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Application of the Graded Approach to Post-closure Safety Assessment for the Disposal of Disused Sealed Radioactive Sources in Boreholes

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APPLICATION OF THE GRADED APPROACH TO POST-CLOSURE SAFETY ASSESSMENT FOR THE DISPOSAL OF DISUSED SEALED RADIOACTIVE SOURCES IN BOREHOLES

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APPLICATION OF THE GRADED APPROACH TO POST-CLOSURE SAFETY ASSESSMENT FOR THE DISPOSAL OF DISUSED SEALED RADIOACTIVE SOURCES IN BOREHOLES

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FOREWORD

Sealed radioactive sources are used all over the world for many beneficial purposes in areas such as health, industry, research and agriculture. These sources are usually managed safely and securely while in use; however, many States experience challenges in managing them once these sources reach the end of their useful lives. Even after they no longer emit sufficient radiation for their intended purpose, they remain radioactive and thus need to be managed as disused sealed radioactive sources.

Some disused sealed radioactive sources are extremely radioactive and hazardous, and accidents, including fatalities, have occurred when people have handled or otherwise been exposed to them. There is also a risk that disused sealed radioactive sources could be used for malicious purposes. Safe and secure storage is a key step in managing these sources, but storage is not a permanent or sustainable management solution. These accidents and risks, and the requirement for a sustainable management option, highlight the need for safe disposal solutions.

To assist Member States in the safe management of disused sealed radioactive sources, the IAEA has developed a concept for their disposal in boreholes. A licence has been granted in Malaysia for a borehole disposal facility following this concept, and in Ghana a safety case for a borehole disposal facility following the concept is under active development. Several other Member States are interested in the concept, and it may also be of interest to other Member States that have disused sealed radioactive sources but lack suitable disposal facilities.

IAEA Safety Standards Series No. SSG-1, Borehole Disposal Facilities for Radioactive Waste, provides specific guidance on the design, construction, operation and closure of borehole disposal facilities. SSG-1 mainly focuses on boreholes a few hundred millimetres in diameter for the disposal of disused sealed radioactive sources. The IAEA publication Generic Post-closure Safety Assessment for Disposal of Disused Sealed Radioactive Sources in Narrow Diameter Boreholes (IAEA-TECDOC-1824) supports SSG-1 by presenting a generic post-closure safety assessment for the IAEA concept for the disposal of disused sealed radioactive sources in narrow boreholes.

It is a fundamental principle that safety has to be assessed for all facilities and activities, consistent with a graded approach. The use of the graded approach is intended to ensure that the level of effort to be applied in carrying out the safety assessment is commensurate with the magnitude of the possible radiation risks arising from the facility or activity, the nature and the particular characteristics of the facility, and the stage in the lifetime of the facility. The present publication complements SSG-1 and IAEA-TECDOC-1824 by describing how to apply the graded approach to post-closure safety assessment for the disposal of disused sealed radioactive sources in boreholes in accordance with the relevant IAEA safety standards. It is intended primarily for those involved in developing or regulating borehole disposal facilities for disused sealed radioactive sources.

The IAEA wishes to express its gratitude to all those who assisted in the drafting and review of this publication. The IAEA officer responsible for this publication was D.G. Bennett of the Division of Radiation, Transport and Waste Safety.

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1. INTRODUCTION

1.1. BACKGROUND

1.1.1. General

Waste disused sealed radioactive sources (DSRS) are one of the most challenging categories of radioactive waste to manage. It is estimated that over 10 million sealed radioactive sources (SRS) have been manufactured over the past century for a wide range of beneficial uses in agriculture, industry, medicine, education, various research areas, and some military applications [1]. Today, millions of these sources are considered disused, and of those, the majority are being managed as radioactive waste [2].

Physically small, and sometimes highly radioactive, DSRS are recognized as both a safety concern (particularly in cases of poor management and accidents) and a security concern (e.g. related to their possible use in malicious acts). The majority of DSRS are kept in storage. However, storage is not a sustainable long-term management solution. While putting DSRS into safe storage is an important and positive step, it remains a short-term measure, with numerous challenges still to be dealt with (e.g. security, sustainability, liability, operational funding, maintenance, management issues, radiation protection, etc.).

Some of the radionuclides that are used in DSRS have half-lives of hundreds to millions of years, which means that a long-term perspective is necessary for the selection and development of appropriate management strategies for DSRS declared as radioactive waste. A concerted effort is needed to ensure compliance over the timescales of concern with the fundamental safety objective [3] and the IAEA safety standards. Relevant guidance is also contained in the Code of Conduct on the Safety and Security of Radioactive Sources and its supplementary Guidance on the Management of Disused Radioactive Sources [4].

Although not explicitly defined, the concept of a ‘cradle to grave’ approach to the control of radioactive sources was conceived in the late 1990s and work in this area included the development of the Borehole Disposal Concept (BDC) as a long-term management solution for DSRS [5]. The cradle to grave approach simply considers all stages in the lifetime of a facility, activity or product [6]. Adopting this approach for the control of radioactive sources recognized the importance of the end of life management of DSRS, including disposal.

