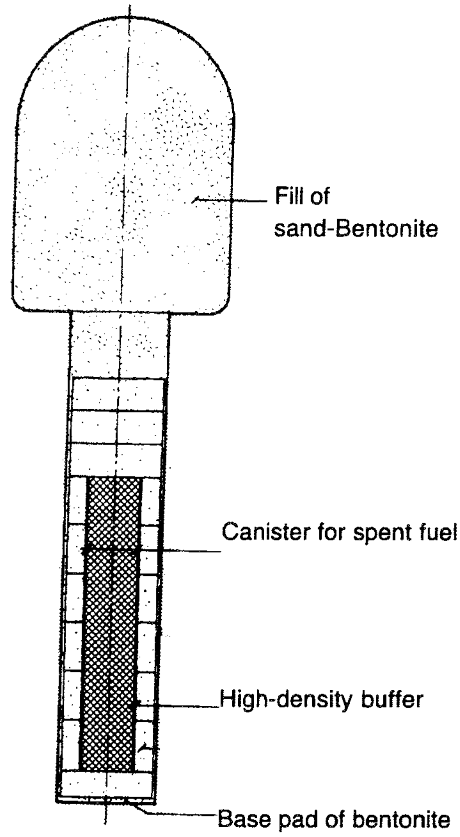

Figure 2-1. *Layout of the deposition hole with canister and bentonite barrier.*

2.1 The canister

The development of a canister for spent fuel that can be manufactured by a cold process was started by TVO in 1987. SKB got interested in the development of the copper/steel composite canister, by TVO, called the Advanced Cold Process (ACP) canister in 1989 and has worked with TVO in the evaluation of the long-term performance of this canister and near-field around it [*Werme, 1990*].

An advantage with the copper/steel canister is that one, in the manufacturing process, avoids the elevated temperatures around the fuel which is not the case in the manufacturing of the lead-filled canister and the solid copper canister (HIP). The cold process encapsulation process results likely in less risk for an accident resulting in the release of radionuclides during manufacturing.

To obtain a sufficient mechanical stability for the canister it is manufactured with an inner canister of carbon steel and an outer copper overpack. This design gives the canister good mechanical properties due to the steal and the corrosion resistance is supplied by the outer copper canister.

2.2 Canister design

The inner steel canister into which the spent fuel bundles are loaded is the structural support for the outer canister made of some high quality copper. An inert granular material will be used to fill some of the void of the inner canister to eliminate the possibility of development of criticality. The residual void after filling will be determined by the porosity of the selected filling material. The filling material has not yet been selected. The detailed design of the canister with dimensions is given in Figure 2-2 and some of the dimensions are also compiled in Table 2-1.

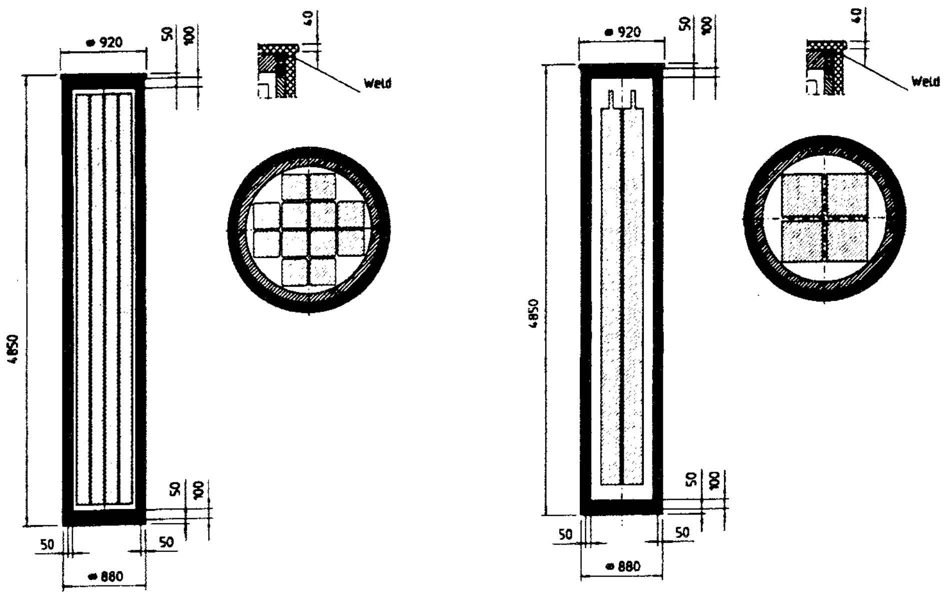

Figure 2-2. *Design of copper/steel canister.*

Table 2-1. Dimensions of the copper/steel canister. These dimensions are for a canister designed for intact fuel bundles.

Outer length	4.85 m
Inner length	4.5 m
Outer diameter	0.88 m
Copper thickness	0.05 m
Steel thickness	0.05 m
Void with fuel before filling	1.2 m^3
Void after filling (assumed ε=0.4)	0.48 m^3

2.3 Properties of the spent fuel

The fuel pellets consist of 95% UO_2 with other radionuclides, for example plutonium, imbedded in the uranium dioxide matrix. The release of these radionuclides is thereby limited by the dissolution rate of the UO_2. For radionuclides with very low solubility, the release rate can be even lower than determined by the dissolution of the fuel matrix. Some nuclides, like I-129, Sr-90 and Cs-137, may be enriched at the surface of the pellets, in this study 10% of the inventory iodine, strontium and cesium is assumed to be available for instantaneous dissolution in a penetrated canister. For C-14 100% of the inventory is assumed to be released instantaneously [*SKB, 1992*].

2.4 Bentonite Buffer

The function of the bentonite clay, placed around the canisters is to provide a suitable environment for the canister for a long time. SKB has chosen MX-80 Volclay as reference bentonite [*SKB, 1992*]. Blocks of compacted bentonite will be positioned around the canisters in the deposition holes. The bentonite will be compacted to give a density of about 2000 kg/m^3 after water-saturation and swelling.

The properties of the sodium bentonite clay that make it suitable as the buffer material are:

- low hydraulic conductivity
- rheological properties keeping the canister in its position in the deposition hole
- a plasticity protecting the canister from rock deformations
- sorption of many radionuclides
- thermal conductivity high enough to transfer the heat from the canister to the rock and thereby keeping the temperature in the clay from reaching above the desired level.

2.4.1 Water-saturation of the bentonite

The bentonite in the deposition holes and the sand/bentonite backfill in tunnels and shafts will take up water from the adjacent rock resulting in a swelling of the materials. After water saturation and swelling, the bentonite material will have a low hydraulic conductivity, which restricts groundwater flow in the material, and a high swelling pressure, which leads to homogenization and self-healing of the material. The initial density and composition of the bentonite material are important for the swelling pressure and hydraulic conductivity in the saturated bentonite material.

The time to reach full saturation of the bentonite surrounding the canisters depends on the initial water content in the bentonite, the amount of water supplied during the emplacement operation, the groundwater flow into the deposition zone from the adjacent rock, and to some extent on the heat produced by the radioactive decay of the waste in the canister.

2.4.2 Chemical alteration

Reactions with components in the groundwater, and with canister corrosion products may chemically alter the buffer material and thereby lead to changes in the physical and chemical properties. The thermal load in the buffer can also influence the buffer alteration. If the buffer material is a sodium-based bentonite, sodium will by ion-exchange be replaced by calcium from the groundwater. This will lead to some decrease of the plasticity and swelling ability of the material.

Copper dissolved during the corrosion of the canisters may interact with the surrounding bentonite barrier by ion-exchange where sodium in the smectite is replaced by copper [*Pusch and Börgesson, 1992*]. However, the solubility of copper in this environment is very low and consequently the effect of dissolved copper on the clay composition is expected to be very minor.

If the copper overpack is damaged, corrosion of the inner steel canister will result in the formation of solid iron oxide/hydroxide and dissolved iron, and in the generation of hydrogen gas. Dissolved iron in the pore water of the bentonite buffer may cause an ion-exchange between sodium and iron [*Pusch and Börgesson, 1992*]. If iron precipitates in the pores in the clay, the permeability of the clay will be reduced, but the precipitates will also have some cementing effect leading to reduced swelling ability. The properties of the sodium bentonite after water saturation was assumed to be as given in Table 2-2.

Table 2-2. Properties of water saturated bentonite clay.

Parameter	Value	Unit
Density	$2\ 10^3$	kg m^{-3}
Hydraulic conductivity	10^{-13}	m s^{-1}
swelling pressure	5	MPa

2.5 Properties of the surrounding rock

The properties of the rock, surrounding the deposition holes, will influence the release of radionuclides from a breached canister. When the breach i small, calculations indicate that the release of many radionuclides will be limited by the size of this hole. Since the Copper/Steel canister is expected to loose its load bearing capacity due to corrosion of the inner steel canister if there is an initial defect in the copper overpack the size and location of fractures intersecting a deposition hole will influence the release rates after the steel has lost its mechanical integrity. Similarly to the calculations in SKB 91 it is assumed that a fracture with flowing groundwater intersects the deposition hole with a damaged canister. The fracture is located at the same level as the breach in the canister to make the transport distance the shortest possible. The disturbed zone around the drift above the canister can also act as a sink for radionuclides and is therefore included in the model. Two more sinks of less importance are also included, a fracture zone below the deposition hole and a fracture/channel intersection the drift above the canister. It has been shown [*Romero et al., 1994*] that the dominating path from the bentonite is to a fracture intersection the deposition hole. This fracture is in this study assumed to be open around the full circumference of the deposition hole and have an aperture of 0.5 mm. The water flowrates used are set to 0.25 l/years for the fracture intersecting the deposition hole, 4 l/year for the disturbed zone, 0.5 l/year for the channel intersecting the deposition drift and 6 l/year for the fracture zone below the deposition hole.

3 Scenarios

The main objective of a scenario development is to make sure that relevant future eventualities are considered. The Features, Events and Processes (FEPs) that can influence the release of radionuclides from a repository should be reviewed and the interactions and combinations between FEPs must be investigated. The FEPS that envisioned to occur and influence the function of the repository is gathered in a so-called "Process System." The more unlikely or disruptive FEPS are classified as "External FEPs" and can initiate event chains in the Process System.

There are three alternative methods that are presently used by SKB to represent the system behaviour, event trees, interaction matrix diagrams based on the Rock Engineering Systems method (RES-methodology) and influence diagrams. The RES-methodology has been used by SKB to define the scenarios that are the basis for the release calculations in the present study.

The following sections of this chapter describe:

- Methodology
- Studied scenarios
- Important processes influencing canister performance

3.1 Rock Engineering System methodology

3.1.1 FEPs Identification

This section includes some of the consideration made by SKB prior to the selection of the model scenarios for the present study. The modelled scenarios are defined below in Section 3.2

The main objective of a scenario development is to make sure that all relevant future eventualities for a repository are properly considered. In other words, that all possible relevant Features, Events and Processes (FEPs) that could, directly or indirectly, influence the release and transport of radionuclides from the repository are reviewed and that relevant interactions and combinations between FEPs are investigated.

All FEPs that with some degree of determinism will occur during the relevant life time of a repository system are gathered in what we call a "Process System". More unlikely or disruptive FEPs are called "external events" and can themselves or in combination be put as initiating events for an event chain effecting the Process System. A Scenario will then constitute the total resulting event and consequence chain for a certain set of initiating FEPs.

To analyze a specific scenario, it is therefore important to have a systematic method to represent the total system behaviour. Presently SKB is investigating alternate ways to make this representation. The number of processes and interactions involved is great and the demand on the form of presentation and possibility for overview is high.