Since the introduction of the BDC in 1998 [5], considerable effort has been expended to develop a safe and secure long-term management solution for DSRS [7] [8]. While not the only long-term management option, the BDC was developed as a viable alternative for States that do not have extensive nuclear or large disposal programmes (e.g. for the development of large geological disposal facilities). The development of the Borehole Disposal System (BDS) as it is known today is consistent with the guidance provided in IAEA Safety Standards Series No. SSG-1, Borehole Disposal Facilities for Radioactive Waste [8].

Historically, a variety of borehole designs has been used for the storage and disposal of radioactive waste and DSRS, specifically with differing depths (a few metres to several hundred metres) and diameters (a few tens of centimetres to several metres) [9]. The BDS as considered in this publication is a safe and secure disposal system that includes a narrow diameter borehole (e.g., 0.26 m)[1] and a multi-barrier system consisting of engineered and natural barriers. This multi-barrier disposal system contains and isolates the wastes from the biosphere in accordance with the requirements established in IAEA Safety Standards Series No. SSR-5, Disposal of Radioactive Waste [10]. The current design can accommodate DSRS of various lengths and diameters. The waste is disposed in the borehole at depths sufficient to reduce the likelihood of inadvertent human intrusion, and of any other potential causes of the waste returning to the surface (e.g. due to erosion).

1.1.2. Development of the generic safety assessment

In addition to developing the disposal concept over the past 20 years, a series of post-closure safety assessments have investigated the key safety features of the BDS under varying disposal system conditions to support the concept design and licensing processes, and to facilitate its site-specific implementation. Each of these post-closure safety assessments was generic, using generic or unit inventories, as well as generic site and climate conditions.

The first post-closure safety assessment study focused on disused radium needles, because radium is long-lived (1600-year half-life) and relatively mobile in the environment [11]. Based on the limited, but positive results from this first assessment, a second generic assessment was undertaken using a larger and more representative inventory of DSRS radionuclides. This second assessment also explored the performance of the disposal system using a wide range of possible engineered barrier materials (e.g., stainless steel, copper, lead, cement and bentonite), for a range of geospheres (i.e., arenaceous, argillaceous, and crystalline) and biospheres (i.e., humid, seasonally humid and arid/semi-arid) [12].

The most recent generic safety assessment (GSA) published in 2017 in IAEA-TECDOC-1824, Generic Post-Closure Safety Assessment for Disposal of Disused Sealed Radioactive Sources in Narrow Diameter Boreholes [13], considered 31 radionuclides, each with a unit activity concentration (1 TBq), disposed in a borehole with stainless steel and cement barriers and emplaced in a range of geosphere conditions. The GSA showed that, with a suitable combination of inventory, near field design and geological environment, the BDS can provide a safe long-term management solution for the disposal of both long-lived and short-lived radionuclides. As always, the inventory of such radionuclides that can be safely disposed of at a particular site in the BDS would need to be assessed in a site-specific safety assessment.

1.1.3. Supporting studies

In addition to the GSA [13], several studies were performed in support of the BDS development process. Most of these studies were performed or finalized after the GSA was finalized and, consequently, were outside the initial scope of the GSA. However, these studies have contributed significantly to the development of the BDS and to confidence in its overall safety, both from an operational and a post-closure safety perspective. These studies include:

- The stainless steel corrosion models and cement degradation models developed as part of the GSA were incorporated into a BDC Scoping Tool [14]. The BDC Scoping Tool allows the containment provided by the capsule and disposal container in the post-

[1] This report considers a narrow diameter (0.26 m) borehole because this diameter is commonly used for drilled water abstraction wells and borehole drilling technology at this diameter is readily available in all countries [8].

closure period to be evaluated along with the chemical and physical degradation of the cement. It also allows radionuclide transport and subsequent exposure of humans via the drinking water pathway to be evaluated using a conservative model that takes no account of the retardation of radionuclides during transport.

- Initially, the GSA did not explicitly consider radiolysis, criticality or thermal effects since at the time such effects were viewed to be insignificant for the typical inventories to be disposed of. However, there are certain high activity Category 1 and Category 2 sources, for which it might be necessary to consider such effects. Ref. [15] addressed the potential impacts of the disposal of such high activity radioactive sources on the post-closure safety of the BDS. This work led to the adoption of various capsule and waste package specifications to accommodate Category 1 and Category 2 sources, as well as to the integration of the mobile hot cell into the conditioning and disposal operations for Category 1 and Category 2 DSRS (see below).

- The study presented in Ref. [15] was based on conservative assumptions and calculations and indicated that, whilst there are no criticality issues, the disposal of some Category 1 and Category 2 DSRS might result in enhanced temperatures and high radiation fields that could significantly reduce the expected lifetime of the waste disposal packages. Consequently, less conservative calculations were adopted in a further study to develop an improved understanding of the thermal and radiation conditions in the borehole for representative Category 1 and Category 2, and Category 3 to Category 5 DSRS, respectively [16]. The study was supported using CHEMSIMUL and Microshield calculations.