SKB has tried three main routes for the representation: 1) event trees, 2) influence diagrams and 3) interaction matrix diagrams according to Rock Engineering Systems methodology (RES-methodology). The influence diagrams are presently used in a study on long-lived low and intermediate level waste. The RES-methodology has been used internally for some test runs on parts of a repository system of KBS-3 type.

In the RES-methodology a square matrix is used for the presentation of the interactions in the process system. The most important subjects, for the purpose of the study, are placed along the leading diagonal (numbered from top left to bottom right corner). Interactions between subjects are shown as the off-diagonal elements. Elements located on the right-hand side of the leading diagonal, represent and interaction from the diagonal element with a lower number (closest to the upper left corner) to the subject with the higher number and vice versa. It is important to choose subjects for the diagonal that illustrate and explain the system behaviour. For an initiating event, an event chain can be followed through the matrix and the consequences for each subject can be followed along the path. The most important pathways through the matrix can easily be constructed. Doing this work all decisions must be documented to ensure that new knowledge or new views can be incorporated in an effective way.

Using the RES approach for this system, focusing on the copper/steel canister, the system parameters were identified and listed along the leading diagonal of a 11x11 matrix. For this specific purpose 11 main parameters were used, but the number of diagonal terms may be greater or lower depending on the purpose of the study. A large matrix adds resolution to the interaction, while a smaller matrix may be easier to work with. The leading diagonal terms chosen for this study are:

1. Fuel Rod Complex
   The spent fuel itself, the cladding and the metal parts of the fuel bundles.

2. Filler/Void
   The type of filling material has not been decided yet. The main alternative is a non-reactive material (quartz sand) that will bring down the void in the canister to about 40% to limit the risk for criticality. The canisters may, however, be empty or filled with something that may have beneficial properties. This parameter also included the void in the filling material.

3. Steel Canister
   The inner steel canister.

4. Gap Steel/Copper
   There will be a gap between the steel canister and the copper overpack for manufacturing reasons. This gap will be about one millimetre at deposition, but the copper overpack will creep on to the steel canister and the gap will disappear after some thousand years.

5. Copper Overpack
   The corrosion resistant copper overpack.

6. Temperature
   The whole near field will be subjected to elevated temperatures. The reason temperature is chosen as a parameter is that the temperature on the copper/buffer interface is a design requirement.

7. Buffer/Backfill
   This parameter consists of the bentonite in the deposition holes (with impurities) and the sand bentonite mixture in the tunnels.

8. Water Movement and Chemistry
   An important parameter that maybe should have been divided into two parts, one hydraulic and one chemical.

9. Fracturing in Rock
   This includes both the natural fracture system and the fractures induced by the construction of the repository.

10. Pressure
    The total pressure of the system: the hydrostatic pressure, the swelling pressure of the bentonite and the possible pressure increase from glaciation.

11. Construction of Repository and Emplacement of Canisters
    An extensive parameter that includes the design, construction, emplacement of canisters and closure of the repository. Reinforcements and stray materials are also included here.

The RES-matrix with the 11 diagonal elements include is shown in Figure 3-1.

1.1 Fuel Rod	1.2	1.3	1.4	1.5	1.6	1.7	1.8	1.9	1.10	1.11
2.1	2.2 Filler/Void	2.3	2.4	2.5	2.6	2.7	2.8	2.9	2.10	2.11
3.1	3.2	3.3 Steel Canister	3.4	3.5	3.6	3.7	3.8	3.9	3.10	3.11
4.1	4.2	4.3	4.4 Gap Fe/Cu	4.5	4.6	4.7	4.8	4.9	4.10	4.11
5.1	5.2	5.3	5.4	5.5 Cu Overpack	5.6	5.7	5.8	5.9	5.10	5.11
6.1	6.2	6.3	6.4	6.5	6.6 Temperature	6.7	6.8	6.9	6.10	6.11
7.1	7.2	7.3	7.4	7.5	7.6	7.7 Buffer/Backfill	7.8	7.9	7.10	7.11
8.1	8.2	8.3	8.4	8.5	8.6	8.7	8.8 Water Movemen	8.9	8.10	8.11
9.1	9.2	9.3	9.4	9.5	9.6	9.7	9.8	9.9 Fracturing	9.10	9.11
10.1	10.2	10.3	10.4	10.5	10.6	10.7	10.8	10.9	10.1 Pressure Constant Load	10.11
11.1	11.2	11.3	11.4	11.5	11.6	11.7	11.8	11.9	11.10	11.11 Constr. Emplac.

Figure 3-1. The SKB RES-matrix with 11 diagonal elements defined.

It should be noted that the order of the parameters is not important. As this selection clearly indicates, the resolution is high around the canister while the far field is treated in a more rough manner. The reason for this is, of course, that the canister is most important in this specific study.

FEPs (features, events and processes) can be identified via one (or potentially more) aspects to the matrix:

1. Parameters, P_i
2. Alterations to Parameters, P_i'
3. Interactions, I_{ij}
4. Alterations to Interactions, I_{ij}'
5. Pathways through the matrix M
6. Evolution of the whole Matrix, M'

Thus, a FEP can be a parameter in itself, although those related to a single parameter are best considered as alterations to parameters. It can also be a binary interaction between two parameters or an alteration of an interaction. Finally, a FEP could be a pathway through the matrix involving two parameters (loop) or more parameters in a pathway or loop - or evolution of the whole matrix through the cumulative effect of many consecutive and concurrent pathways. The list of FEPs can be found in Appendix I. This first list does not add any new important processes, not surprisingly, since it done from a present state of knowledge and needs to be reviewed by more expertise. All FEPs have to be judged for importance and given a proper treatment. References have to be added. The final result indicates that the RES approach is an excellent tool to get a complete and transparent picture of the processes that are affecting the canister. The method also simplifies the analysis of new scenarios, since the connections between all parameters and interactions are clear and well represented. In this study only the "Process system" and one initially defect canister have been considered. The RES approach has been used earlier for FEP identification for a full KBS-3 type system [*Stephansson and Hudson, 1993*].

3.2 Studied scenarios

In safety assessments, one usually defines a "reference scenario/case" or "central scenario/case" for the release calculations that seldom describes most probable function of the repository. In this study we call it the "reference failure scenario" to emphasize that the release calculations correspond to failure in the function of the canister not to the expected function.

3.2.1 Reference failure scenario

The following sections give the chain of events and processes defining the reference failure scenario and the basis for the modelled cases of the release of radionuclides if a canister is emplaced in the repository with an initial defect in the copper overpack are given below. Descriptions of important processes that influence the performance of the canister if there is a breach in the copper is given in Sections 3.3 and 3.4.

A defect in the weld of the copper canister is not detected in the quality testing and is placed in the repository. Water from the bentonite (8) pass through the hole in the weld into the gap between the copper and steel canister (4) and corrosion of steel canister is initiated (3). The corrosion can for a short period be aerobic but the main mechanism for the degradation of the steel will be anaerobic corrosion. Corrosion products and hydrogen are formed in the gap (4). The hydrogen accumulates in the gap both dissolved in the water and in gaseous form. Eventually hydrogen gas will be released through the hole in the copper into the bentonite (7) due to the pressure increase (10).

The corrosion of the steel will eventually result in a breach in the inner canister and finally cause a loss of mechanical integrity (3). When the steel canister is penetrated, water can enter and contact the fuel rod complex (1). As long as the Zircalloy tubes are intact the water and fuel are not in direct contact, so the dissolution of the fuel is delayed until the Zircalloy is also penetrated. When the dissolution of the fuel has started, the concentration of radionuclides in the canister void will increase and later be constant for some nuclides, while others will start to decrease because of decay or because the amount diffusion out from the canister in larger than that supplied by the dissolution of the fuel matrix.

Some fission products, like Cs, Sr, I and C, are accumulated on the grain boundaries of the fuel matrix. A fraction of these species also accumulates in the gap between the fuel and the Zircalloy cladding and can thereby be dissolved quickly after water contact the fuel pellets. The radionuclides embedded in the fuel matrix can be dissolved at a maximum rate controlled by the dissolution of the uranium dioxide.

The release of the radionuclides from the canister into the bentonite and subsequently to the host rock is dominated by diffusion. Since all other welds can be tested at canister manufacturing before the fuel is placed in the canister, we assume that the hole is located in the final weld at the top of the canister. Thus, it is not likely that the pressure buildup from the hydrogen gas generation can displace significant amounts of water from the canister.

3.2.2 Other studied variations

To investigate the sensitivity, for some of the model parameters on the release of radionuclides, the following variations changing initiating events or changing transport processes/parameters have also been studied:

Glaciation (variation V1)

- Glaciation at 50 000 years with rock movement resulting in a large breach in an initially intact canister and doubled water flow rate in the host rock.
- This scenario is an external event. The potential effects are alterations to Parameters 8 and 10 (change in water flowrate and external pressure), P_8',P_{10}'.

Zircalloy tubes (calculations made only for Sr and Cs, variations V2a and V2b)

- Only 1 tube damaged up to 5000 years
- 1% of the tubes damaged up to 5000 years
- No calculation of release after 5000 years is made since both Sr-90 and Cs-137 have decayed to insignificant levels at that time.
- These scenarios are alterations of Parameter 1, P_1'.

Filling material (variations V3a, V3b and V3c)

- Larger residual void inside when filling material is not used
- Stabilizing filling material that prevent the canister from collapsing
- Filling material with ability to weakly sorb Sr and Cs.
- These scenarios are alterations of Parameter 2, P_2'.
- Variation V3c occurs only when both the copper and the steel is penetrated and is a Pathway between Parameters 8, 1 and 2, M_{812}.

Immediate canister failure (variation V4)

- Large breach in canister immediately after deposition.
- This scenario has no scientific meaning and can therefore not be found in the RES-matrix.

Lower sorption for Plutonium (variation V5)

- K_d-value decreased ten times for Pu-239.
- This scenario occurs only when both the copper and the steel is penetrated and is a Pathway between Parameters 8, 1 and 7, M_{817}.

Lower corrosion rate for steel (variation V6)

- Collapse of steel canister after 300 000 years corresponding to a steel corrosion rate of 0.1 mm/year (reference 6.5 mm/year).
- This scenario occurs only the copper is penetrated and is a Pathway between Parameters 8, 4 and 3 M_{817}.

Smaller initial breach (variation V7)

- Since it is very unlikely that a hole as large as 5 mm^2 could pass through quality control, the release assuming an initial hole in canister 0.1 mm^2
- This scenario is a alteration of Parameter 5, P_5'

The input data used are the same as for the reference failure scenario for the parameters that are not mentioned in the description of the variations above. All radionuclides are not studied in every variation. For example, there is no need to perform variation V1 for the nuclides with short half-lives (Sr-90, Cs-137 and Am-241) since they have decayed to insignificant levels long before 50 00 years.

3.3 Processes occurring when canister is intact

3.3.1 Processes inside the canister

Some chemical reactions will start inside the canisters even if they are correctly constructed without any initial defects. Should air or water be trapped inside the canister, water, oxygen and nitrogen can result in limited corrosion of the steel and production of small amounts of nitric acid or ammonia. To eliminate any harmful effects from these reactions, the canister will be filled with a dry inert gas. However, it has been shown [Marsh, 1990] that if some water (10 g) and air (4 mols) is left in the canister the corrosion of the steel would be in on the order of 10 μm which is insignificant for the integrity of the canister.

The conversion to nitric acid is so slow that the water would be consumed by reacting with the iron [*Werme, 1990*]. If no oxygen is present inside the canister, the hydrogen produced from corrosion and by radiolysis of water can theoretically result in formation of ammonia. Since the amount of water and nitrogen available inside the canister will be small, no significant damage to the canister can be caused by this.