- Ref. [17] undertook a review of the rates of general and localized corrosion of stainless steel in cementitious environments. The review also considered the potential effects of gamma radiation and galvanic corrosion between carbon and stainless steels in concrete. The focus of this review was on 304 and 316 austenitic stainless steels. The work led to recommendations for the use of superaustenitic or superduplex stainless steel or a Pd-containing titanium alloy for the capsule and the disposal container for the disposal of heat-generating and gamma-emitting Category 1 and Category 2 DSRS.

- The conditioning of the DSRS into the disposal containers and the associated operational safety aspects were outside the scope of the GSA, which addressed post-closure safety. However, after the development of the GSA, DSRS conditioning equipment has been designed and manufactured that also facilitates the waste package emplacement operation. While not directly of relevance to the post-closure safety of the BDS, the mobile hot cell and the mobile tool kit now facilitate the emplacement and disposal of Category 1 and Category 2 DSRS using the mobile hot cell [1], and Category 3 to Category 5 DSRS using the mobile tool kit [18].

1.1.4. Site-specific safety assessment and safety case development

The aim of developing the BDS over the past 20 years was to establish a long-term management solution for DSRS that can be implemented in a range of geological and climatic conditions encountered in different States. The GSA was consequently developed to identify the concept's key safety features on a generic basis, considering ranges of disposal system conditions and using a unit inventory of the most relevant DSRS. Despite its generic nature, the GSA can be used as a starting point for site-specific safety assessment and safety case development.

Ghana and Malaysia are the first Member States to begin the site-specific implementation of the BDS for the long-term management of DSRS. With support from the IAEA, considerable effort has gone into the development and independent review of site-specific safety assessments and safety cases, while at the same time contributing to the further development and enhancement of the BDS [20, 21]. The implication is that Members States considering the implementation of the BDS in future will benefit significantly from the lessons learned from the Ghana and Malaysia safety assessment and safety case development processes, and the use of the GSA in these processes as part of the site-specific implementation of the BDS.

1.2. OBJECTIVE

The requirements to develop a safety case including a supporting post-closure safety assessment for a radioactive waste disposal facility are described in SSR-5 [10] and in IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [19]; guidance on such safety assessments is provided in IAEA Safety Standards Series No. SSG-23, The Safety Case and Safety Assessment for the Disposal of Radioactive Waste [22]) and in SSG-1 [8].

The application of the graded approach is an integral part of the broader radioactive waste management framework, including the development of the safety case and supporting safety assessment [22]. As discussed in Section 2.2, the application of the graded approach is of particular importance and relevance to the BDS.

The purpose of this publication is to complement the guidance provided in the above mentioned publications, with further information on how to perform post-closure safety assessment in support of site-specific safety cases for the implementation of the BDS using a graded approach. The information provided here largely stems from lessons learned and experiences gained during the development of the GSA, as well as the review of the site-specific safety cases and safety assessments, notably those being developed in Ghana and Malaysia. The specific objectives for this publication are:

- To provide information on how to apply the graded approach when assessing the safety of borehole disposal of DSRS, using the models developed for the GSA in combination with additional simplified models;
- To provide information on the use of the detailed GSA and more simplified models to build confidence in the post-closure safety assessment results and to develop safety arguments in support of the safety case; and
- To provide information on the use of the various safety assessment models to guide disposal system characterization.

1.3. SCOPE

The scope of the publication is limited to the BDS as introduced in Section 1.1. While the principles applied here may in concept be applicable, this publication does not apply to the Greater Confinement Disposal boreholes [23], to the RADON type borehole facilities [24], or to very deep (e.g., deeper than 1 km) borehole disposal facilities [25].

The scope of this publication is limited to the provision of information on post-closure radiological safety assessment of the BDS. Further reference to *safety assessment* in this publication, therefore, implies the post-closure radiological safety assessment process.

The post-closure safety assessment methodology described in Refs [22] and [26], for example, includes many elements. It is beyond the scope of the publication to address and provide detailed information on each of these elements. Consistent with the purpose and objectives, the focus of this publication is on model development (i.e., conceptual and mathematical model development) and the use of models of differing complexity in the safety analysis process in accordance with the graded approach.

The focus of this publication is, therefore, to provide information on how to perform a post-closure safety assessment of the BDS in support of the safety case. Viewed from this perspective, the intent is to complement the work that has been undertaken as part of the GSA and described in Ref. [13]. The intent is not to replace the models presented in the GSA but to complement them and describe how the graded approach can be applied when using models with different levels of complexity and conservatism.

While the focus of this publication is on model development and the use of these models, a high level overview of the safety assessment and safety case development process is provided, emphasising their respective roles when applying the graded approach to safety assessment for the implementation of the BDS. Consistent with the purpose and objectives of the publication, the use of the safety assessment in support of the safety case development process is emphasized, with specific reference to confidence building, the management of uncertainties, and the integration of safety arguments.

1.4. STRUCTURE

The remainder of the publication is structured as follows:

- Section 2 describes the role of the safety assessment within the safety case development process for the BDS, especially to build confidence, manage uncertainties, and to present safety arguments in support of the safety case within a graded approach.

- Section 3 provides a summary of the conceptual models as used and implemented in the GSA for the near field, geosphere and biosphere, for the Design Scenario, as well as for the alternative Defect Scenarios considered in the GSA.

- Section 4 provides a detailed description of the hierarchy of conceptual and mathematical models defined in this publication that can be used in a graded approach to support the safety assessment and safety case development process for the BDS.

- Section 5 provides information on how to use the hierarchy of models described in Section 4 as part of the safety assessment process, with the emphasis on the graded approach and the presentation of safety arguments in the safety case.

- Section 6 presents some general conclusions on the results and on the use of the publication.

- Appendix I provides an example of the application of the graded approach to post-closure safety assessment for the BDS.

- Appendix II contains the data used in the example discussed in Appendix I.

- Appendix III details the Tier 4 mathematical model that was developed and used during this work and which, unlike the models at the other tiers, has not previously been documented in other publications.

2. SAFETY ASSESSMENT AND SAFETY CASE DEVELOPMENT

2.1. GENERAL

Experience has shown that the availability of the GSA with its supporting conceptual and mathematical models has assisted Member States that intend to implement the BDS in the preparation of site-specific post-closure safety assessments. However, the preparation of site-specific safety cases has proved to be more challenging. Member States have had difficulties in transferring the guidance provided in SSG-1 [8] and SSG-23 [22] into systematic and structured site-specific safety cases for the BDS.

The purpose of this section is not to reproduce the guidance provided in SSG-1 [8] and SSG-23 [22] and other related publications, but to emphasize in Section 2.2 the role and use of the safety assessment within the safety case development process, especially when adopting a graded approach for the implementation of the BDS. This is followed in Section 2.3 by a general, high level overview of the safety case concept and structure, and by Section 2.4 which emphasizes the role and importance of the safety assessment within the safety case structure. Section 2.5 discusses the use of the safety assessment results in support of the safety arguments presented in the safety case for a BDS.

2.2. GRADED APPROACH

The graded approach is a well established concept and it is an integral part of the broader radioactive waste management framework. According to the IAEA Safety Glossary [6], "the use of the graded approach is intended to ensure that the necessary levels of analysis, documentation, and actions are commensurate with, for example, the magnitudes of any radiological hazards and non-radiological hazards, the nature and the particular characteristics of the facility, and the stage in the lifetime of a facility."

This is consistent with Requirement 1 established in GSR Part 4 (Rev. 1) [19], which identifies the criteria to be taken into consideration in the application of the graded approach, namely the magnitude of the possible radiation risks and the maturity and complexity of the facility. According to SSG-23 [22], this also is consistent with Principle 5 of the Safety Fundamentals SF-1 [3] where it is stated that "the resources devoted to ensuring safety have to be commensurate with the magnitude of the possible radiation risks and their amenability to control." In accordance with this principle, the safety case and safety assessment ought to be developed and conducted only to the level of detail that is appropriate, both to the magnitude of the risks and to the stage of development of the disposal facility.

The BDS was always conceived as being a small disposal facility for a limited inventory of radioactive waste in terms of volume and activity, as compared, for example, to the larger facilities and waste inventories typically involved in geological or near surface disposal. According to the above mentioned approach and criteria, this in itself calls for a direct application of the graded approach to the development of the safety case and the supporting safety assessment for the BDS, i.e. the work required to develop the safety case and supporting safety assessments for a borehole disposal facility for a small inventory of DSRS can be expected to be less extensive than for a larger waste disposal facility.

2.3. THE ROLE OF THE SAFETY CASE

A safety case is the collection of scientific, technical, administrative and managerial arguments and evidence in support of the safety of a disposal facility, covering the suitability of the site and the design, construction and operation of the facility, the assessment of radiation risks and assurance of the adequacy and quality of all of the safety related work associated with the disposal facility [22]. The overall purpose of the safety case for a disposal facility is to demonstrate, with an appropriate level of confidence, that the disposal system is feasible to implement, and is and will be safe. Figure 1 presents the components that constitute the safety case structure.