3.3.2 Processes outside the canister

The chemical processes that are active outside the canister and thus influencing the copper overpack is, as long as the canister is intact, practically identical to the HIP- or lead-filled canisters. Of the different copper corrosion mechanisms, both before and after repository closure, the most important is attack from sulphides dissolved in water. Sulphides are found in low concentrations in water at repository depth and some sulphide is also available in the bentonite clay and can be produced by microbial activity. The amount of sulphides, originating from the ground-water, which can be transported to the canister surface is less than 1 mg/year [*KBS-3, 1983*]. Calculations in [*Werme, 1990*] show that the corrosion due to sulphides transported with the ground water, from the bentonite and from microbial metabolism is less than 1 mm average corrosion of the copper over 100 000 years. Thus, the corrosion of copper with sulphide cannot have any effect on the integrity of the canister for millions of years. Corrosion of the copper by reaction with oxygen can only proceed until the oxygen left after the closure of the repository has been consumed. Oxygen originating from the surface is negligible at repository depths since oxygen dissolved in the infiltrating water is consumed by reaction with reducing agents close to the surface.

3.4 Processes occurring when canister is breached

3.4.1 Gas generated by corrosion of steel

If a canister is not intact, due to an initial defect or by another form of failure, water will be transported into the canister and initiate anaerobic corrosion of the inner steel container. The corrosion process will produce hydrogen gas that must escape from the canister by some means.

Since the canister and bentonite are designed to retain the content of the canister, the hydrogen gas cannot easily escape. Hydrogen can be transported by being dissolved in the water and migrate by diffusion and advection. The main transport mechanism for the dissolved hydrogen is diffusion in the bentonite and advection in the far-field. If the production of hydrogen is higher than the amount that can be dissolved the hydrogen must be released in gaseous form. The solubility of hydrogen at the anticipated pressure is so low that, with the expected gas production, transport in gaseous form will occur. [*Wikramaratna et al., 1994*]. The gas pressure will rise inside the canister since the production of hydrogen will initially be larger than the amount that can be transported out from the canister.

An assessment of the consequences of the generation of hydrogen gas inside the copper/steel canister has been made in [*Wikramaratna et al., 1994*]. In the study by Wikramaratna et al., it is assumed that one out of a thousand canisters has an initial breach in the copper with a size of 5 mm^2, corresponding to the 2.5 mm diameter of the beam used for welding. The size of the breach was assumed to grow, but since the actual growth rate is unknown, the effects of different breach sizes were assessed.

3.4.2 Gas generation rates and transport

The range of corrosion rates expected under repository conditions is 0.1 to 6.5 μm/yr [*Marsh, 1990*]. A release rate of 5 dm^3/yr hydrogen at 10 MPa was used for a corrosion rate of 6.5 μm/yr which was regarded as a realistic upper value for the corrosion rate in [*Werme, 1990*]. This corrosion rate is also used in the present study.

In [*Wikramaratna et al., 1994*] a constant value of 8 dm^3/yr at 10 MPa was used and calculations of capacity for advective transport of gaseous hydrogen through the bentonite by passage in capillaries, and by diffusion, was made. The conclusion of [*Wikramaratna et al., 1994*] is that the gas easily can be transported from the repository once outside the compacted bentonite.

3.4.3 Corrosion induced stresses

There has been concern that the corrosion of the steel canister could cause a breach in the copper to be enlarged. Because the metal oxide produced by the corrosion reaction occupies a greater volume than the metal, the corrosion of the steel will eventually lead to stresses being exerted on the copper canister. This has been studied by [*Hoch and Sharland, 1993*]. Their work considers the effect of an early failure of the copper overpack. It is assumed that there is a circumferential crack in the copper canister. The crack will allow water to flow into the annular gap between the steel and copper vessels, which will cause corrosion of the steel.

The model of the formation of corrosion product in the annular gap is that the steel will corrode uniformly around the crack at a constant rate until the gap is full of residue. Thereafter, the corrosion reactants and/or the charge carrying species must diffuse from the crack in the copper canister through the corrosion residue to the steel. The effect of this diffusion process on the corrosion reaction is modelled by assuming that corrosion will continue to occur at the same rate, but over a restricted area of the steel surface.

Two cases of aeration conditions outside the canister were considered, oxygen present and that oxygen is absent. Under aerobic conditions [*Hoch and Sharland, 1993*] assume that local corrosion occurs but that lateral increase of the corrosion pits is faster than the increase in depth and that the area of metal surface subjected to corrosion is increasing with time. After two years, which is the estimated time it will take the annular gap in the canister to fill with corrosion residue, it is assumed that the corrosion is uniform with a corrosion rate generating an increase in corrosion residue in the range 1-100 μm/year. Under anaerobic conditions uniform corrosion is also assumed to occur, and a corrosion rate of steel causing an increase in corrosion product of 0.1-1 μm/year [*Blackwood et al., 1993*] was used.

The stresses on the copper canister will increase until the thermodynamic limit of the corrosion reaction is reached. The maximum stresses on the copper canister occur when the steel is corroding aerobically. The amount of oxygen in the repository will decrease as the metals corrode, and the rate at which the stresses increase could fall by as much as two orders of magnitude when the corrosion is of the anaerobic form.

The diffusion calculation implies that for most combinations of aeration and saturation conditions in the repository, corrosion will take place over a significant fraction of the steel surface and the stresses on the copper will be distributed over this corrosion surface, and therefore the crack in the copper canister is unlikely to "yawn" [*Hoch and Sharland, 1993*]. The crack could, however, open if the corrosion is limited to the area close to the breach of the copper canister.

The steel surface over which corrosion will occur was investigated as a function of the crack width (10^{-4}-10^{-2} m), the diffusivity of the reactant and the concentration of corrosion reactant. The reactants are assumed to be oxygen under aerobic conditions and water under anaerobic conditions.

The crack in the copper canister was found not to "yawn" if the crack width is small and the diffusivity and the concentration of the reactants are large. A scenario that could lead to a widening of the crack in the copper canister occurs if the repository is saturated with ground water and the corrosion is aerobic. Such conditions exist for only a short time in the lifetime of the repository.

3.4.4 Criticality

A canister with 12 (low burn-up) fuel elements and no filling material around the fuel is a potential criticality risk. This problem must be solved before the canisters are deposited and is therefore not considered in this study, which only deals with the long-time safety. Studies are ongoing on the effects of different filling materials to bring down the void volume and on the effects of fuel burn-up.

4 Calculations of radionuclide release

4.1 Previous studies

A comprehensive study of the radionuclide release from a penetrated canister is given in [*Romero et al., 1994*]. In the study, a comparison of the release rates from the copper/steel canister and the lead-filled copper canister is made for different assumptions. In the modelling, the transport resistance due to the Zircalloy tubes around the fuel pellets in addition to the resistance from the canister wall, the bentonite buffer and the narrow openings of the rock fractures are considered. The model used in [*Romero et al., 1994*] is a compartment model, i.e., a model that calculates non-stationary release rates of the radionuclides. The same model is used in the present study.

The modelling results for I-129, Cs-137, Cs-135 and Pu-239 show that the dominant transport paths leaving the bentonite is to a fracture intersecting the deposition hole and to the disturbed zone around the drift. For one case, where the breach in the canister is assumed to be in the top of the canister and the closest fracture is located at half the height of the canister, the highest release is to the disturbed zone. For the cases where the breach is assumed to be in level with the fracture in the deposition hole the release to this fracture is the highest.

In previous performance assessments [*SKB, 1992*] the diffusion resistance of the canister interior has not been included. In [*Romero et al., 1994*] it is concluded that this resistance is important only when a small number of the Zircalloy tubes are perforated. If the major part of the tubes are damaged, the total resistance is dominated by the path through the hole in the canister wall.

4.2 Definition of studied cases

The mathematical modelling is in several aspects simplified compared to the description in Section 3.2. The reference failure scenario is in modelling terms defined as follows:

1. The release calculations are made for a single canister.

2. A small hole 5 mm^2 in the copper overpack permits water to enter inside the steel canister and the water intrusion initiate the dissolution of the fuel, i.e., no time delay is imposed from the steel canister or the cladding. No transport resistances from the Zircalloy tubes or in the filling material are included.

3. The corrosion of the steel is assumed to result in loss of mechanical integrity of the canister after 5000 years resulting in a large breach in both the steel and copper canisters. The steel is assumed to loose its load bearing capacity when the steel thickness has been reduced to 1/3. With a corrosion rate of 6.5 mm/year the time to failure for the steel will be about 5000 years. In the model, this means that the

transport resistance of the small hole through the canister wall is very important up to the collapse of the steel canister and insignificant thereafter.

4. Calculations are only made for a limited set of radionuclides:

Cs-137 and Sr-90

These are the two most important radionuclides at the time of deposition, because of their relatively large inventories and high specific activity. They are also considered rather mobile.

Am-241

Am-241 dominates the α-activity of the repository for the first hundreds of years. It is an interesting nuclide for illustrations of effect of variations because of its relatively short half-life. Its is, though, very unlikely that Am-241 ever will become a radiation hazard in the biosphere since the mobility, both in the near- and far field, is very low.

C-14

C-14 releases have appeared in some safety assessments. It is interesting to study the near field retardation, since its half-life is in the range where the near field may have an effect.

Pu-239

Pu-239 has the largest inventory of the actinides (except U) and is often considered one of the most hazardous nuclides even though its mobility is very low. Pu-239 can also be used as an analogue to Pu-240, which is not included in the calculations.

I-129

I-129 has been the main dose-contributor in most scenarios in all earlier performance assessment. It will not decay to any extent in the near field if the canister is defect, but it is important to calculate the maximum annual release.

Nuclides not included

This study focuses on the transient effects in the near field. The very long-lived nuclides (Pu-242, Np-237 and U + daughters) will always reach steady state and are therefore not included. Some potentially important fission are also missing, but the calculations on the selected set will demonstrate the process that are important for must nuclides.

The concentration inside the canister of the species with low solubility (Am and Pu) is determined by their own solubility and no reduction in concentration is made to reflect that more than one isotope in reality exists in the solution simultaneously.

4.3 Transport pathways from the canister

The application of the compartment model in this study is conservative by underestimating the delay of the ions diffusing through the bentonite barrier. Since there is only one compartment between the canister and the fracture intersecting the deposition hole, radionuclide release is detected earlier than to the disturbed zone where several compartments are passed in series. This selection of discretization somewhat overestimates the release of radionuclides with short half-lives (Sr-90 and Cs-137) and nuclides with high sorption (Pu-239 and Am-241). For the nuclides with low or no sorption (I-129 and C-14) this effect is of no importance.

In connection with the SKB 91 study, transient modelling of the release from the canister using a very fine discretization of the bentonite buffer was performed [*Bengtsson and Widén, 1991*]. This type of model calculations show which radionuclides that have a potential for decaying to insignificant levels during the diffusion through the bentonite. The results in [*Bengtsson and Widén, 1991*] show that the high sorption result in that Pu-239 and Am-241 to very large extent decay before leaving the bentonite buffer. Thus, the release calculated with the coarse discretization used with the compartment model will be especially conservative for these two nuclides. The high diffusivity and weak sorption used for cesium and strontium result in less difference between the two model concepts than for the more strongly sorbing species.

The four different release paths from the canister considered with the compartment model are:

- To a fracture intersecting the deposition hole
- To the disturbed zone around the drift above the canister
- To a channel intersecting the drift
- To a fracture zone below the canister deposition hole.

These four pathways are illustrated in Figure 4-1.

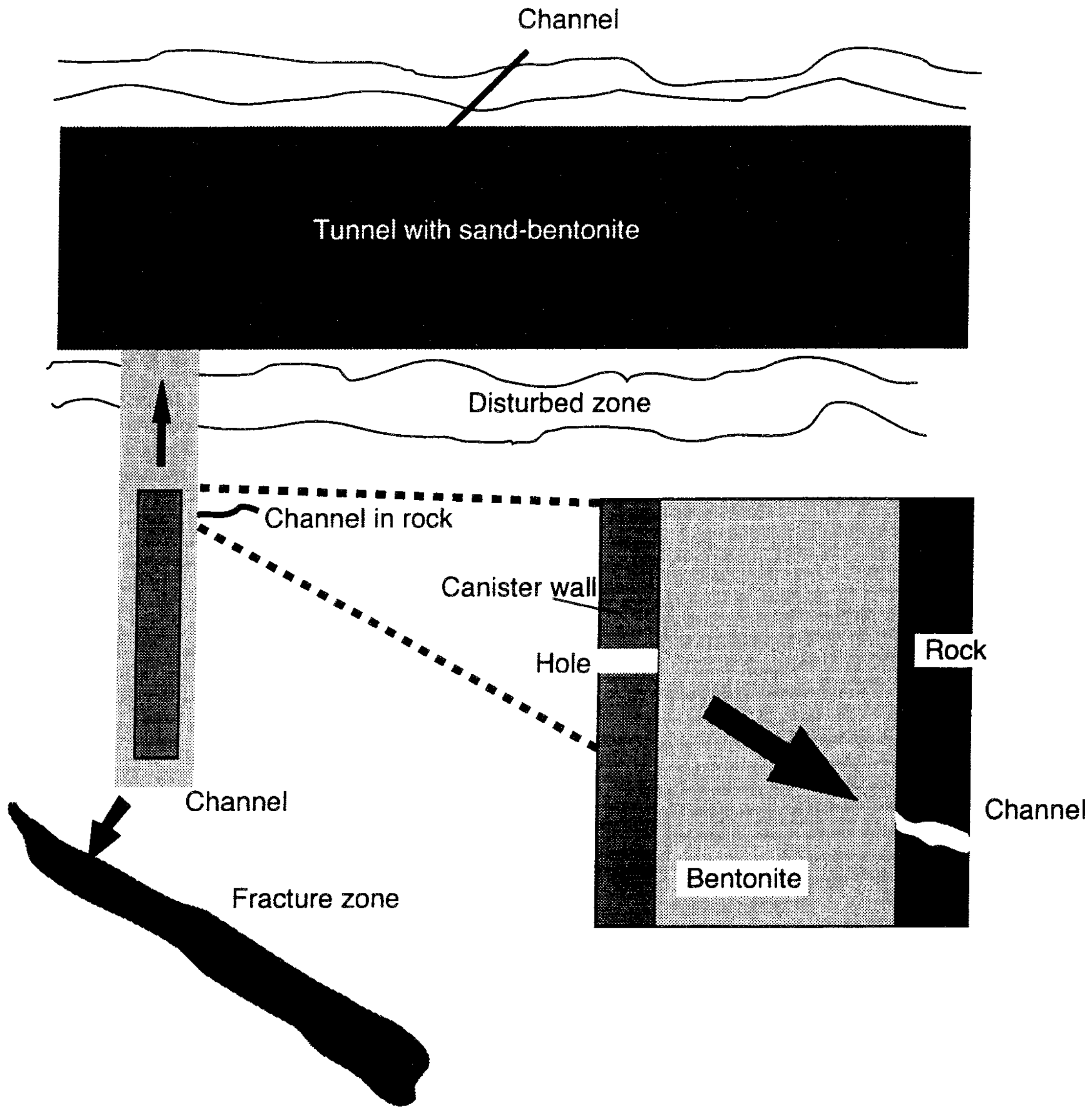

Figure 4-1. Illustration of release paths from a hole in the canister.

In the present report the same paths as above are considered but only the sum of the release from the different pathways into the far-field rock is presented.

4.4 Barrier dimension and properties

The assumed dimensions and properties of the canister and barriers in the present study are in close correspondence to those used in SKB 91 [*SKB, 1992*] and in [*Romero et al., 1994*]. The basic assumptions in the calculations are given in Tables 4-1 and 4-2. Data specific to the studied radionuclides are compiled in Tables 4-3 and 4-4.

Table 4-1. Geometry of important features considered in the near-field model. For data on the canister refer to Table 2-1.

Deposition Hole	diameter	1.5 m
	length	7.5 m
Drift	height	4 m
	width	3.4 m

Table 4-2. Properties of buffer, backfill and host rock.

Parameter	Bentonite	Sand-bentonite	Rock	Unit
Density	$2 \cdot 10^3$	$2.28 \cdot 10^3$	$2.7 \cdot 10^3$	kg m^{-3}
Porosity	0.25	0.24*	0.005	-

*This value is a bit low but the results are insensitive to this assumption in the present calculations.

4.5 Radionuclide inventory

The fuel in the present study is assumed to be of the same composition as the reference fuel used in SKB 91, BWR fuel with a burnup of 38 000 MWd/tU and a canister is assumed to contain 12 BWR assemblies. The inventory, at the starting time for the calculations is based on closure of the repository in year 2050.

Based on knowledge from previous safety assessment the following radionuclides have been selected for the release calculations, C-14, Sr-90, I-129, Cs-137, Pu-239 and Am-241. The amount present of the chosen nuclides, in year 2050 for one canister is compiled in Table 4-3.

Table 4-3. Inventory and half-life of the selected nuclides in one canister with 2.2 tonnes of uranium. Same fuel as in SKB 91 [*SKB, 1992*].

Nuclide	Inventory at year 2050 [Bq]	Half-life [years]
C-14	$5.85 \cdot 10^{10}$	5730
Sr-90	$1.36 \cdot 10^{16}$	28.8
I-129	$1.95 \cdot 10^{9}$	$1.57 \cdot 10^{7}$
Cs-137	$2.1 \cdot 10^{16}$	30.1
Pu-239	$1.63 \cdot 10^{13}$	$2.41 \cdot 10^{4}$
Am-241	$2.1 \cdot 10^{14}$	433

4.6 Nuclide and barrier specific data

The data used in the release calculations of the transport through the bentonite and the rock are the same as was used in SKB 91. In Table 4-4 the values on the effective diffusivity (D_e) in the bentonite and the K_d-values for both the bentonite and the rock are given. The values for the sand-bentonite mixture in the deposition drift is taken from [*Wiborgh and Lindgren, 1987*] and are also included in Table 4-4. The effective diffusivities in the sand-bentonite mixture in the deposition drift and in the rock is assumed to be 3.2 10^{-3} [m^2/a] and 3.2 10^{-6} [m^2/a] respectively for all nuclides. The results are insensitive to the diffusivity and sorption capacity of the sand-bentonite mixture and the rock since the dominating release path is by the fracture intersecting the deposition hole.

Table 4-4. Nuclide specific data used for bentonite buffer, sand-bentonite mixture in deposition drift and in the host rock. [*Brandberg and Skagius, 1991, and Wiborgh and Lindgren, 1987*].

Nuclide	Solubility [mol/m^3]	D_e [m^2/a] bentonite	K_d [m^3/kg] bentonite	K_d [m^3/kg] drift	K_d [m^3/kg] rock
C-14	high	$3.2 \cdot 10^{-3}$	0	0	0.001
Sr-90	1	$7.9 \cdot 10^{-1}$	0.01	0.032	0.015
I-129	high	$7.9 \cdot 10^{-5}$	0	0.0003*	0
Cs-137	high	$7.9 \cdot 10^{-1}$	0.05	0.025	0.15
Pu-239	$2 \cdot 10^{-5}$	$3.2 \cdot 10^{-3}$	50	0.1	0.2
Am-241	$2 \cdot 10^{-5}$	$3.2 \cdot 10^{-3}$	3	1.26	0.2

*A conservative value of 0 was used in the present calculations.

4.7 Radionuclide release from the near-field

For the selected radionuclides the releases for the different calculated cases described in Section 4-2 are presented in this section. The sum of the near-field release from the four different paths for a single canister is presented in Figure 4-2 for the reference failure scenario.

In Figure 4-2 the release of all studied radionuclides are shown in the same diagram for a comparison of the amount of activity contributed by each species.

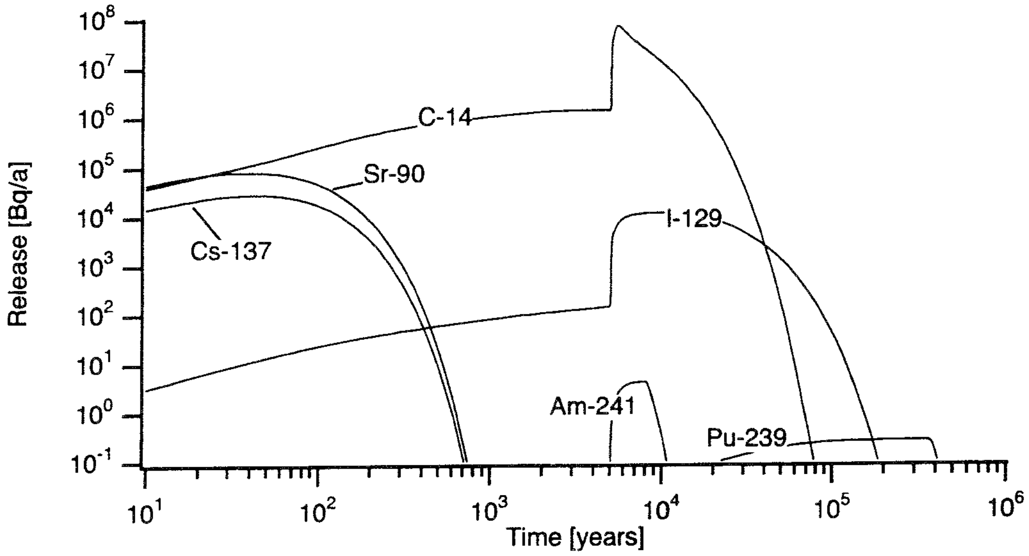

Figure 4-2. *Release from the near-field of Sr-90, Cs-137, I-129, Pu-239 and Am-241. Reference failure scenario.*

4.8 Sensitivity to parameter variations

To better illustrate the sensitivity to parameter variation, the following figures are constructed with only one nuclide in each diagram. Not all variations are applicable to all radionuclides. For example in the glaciation variation, where the canister is intact for 50 000 years before any release is initiated, Sr-90, Cs-137 and Am-241 have already decayed to insignificant levels.

The variation cases are indicated in Figures 4-3 to 4-8 according to the following codes:

V1 Canister intact from 0 to 50 000 years. Glaciation after 50 000 causing a large hole through the canister.

V2a Only one Zircalloy tube penetrated (only Sr-90 and Cs-137 studied).
V2b 1% of the Zircalloy tubes penetrated (only Sr-90 and Cs-137 studied)

V3a No filling inside steel canister
V3b Filling material with slight ability to sorb Cs and Sr
V3c Stabilizing filling material

V4 A large breach in the canister immediately after deposition.

V5 Kd-value decreased ten times for Pu-239.

V6 Collapse of steel canister after 300 000 years corresponding to a steel corrosion rate of 0.1 mm/year (reference 6.5 mm/year).

V7 Initial hole in canister 0.1 mm^2 instead of 5 mm^2.