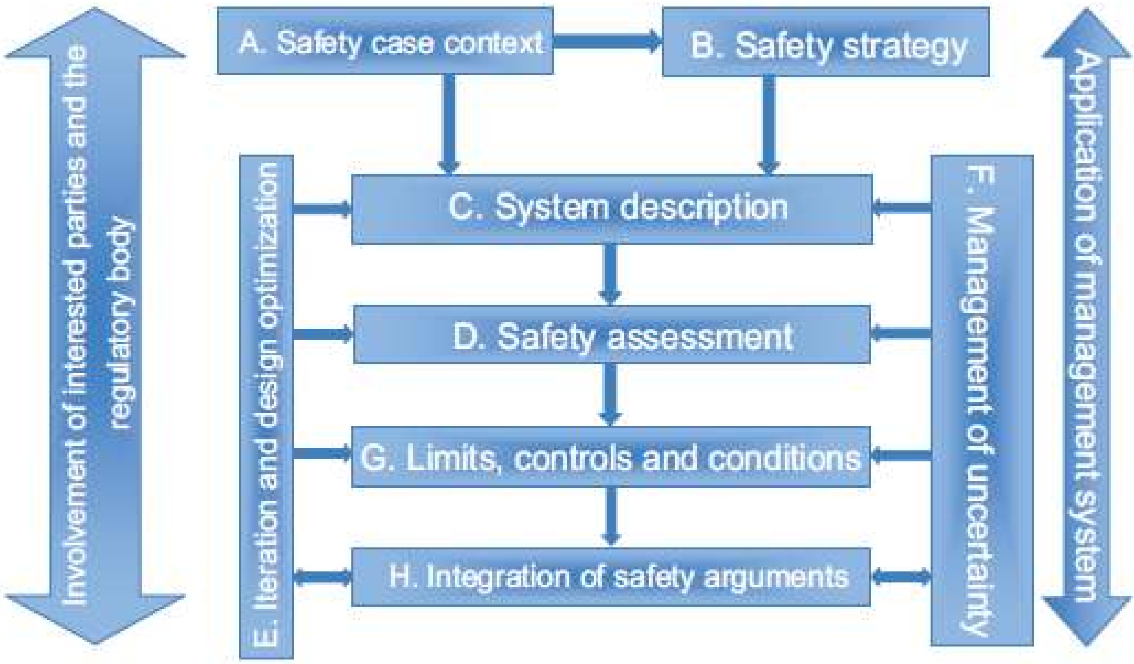

Figure 1. Relationship between the different elements of the safety case for a radioactive waste disposal facility [22].

Figure 1 shows that most of the safety case components are structured in an interrelated manner with the aim of presenting all the arguments and supporting assessments, analyses and evidence that demonstrate the safety of a disposal facility. In this regard, the role of the safety case is to provide (see para. 4.6 of SSG-23 [22]):

- “Integration of relevant information in a structured, traceable and transparent way that demonstrates an understanding of the behaviour and performance of the disposal system in the period after closure;

- Identification of uncertainties in the behaviour and performance of the disposal system, analysis of the significance of the uncertainties, and identification of approaches for the management of significant uncertainties;

- Demonstration of long-term safety by providing reasonable assurance that the disposal facility will perform in a manner that protects human health and the environment;

- Support to decision making in the step by step approach to the development of a disposal facility; and
- Facilitation of communication between interested parties on issues relating to a disposal facility."

Figure 2 illustrates the typical evolution of the safety case during the development of a radioactive waste disposal facility. Figure 2 also shows the typical sequence of key decisions that would be made, and who would make them, during this process. IAEA-TECDOC-1814, Contents and Sample Arguments of a Safety Case for Near Surface Disposal of Radioactive Waste [27], provides a detailed description of these key decision steps.