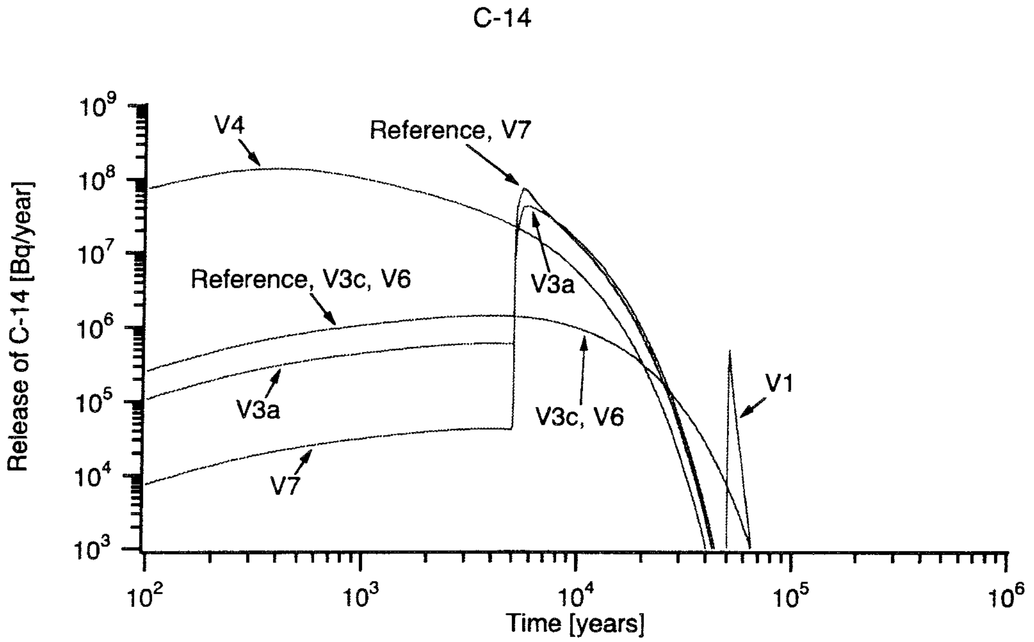

Figure 4-3. Release of C-14.

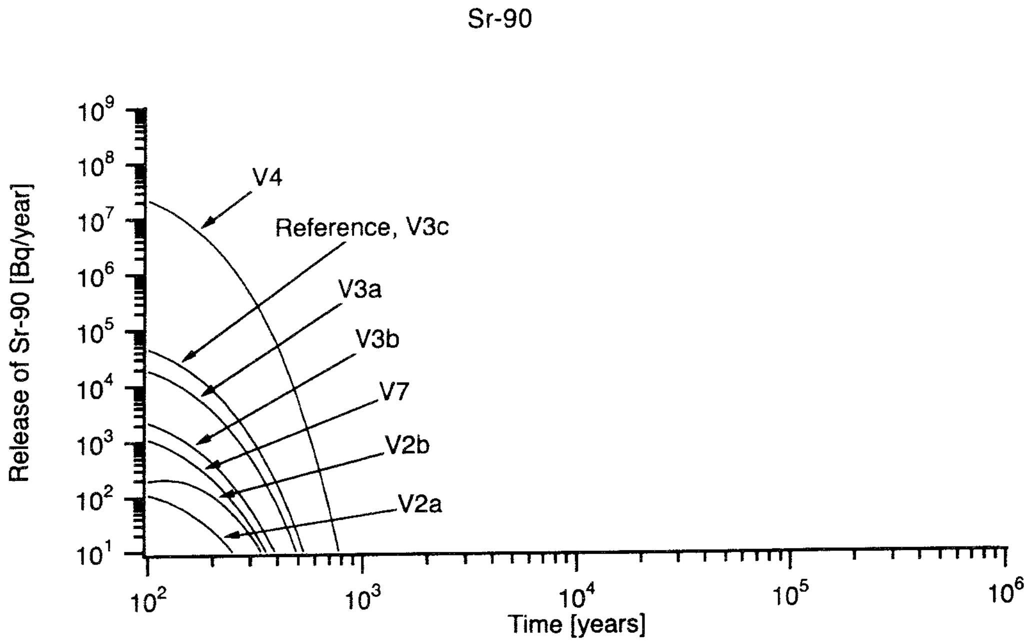

Figure 4-4. *Release of Sr-90.*

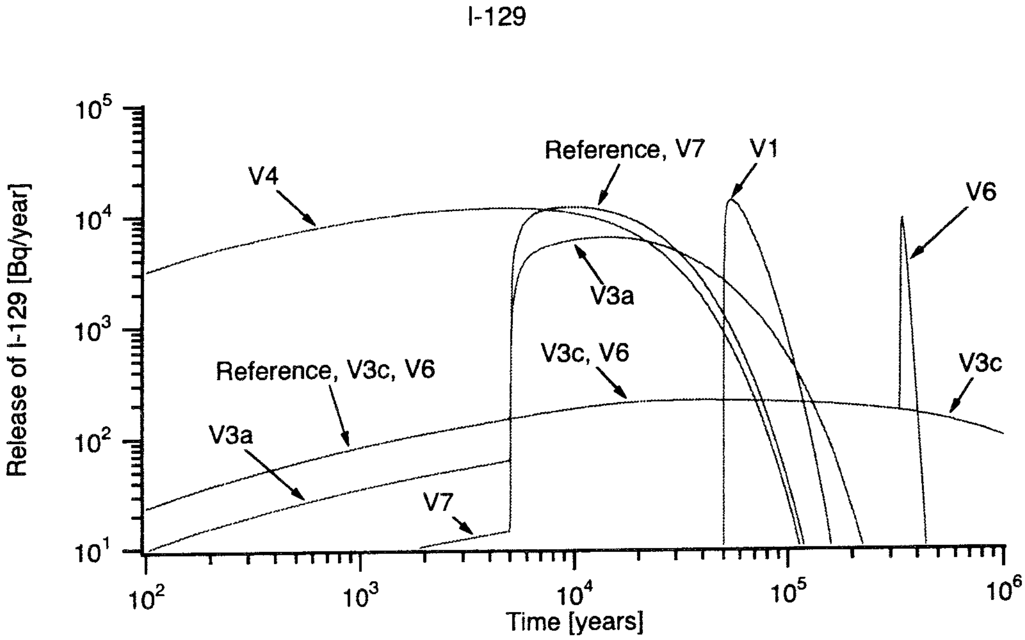

Figure 4-5. *Release of I-129.*

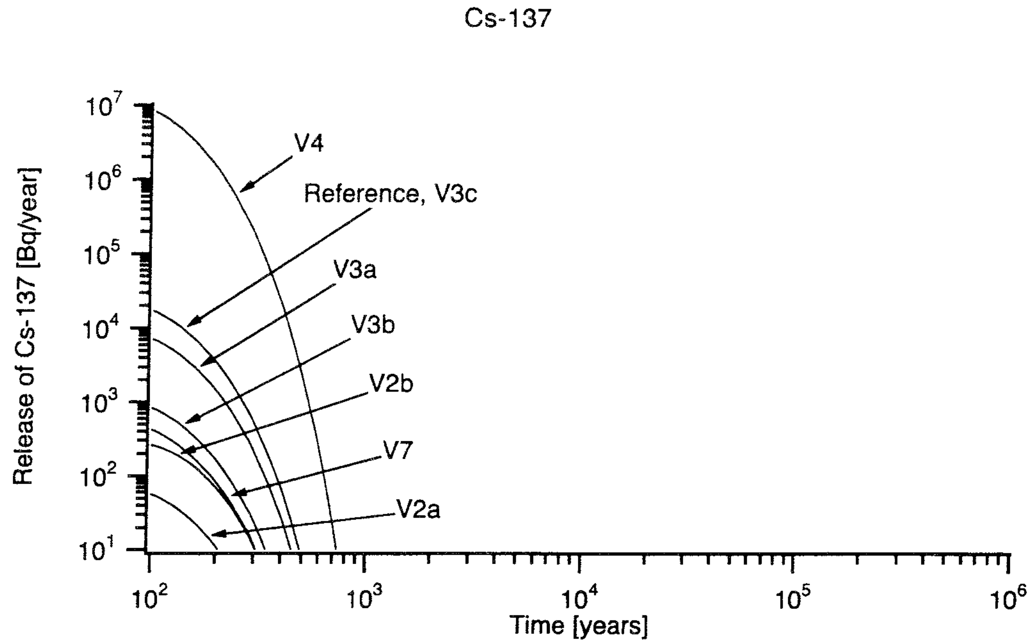

Figure 4-6. Release of Cs-137.

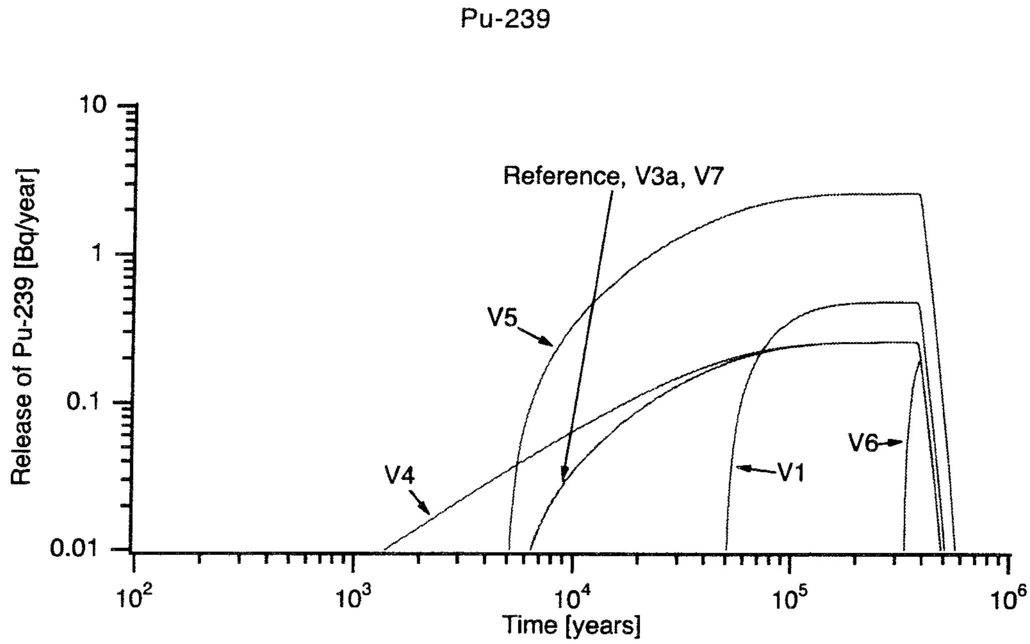

Figure 4-7. Release of Pu-239.

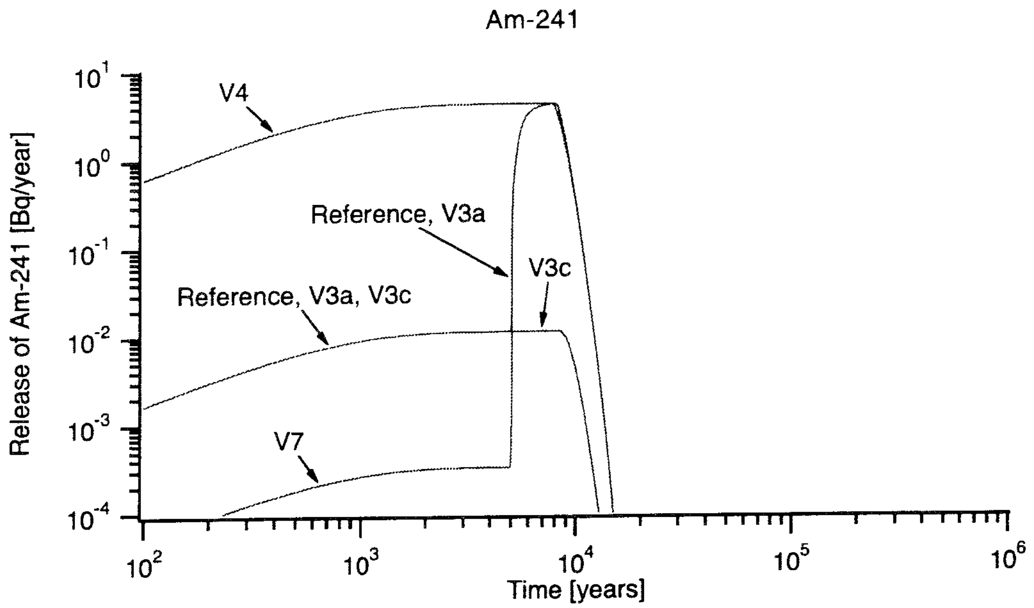

Figure 4-8. *Release of Am-241.*

4.9 Summary of modelling results

The maximum release rates and the time for the maximum release rate before and after collapse of the steel canister is compiled in Table 4-5.

Reference failure scenario

The overall release rates are similar to those obtained in SKB 91. C-14 is the dominating nuclide up to about 40 000 years after which the release of I-129 is higher due to the decay of the C-14. Both Am-241 and Pu-239 have very low release rates also after 5 000 years when the canister is assumed to collapse. The release rates for nuclides with high solubility are lowered in comparison to SKB 91 due to the larger available void volume inside the canister. This should not be taken as a clear advantage for the copper/steel canister over the lead-filled canister due to the fact that the transport resistance from the lead was disregarded in SKB 91 which in this regard was more conservative.

C-14

The differences for C-14 between the variations are as expected. The effect of the hole size is visible by comparing variations V4 (largest hole size), V6 (5 mm^2 hole size) and V7 (0.1 mm^2 hole size). The collapse of the steel canister cause a rapid increase in release rates but the peak after 50 000 years in case V1 is less pronounced since a large amount of the initially available C-14 inventory has decayed.

Sr-90

All variations except V4 give the same of lower release rates than the reference failure scenario. The lowest release rates is obtained for the variations with only a fraction of the Zircalloy tubes damaged. Using an empty canister give lower release rates than using an inert filling material as i shown by variation V3a but even a very slight sorption on the filling material (V3b) give a stronger effect than the additional void volume.

I-129

The long half-life and lack of sorption make the release fairly predictable. The effect of the canister and near-field regarding I-129 is to cause a distribution of the release over a large time-span and thereby reducing the peak dose. The maximum release rates are lower than those obtained in SKB 91 due to the larger void volume inside the canister. If, on the other hand the effect of the transport resistance from lead-filling had been included in SKB 91 the difference would be less or reversed.

Cs-137

The sensitivity to parameter variations is the same as for Sr-90 due to their similar properties for modelling purposes.

Pu-239

The effect of the hole size is obvious since the variations with an initially small hole is not even visible in the diagram starting at 0.001 Bq/year. For variation V4 with an initial large breach the maximum release rate occur so late that is coincides with re reference failure scenario. Variation V5, where a 10 times lower sorption capacity in the bentonite is used. This cause almost exactly a ten-fold increase in the release rates. The fast increase of the release rates after collapse of the canister is due to the course discretization of the bentonite between the canister and the fracture intersecting the deposition hole.

Am-241

The effect of the hole size the same as for Pu-239 with an instantaneous increase in release rate after 5 000 years. Variation V1 (glaciation) is not of interest since more than 100 half-lives have passed at 50 000 years.

Table 4-5. Maximum release rates from the near-field of the studied radionuclides. Grayed-out cells indicate that these are variations is not applicable (N/A).

Nuclide	Case	Time for max release, small hole [years]	Max release, small hole [Bq/year]	Time for max release, large hole [years]	Max release, small hole [Bq/year]
C-14	Reference	$3.8 \cdot 10^3$	$1.4 \cdot 10^6$	$5.3 \cdot 10^3$	$7.5 \cdot 10^7$
	V1	N/A	N/A	$5.1 \cdot 10^4$	$5.1 \cdot 10^5$
	V3a	$3.8 \cdot 10^3$	$6.0 \cdot 10^5$	$5.6 \cdot 10^3$	$4.3 \cdot 10^7$
	V3c	$3.8 \cdot 10^3$	$1.4 \cdot 10^6$	N/A	N/A
	V4	N/A	N/A	$3.8 \cdot 10^2$	$1.4 \cdot 10^8$
	V6	$3.8 \cdot 10^3$	$1.4 \cdot 10^6$	N/A	N/A
	V7	$3.8 \cdot 10^3$	$4.2 \cdot 10^4$	$5.3 \cdot 10^3$	$7.6 \cdot 10^7$
Sr-90	Reference	$3.9 \cdot 10^1$	$8.1 \cdot 10^4$	N/A	N/A
	V2a	$7.3 \cdot 10^1$	$1.3 \cdot 10^2$	N/A	N/A
	V2b	$1.2 \cdot 10^2$	$2.0 \cdot 10^2$	N/A	N/A
	V3a	3.9·101	$3.4 \cdot 10^4$	N/A	N/A
	V3b	$3.9 \cdot 10^1$	$4.0 \cdot 10^3$	N/A	N/A
	V3c	$3.9 \cdot 10^1$	$8.1 \cdot 10^4$	N/A	N/A
	V4	$3.5 \cdot 10^1$	$4.6 \cdot 10^7$	N/A	N/A
	V7	$3.9 \cdot 10^1$	$2.0 \cdot 10^3$	N/A	N/A
I-129	Reference	$5.0 \cdot 10^3$	$1.4 \cdot 10^2$	$9.5 \cdot 10^3$	$1.2 \cdot 10^4$
	V1	N/A	N/A	$5.4 \cdot 10^4$	$1.3 \cdot 10^4$
	V3a	$5.0 \cdot 10^3$	$6.0 \cdot 10^1$	$1.4 \cdot 10^4$	$6.1 \cdot 10^3$
	V3c	$3.6 \cdot 10^4$	$2.1 \cdot 10^2$	N/A	N/A
	V4	N/A	N/A	$4.5 \cdot 10^3$	$1.1 \cdot 10^4$
	V6	$3.6 \cdot 10^4$	$2.1 \cdot 10^2$	$3.3 \cdot 10^5$	$8.9 \cdot 10^3$
	V7	$5.0 \cdot 10^3$	$1.4 \cdot 10^1$	$9.5 \cdot 10^3$	$1.2 \cdot 10^4$
Cs-137	Reference	$4.3 \cdot 10^1$	$2.8 \cdot 10^4$	N/A	N/A
	V2a	$5.1 \cdot 10^1$	$8.6 \cdot 10^1$	N/A	N/A
	V2b	$7.1 \cdot 10^1$	$3.0 \cdot 10^2$	N/A	N/A
	V3a	$4.3 \cdot 10^1$	$1.2 \cdot 10^4$	N/A	N/A
	V3b	$4.3 \cdot 10^1$	$1.4 \cdot 10^3$	N/A	N/A
	V3c	$4.3 \cdot 10^1$	$2.8 \cdot 10^4$	N/A	N/A
	V4	N/A	N/A	$3.9 \cdot 10^1$	$1.6 \cdot 10^7$
	V7	$4.3 \cdot 10^1$	$7.0 \cdot 10^2$	N/A	N/A

Table 4-5. Continued.

Pu-239	Reference	$5.0 \cdot 10^{3}$	$9.0 \cdot 10^{-5}$	$3.7 \cdot 10^{5}$	$2.6 \cdot 10^{-1}$
	V1	N/A	N/A	$3.3 \cdot 10^{5}$	$4.8 \cdot 10^{-1}$
	V3a	$5.0 \cdot 10^{3}$	$9.0 \cdot 10^{-5}$	$3.7 \cdot 10^{5}$	$2.6 \cdot 10^{-1}$
	V3c	$5.2 \cdot 10^{5}$	$6.7 \cdot 10^{-4}$	N/A	N/A
	V4	N/A	N/A	$3.7 \cdot 10^{5}$	$2.6 \cdot 10^{-1}$
	V5	$5.0 \cdot 10^{3}$	$9.0 \cdot 10^{-4}$	$3.7 \cdot 10^{5}$	2.6
	V6	$3.3 \cdot 10^{5}$	$6.7 \cdot 10^{-4}$	$3.8 \cdot 10^{5}$	$2.0 \cdot 10^{-1}$
	V7	$5.0 \cdot 10^{3}$	$2.6 \cdot 10^{-6}$	$3.7 \cdot 10^{5}$	$2.6 \cdot 10^{-1}$
Am-241	Reference	$5.0 \cdot 10^{3}$	$1.1 \cdot 10^{-2}$	$8.0 \cdot 10^{3}$	4.2
	V3a	$5.0 \cdot 10^{3}$	$1.1 \cdot 10^{-2}$	$7.8 \cdot 10^{3}$	4.2
	V3c	$8.6 \cdot 10^{3}$	$1.1 \cdot 10^{-2}$	N/A	N/A
	V4	N/A	N/A	$8.0 \cdot 10^{3}$	4.3
	V7	$5.0 \cdot 10^{3}$	$3.3 \cdot 10^{-4}$	$8.0 \cdot 10^{3}$	4.2

It is difficult to make direct comparisons between the release rates obtained within the present study and SKB 91 due to differences in the assumptions. Since the lead filling is only limiting the release for the first 1 000 years in SKB 91 an important barrier function of the lead filled canister is underestimated. On the other hand, no sorption on the filling material in the present study is accounted for the reference failure scenario in the present study. With this in mind, the performance of the canister and near-field, concerning the release rates of the studied radionuclides is comparable to SKB 91.

5 References

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Appendix A

Release of radionuclides for individual cases

Release of radionuclides for individual cases

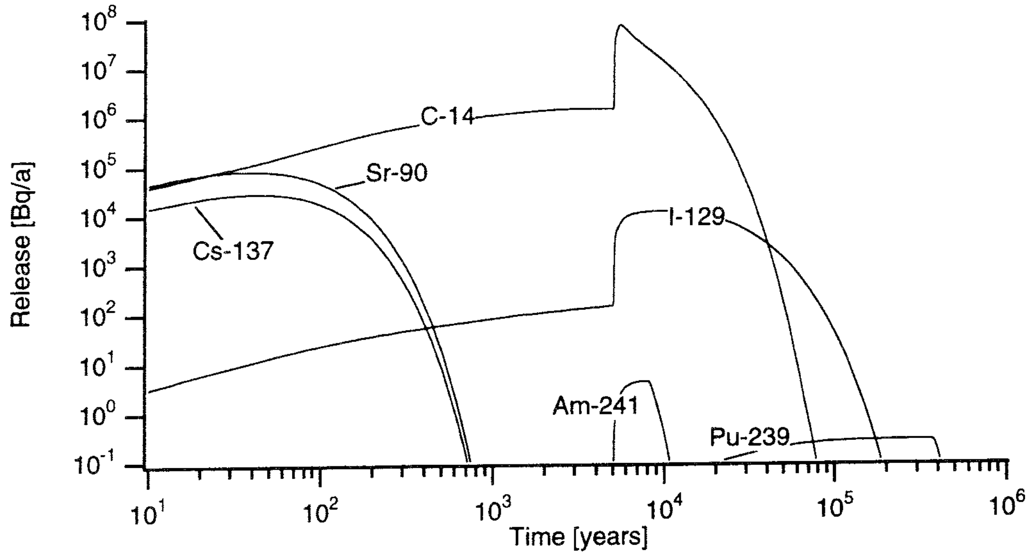

Figure A-1. Release from the near-field of Sr-90, Cs-137, I-129, Pu-239 and Am-241. Reference failure scenario.