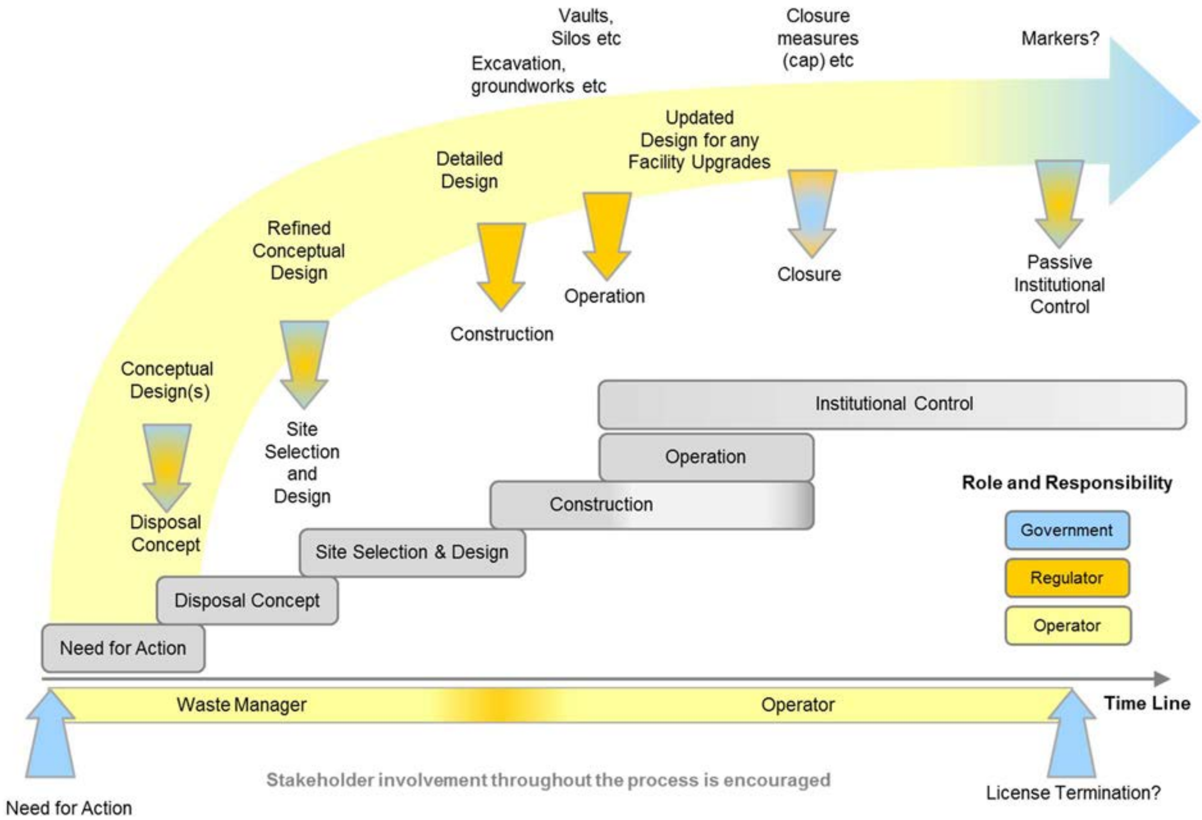

Figure 2. The typical sequence of key decisions in the development of a disposal facility for radioactive waste, presented in Ref. [27].

While interested parties are generally not decision makers, their input and careful consideration of their concerns are critical to the development of any disposal facility. A decision to move from one phase of facility development to another is a strategic step, in which it is necessary to consider a range of factors, including safety, legal requirements, costs and available resources, schedules, and interested parties' views [27].

As an integral part of the safety case, the safety assessment is an iterative process that is carried out throughout the safety case evolution depicted in Figure 2, and throughout the lifetime of the radioactive waste disposal facility in general. However, the operating life of the BDS is very short relative to that of typical near surface or geological disposal facilities. In fact, in most cases, it is foreseen that the construction, waste emplacement and facility closure operations for a BDS will be conducted in a one-off campaign lasting just a few months to a year.

2.4. THE ROLE OF THE SAFETY ASSESSMENT

Safety assessment is one of the key elements of the safety case and generally involves several aspects within the safety case structure presented in Figure 1. As illustrated in Figure 2, it encompasses all safety aspects that are relevant for the development, operation, closure and long-term safety of the disposal facility. The analysis of the post-closure radiological impact on humans and the environment in terms of both radiation dose and radiation risks is a fundamental aspect of safety case development. Other important aspects subject to safety assessment include site and engineering aspects, operational safety, non-radiological impacts and the management system (see Figure 3). Addressing these aspects is beyond the scope of this publication.

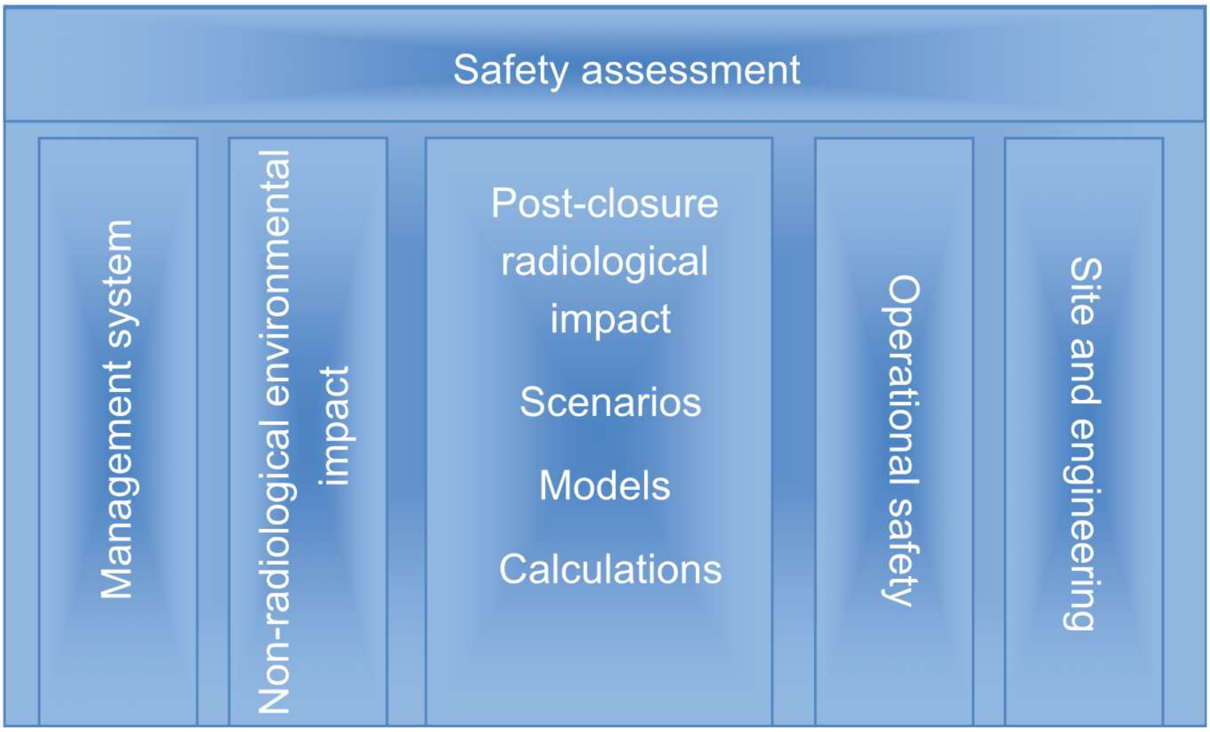

Figure 3. Aspects included in the safety assessment presented in Ref. [22].

Consistent with Figure 2, Figure 4 shows that the post-closure safety assessment has different purposes at different stages during the lifetime of a radioactive waste disposal facility. As the operation moves from the pre-operational to the post-closure period, the purpose, scope and focus of these assessments may change and, therefore, needs to be clearly defined at each point. Due to the relatively short operating period of the BDS, less time is available for undertaking post-closure safety assessment. This suggests that emphasis can be put on the post-closure safety assessment conducted to support the application for a disposal licence, with less expectation for subsequent iterations.

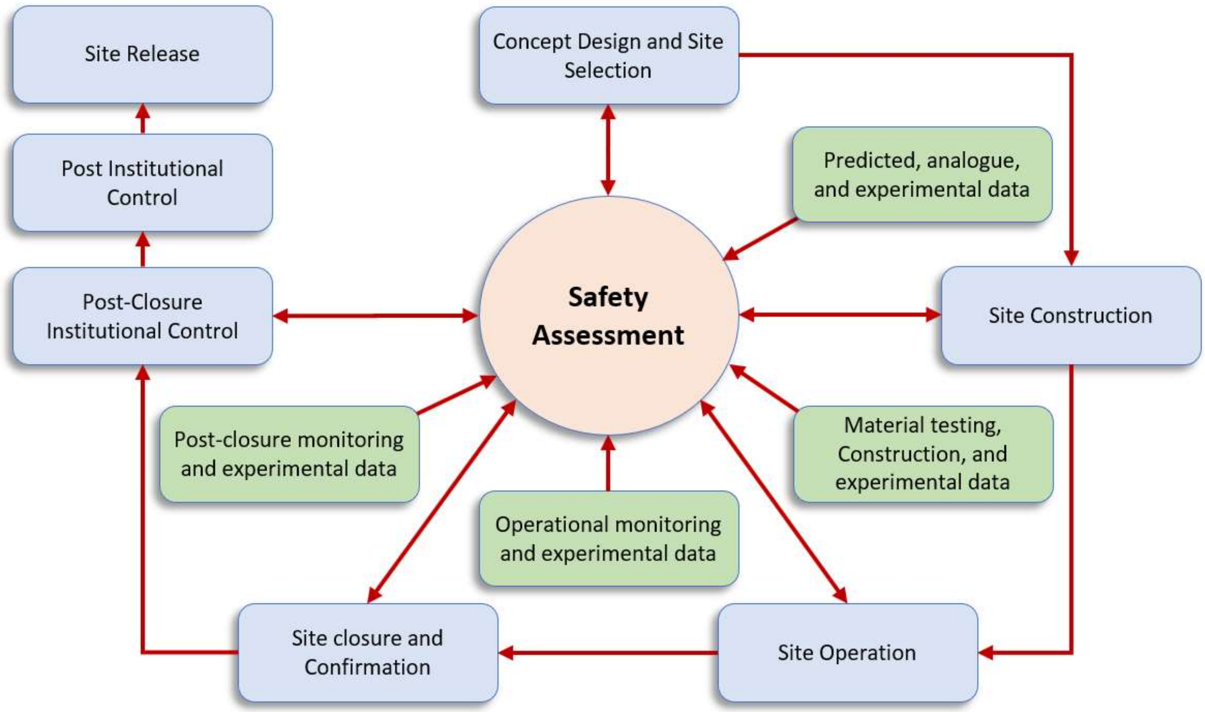

Figure 4. The role of safety assessment in the lifetime of a radioactive waste disposal facility.

With the focus on the post-closure period, the IAEA safety assessment methodology presented in Ref. [26] was used to develop the GSA and remains an important reference for developing a post-closure safety assessment. The methodology is consistent with best practice elsewhere (e.g., Ref. [28]) and it has been adapted and tested and found to be equally useful in assessing the impact of other types of disposal facility [22].

2.5. THE USE OF THE SAFETY ASSESSMENT IN SUPPORT OF THE SAFETY CASE

2.5.1. General

Section 2.3 described the role of the safety case, while Section 2.4 described the role of safety assessment within the safety case structure. These descriptions are general and apply to all types of radioactive waste and disposal facilities. It was noted that the safety assessment is a key element within the safety case structure in Figure 1 and is developed in support of the safety case. The purpose of this section is to provide an overview of how the safety assessment and associated safety analysis results are used in support of developing the safety case.