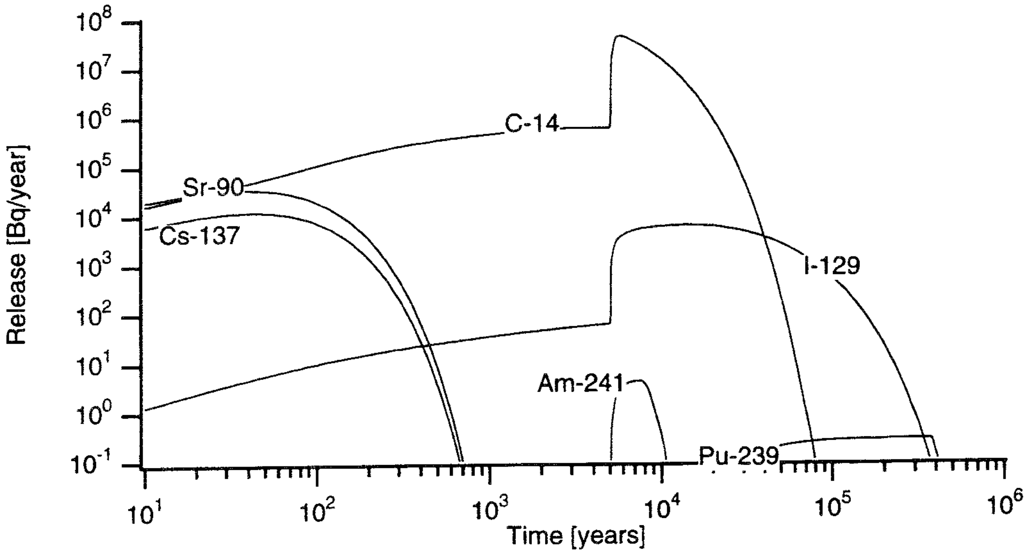

Figure A-2. Release from the near-field of Sr-90, Cs-137, I-129, Pu-239 and Am-241. No filling material. Variation V3a.

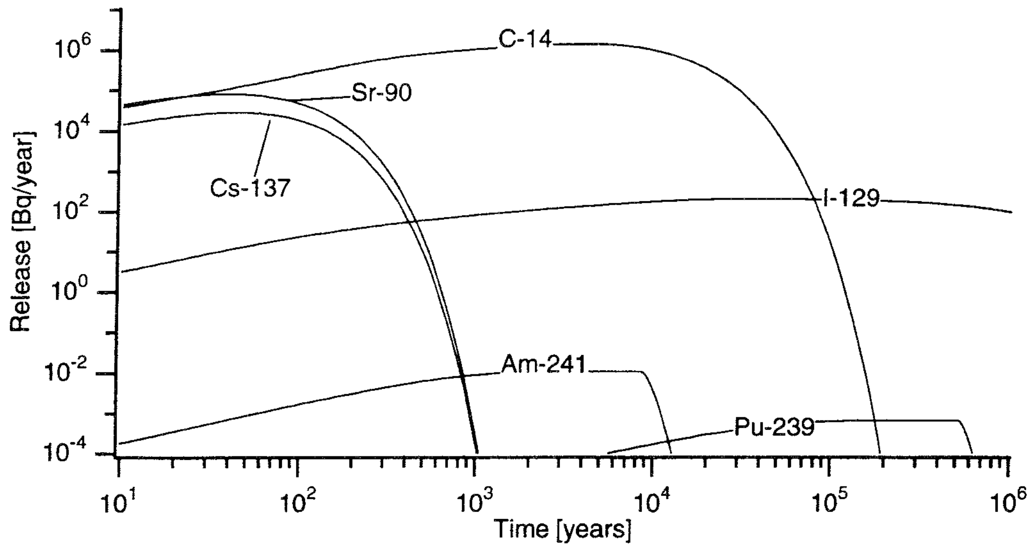

Figure A-3. *Release from the near-field of Sr-90, Cs-137, I-129, Pu-239 and Am-241. Stabilizing filling material. Variation V3c.*

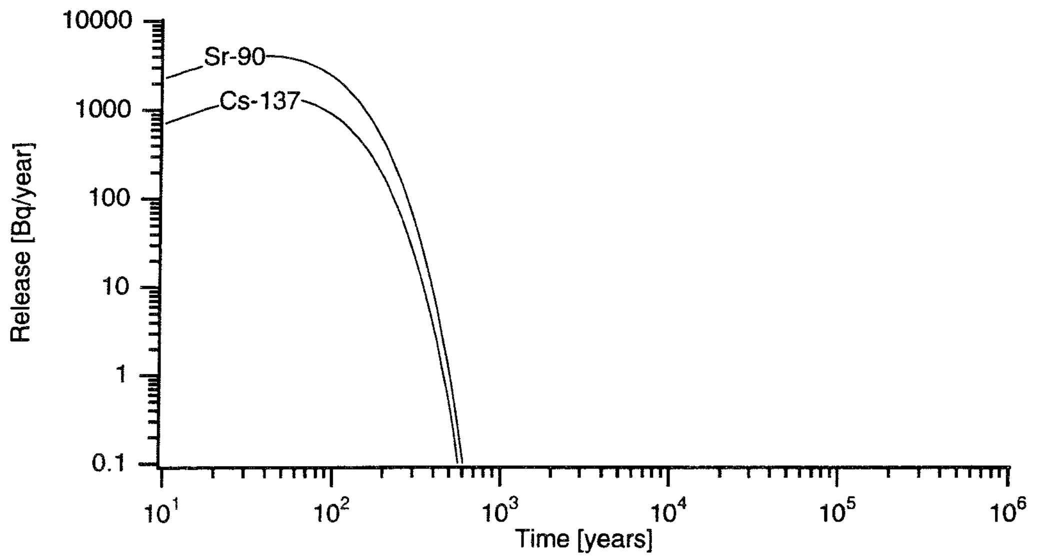

Figure A-4. *Release from the near-field of Sr-90 and Cs-137. Filling material with capacity to sorb Sr and Cs. Variation V3b.*

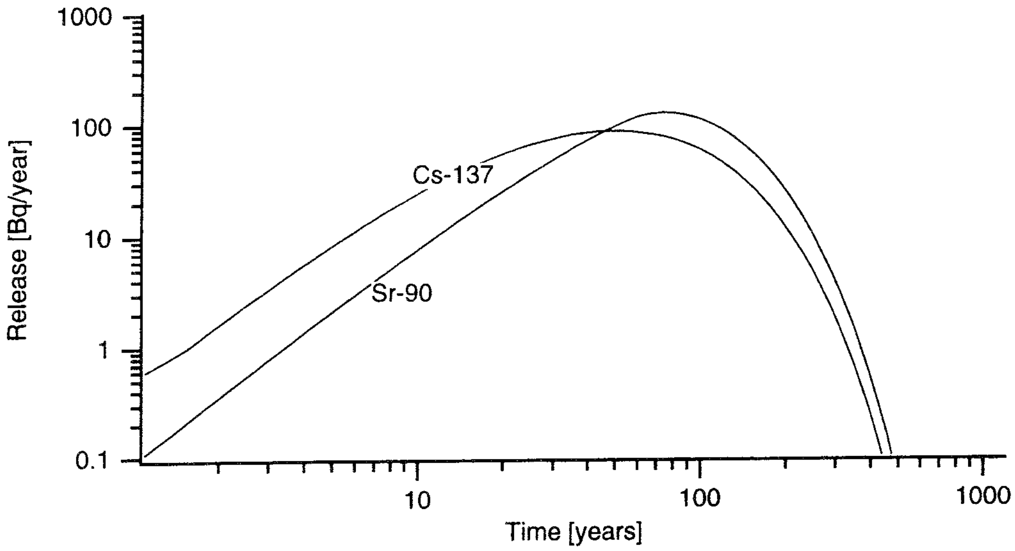

Figure A-5. *Release from the near-field of Sr-90 and Cs-137.Only one Zircaloy tube penetrated. Variation V2a.*

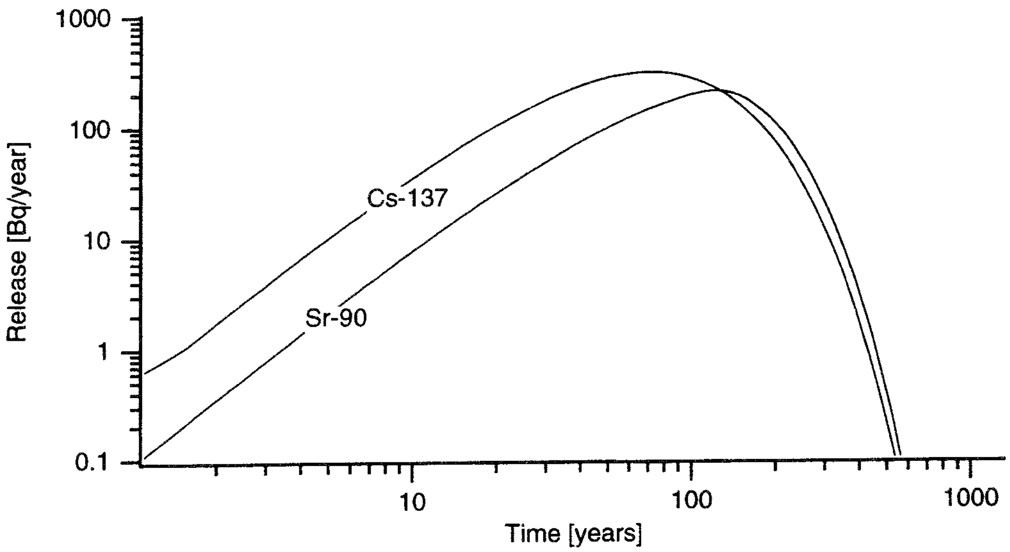

Figure A-6. *Release from the near-field of Sr-90 and Cs-137. 1% of the Zircaloy tubes penetrated. Variation V2b.*

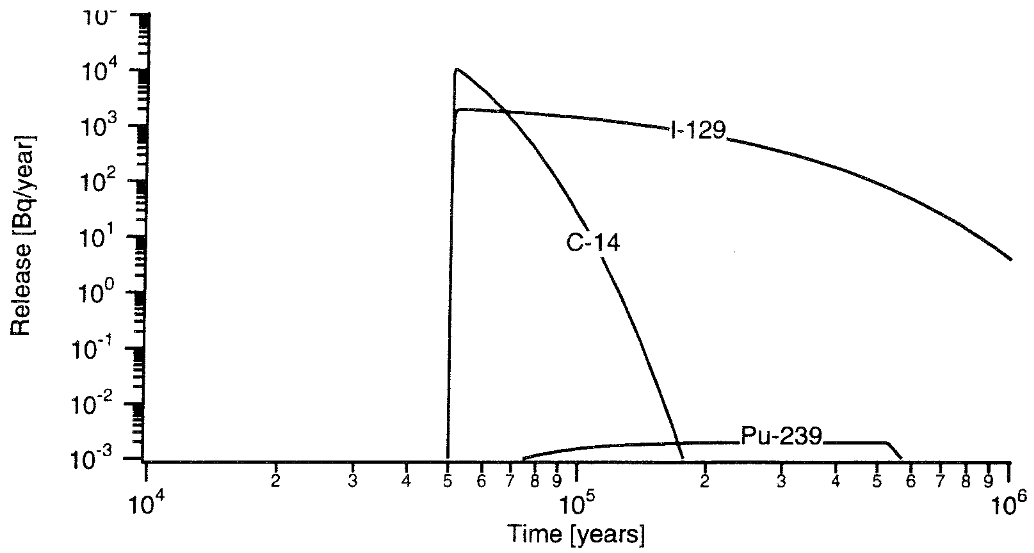

Figure A-7. *Release from the near-field of Sr-90, Cs-137, I-129, Pu-239 and Am-241. Canister collapse after a glaciation ending at 50 000 years and doubled water flowrate. Variation V1.*

Appendix B

Immediate canister failure

Immediate canister failure

The radionuclide release calculated assuming negligible transport resistance from the canister directly after deposition is shown in Figure A-1. All data used is identical to the reference failure scenario except regarding the time for collapse of the inner steel canister.

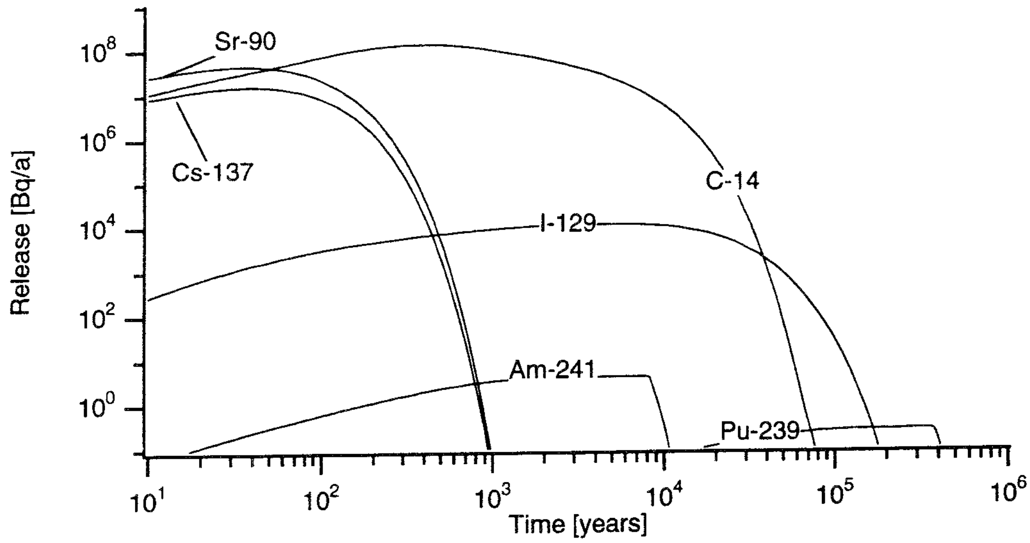

Figure B-1. *Radionuclide release with no resistance from canister.*

Appendix C

List of FEPs

Table 1-1. List of FEPs.

FEP #	RES Type	FEP Name	Comments
1	P_1'	Decay and chain decay of radioelements	Essential, well understood
2	P_1'	Helium production	Easy to estimate, limited effect
3	P_1'	n-activation of fuel, cladding and metal parts	Small effect compared to reactor
4	P_1'	Radiation effects on fuel matrix	Limited long-time effect
5	P_2'	Compaction of material	May be of importance for criticallity
6	P_2'	Impurities	Effect depends on type
7	P_3'	Loss of interity after welding of copper	Difficult to estimate, important
8	P_5'	Initially damaged	Low probabilty, large effect
9	P_{10}'	Glaciation	Important scenario
10	I_{12}	1) Radiolysis of air + water 2) Radiation effects	1) Evaluation in progress 2) Limited concern
11	I_{13}	Radiation effects (n)	Limited concern
12	I_{15}	Radiation effects (n)	Limited concern
13	I_{16}	Decay heat	Essential, well understood
14	I_{17}	Radiation effects	Limited concern
15	I_{21}	Surface coating	Limited concern
16	I_{23}	Surface coating	Limited concern, may have effect on corrosion
17	I_{26}	Temperature gradient	Material dependent, limited concern
18	I_{32}	Confinement	Obvious
19	I_{34}	Causes the gap	Obvious
20	I_{35}	Load on the bottom of canister	Design parameter
21	I_{36}	Temperature gradient	Has been calculated

22	I_{46}	Temperature gradient	-
23	I_{53}	Confinement	Obvious
24	I_{54}	Causes the gap	Obvious
25	I_{56}	Temperature gradient	Has been calculated
26	I_{57}	1) Cu-ion exchange 2) Cementation 3) Pressure	1) Limited concern 2) Limited concern 3) Design parameter
27	I_{58}	Change of natural flow paths	Small effect compared to I_{78}
28	I_{511}	Repository lay-out	Design parameter
29	I_{61}	1) State of the fuel 2) Pressure	Small effect Limited concern
30	I_{62}	State of the filler	Small effect
31	I_{63}	Thermal expansion	Easy to estimate
32	I_{65}	Thermal expansion	Easy to estimate
33	I_{67}	1) Mineral alteration 2) Change of properties	1) Important, difficult 2) Small direct effect due to relativly small Δt
34	I_{68}	Convection cells	Has been calculated, limited effect
35	I_{69}	1) Formation of new 2) Change of properties of existing	Should have minor effect, but must be evaluated
36	I_{611}	Repository layout	Design parameter
37	I_{75}	1) Confinement 2) Pressure	1) Obvious 2) Design parameter
38	I_{76}	Temperature gradient	Dependent on satuaration, important
39	I_{78}	Decides the local hydrology and chemistry	Very important, needs careful evaluation
40	I_{79}	Intrusion into fractures	Could have minor importance, included in near-field model
41	I_{710}	Swelling pressure	-
42	I_{711}	Repository layout	Design parameter
43	I_{85}	Transport of corrodants	Important, included in near-field model

44	I_{86}	Temperature gradient	Limited concern
45	I_{87}	1) Saturation 2) Swelling 3) Mineral alteration 4) Ion echange 5) Erosion	All processes important
46	I_{89}	Fracture filling minerals: dissolution/ precipitation	Limited imporatance in near-field
47	I_{811}	Positioning of deposition holes	Work in progress: safety-construction
49	I_{97} (M_{975})	Large movements might damage canister	Low probabilty, large effect
50	I_{98}	The fracture system decides the waterflow	Yes
51	I_{910}	Rock movements might give transient loads	Work in progress
52	I_{105}	1) Creep 2) Stress corrosion cracking	Work in progress on both processes, may be important
53	I_{108}	Decides the gradient	Obvious
54	I_{109}	Sealing and possible widening of fractures	-
55	I_{115}	Damage during emplacement	Difficult to predict, important
56	I_{117}	Affects properties	Quality control
57	I_{118}	Chemical effects from man-made materials	Important to evaluate, little has been done
58	I_{119}	Fracture injection and plugs	
59	I_{1110}	Repository depth - hydrostatic pressure	Obvious

Interactions occuring only when copper overpack is penetrated

60	I_{57}	Loss of density	No concern
61	I_{58}	Water intrusion	Important
62	I_{84}	Fills with water	-

63	M_{843}	1) Steel corrosion: - corrosion product buildup - hydrogen gas generation	Critical questions, rather easy to get a fair estimate
64	M_{8434}	Fills with corrosion products and H_2 gas	-
65	M_{84343}	1) Is penetrated 2) Loses mechanical integrity	Time might be critcal
66	M_{8437}	1) Gas pressure and transport 2) Fe-ions	1) Not very well understood, additional work planned 2) May lead to cementation
67	M_{8438}	Water consumption	No concern
68	M_{8435}	1) Galvanic effects 2) Pressure, "yawning"	1) Limited effect, has been evaluated 2) Potential importance, has been calculated
69	M_{846}, M_{8436}	Change of thermal properties	Limited concern
70	M_{8438}	May change local transport paths	Minor effect
71	M_{84378}	Displaces water	Near field may be gas filled
72	M_{84379}	Gas transport in rock	Important, has been calculated
73	M_{843710}	Local pressure changes	-

Interactions occuring only when steel canister is penetrated

74	I_{37}	Buffer intrusion - loss of density	The effect should be calculated
75	I_{38}	Water intrusion	
76	I_{86}	Change of thermal properties	Limited concern
77	I_{83}	Corrosion on inside	
78	I_{82}	Chemical reactions - dissolution	Material dependent

79	I_{81}	1) Fuel and clad corrosion 2) Precipitation/ dissolution of secondary minerals 3) Criticality	Important questions, model available
80	M_{812}	Radionuclide sorbtion	May have effect
81	M_{813}	Radionuclide sorbtion	May have effect (corrosion products)
82	M_{816}	Reaction heat	Limited concern
83	M_{817}	1) Radionuclide sorbtion 2) Radiation damage 3) Reconcentration	1) Very important 2) Limited concern 3) May be important for certain nuclides
84	M_{818}	1) Contamination 2) Radiolysis	1) The transport path to the biosphere 2) Limited concern
85	M_{819}	1) Radionuclide sorbtion 2) Matrix diffusion	Important processes in the far field
86	M_{832}	1) Gas phase formation 2) Corrosion product intrusion	1) Limited concern as a singular phenomena 2) Limited concern
87	I_{81},M_{832}	Radionuclides enters the gas phase	Must be evaluated
88	I_{81},M_{8327}	Radioactive gas penetrates buffer	Crucial

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N Platts, D J Blackwood, C C Naish
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Paul Wersin, Kastriot Spahiu, Jordi Bruno
MBT Technología Ambiental, Cerdanyola, Spain
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Calin Cosma[1], Christopher Juhlin[2], Olle Olsson[3]
[1] Vibrometric Oy, Helsinki, Finland
[2] Section for Solid Earth Physics, Department of Geophysics, Uppsala University, Sweden
[3] Conterra AB, Uppsala, Sweden
February 1994

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Jan Cramer (ed.)[1], John Smellie (ed.)[2]
[1] AECL, Canada
[2] Conterra AB, Uppsala, Sweden
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Sven Åke Larsson[1,2], Eva-Lena Tullborg[2]
[1] Department of Geology, Chalmers University of Technology/Göteborg University
[2] Terralogica AB
November 1993

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Göran Bäckblom (ed.), Christer Svemar (ed.)
Swedish Nuclear Fuel & Waste Management Co, SKB
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Peter Andersson (ed.)[1], Anders Winberg (ed.)[2]
[1] GEOSIGMA, Uppsala, Sweden
[2] Conterra, Göteborg, Sweden
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The structure of conceptual models with application to the Äspö HRL Project
Olle Olsson[1], Göran Bäckblom[2], Gunnar Gustafson[3], Ingvar Rhén[4], Roy Stanfors[5], Peter Wikberg[2]
1 Conterra AB
2 SKB
3 CTH
4 VBB/VIAK
5 RS Consulting
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Kjell O Wannäs, Tom Flodén
Institutionen för geologi och geokemi, Stockholms universitet
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Hans Wanner[1], Yngve Albinsson[2], Erich Wieland[1]
[1] MBT Umwelttechnik AG, Zürich, Switzerland
[2] Chalmers University of Technology, Gothenburg, Sweden
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D J Blackwood, A R Hoch, C C Naish, A Rance, S M Sharland
AEA Technology, Harwell Laboratory, Didcot, Oxfordshire, UK
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A R Hoch, S M Sharland
Chemical Studies Department, Radwaste Disposal Division, AEA Decommissioning and Radwaste, Harwell Laboratory, UK

October 1993