2.5.2. Management of uncertainties

Uncertainty in the behaviour of radioactive waste disposal systems, particularly over long timescales, is inevitable [28]. Uncertainty management, as one of the elements within the safety case structure (see Figure 1), recognizes that uncertainties need to be acknowledged, evaluated and, where significant, acted upon. In many cases, this will be done routinely within and as part of the safety assessment using scenarios, conceptual models, and ranges of parameter values. Figure 5 shows a general structure for analyzing uncertainties in this way.

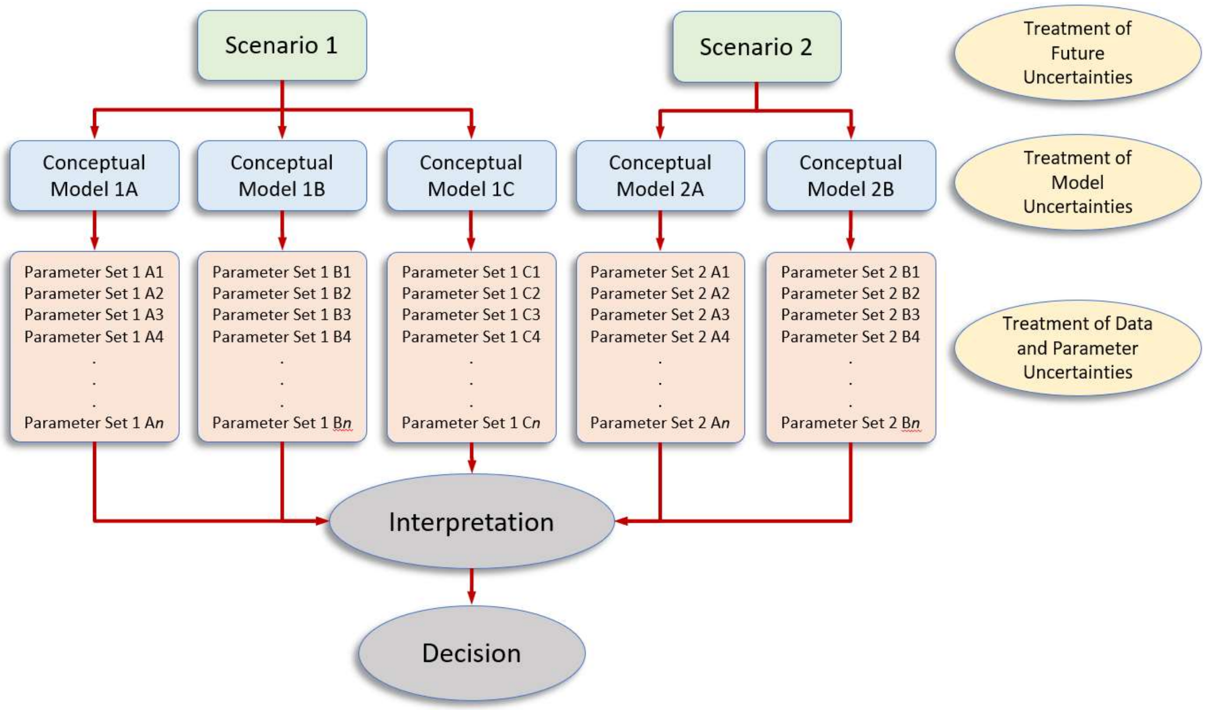

Figure 5. Structure of uncertainty analysis, showing the relationship of scenario, conceptual model, and parameter uncertainty [29].

It is important to recognize the unusual nature of post-closure safety assessments when considering uncertainty [29]. The uncertainty analysis needs to match the phase of the safety assessment to which it is applied. First, the magnitude of the calculated dose needs to be known and compared to the performance objective. Second, the credibility of the assessment output needs to be established. Because there are always limitations to the amount of data available to support safety assessments, calculated doses will always be uncertain to some extent. Consequently, the key issue is to identify the conditions for which uncertainty in assumptions or parameters are significant and could result in an altered decision. If the decision can be demonstrated to be insensitive to judgments about uncertainty in scenarios, conceptual models, and parameter values, the decision can be defended with confidence.

2.5.3. Building confidence in safety

The NEA-OECD defines confidence as a positive judgement that a given set of conclusions are well supported [30]. It is from this perspective that confidence building has become an integral part of safety assessment and the safety case development process. To build confidence in the post-closure safety of the disposal system, the safety assessment documents can present a series of arguments that are intended to build confidence in safety. These may include arguments related to defence in depth, multiple lines of reasoning, institutional control, environmental monitoring, information from natural analogues, and the adoption of conservative approaches.

Uncertainty and sensitivity analyses can be used to demonstrate defence in depth: that the safety of the system is not unduly reliant on any one feature of the design or to the assumptions made in the safety assessment; further, that if one barrier were to fail prematurely, safety would still be provided. Such analyses can also support a multiple lines of reasoning approach by citing a series of arguments, any one of which could be used to justify the safety of the facility. Such arguments may be aided by the inclusion of a diagram like Figure 6, which shows some key factors that contribute to the long-term safety of the BDS, and by implication contribute to confidence in the long-term safety of the system.

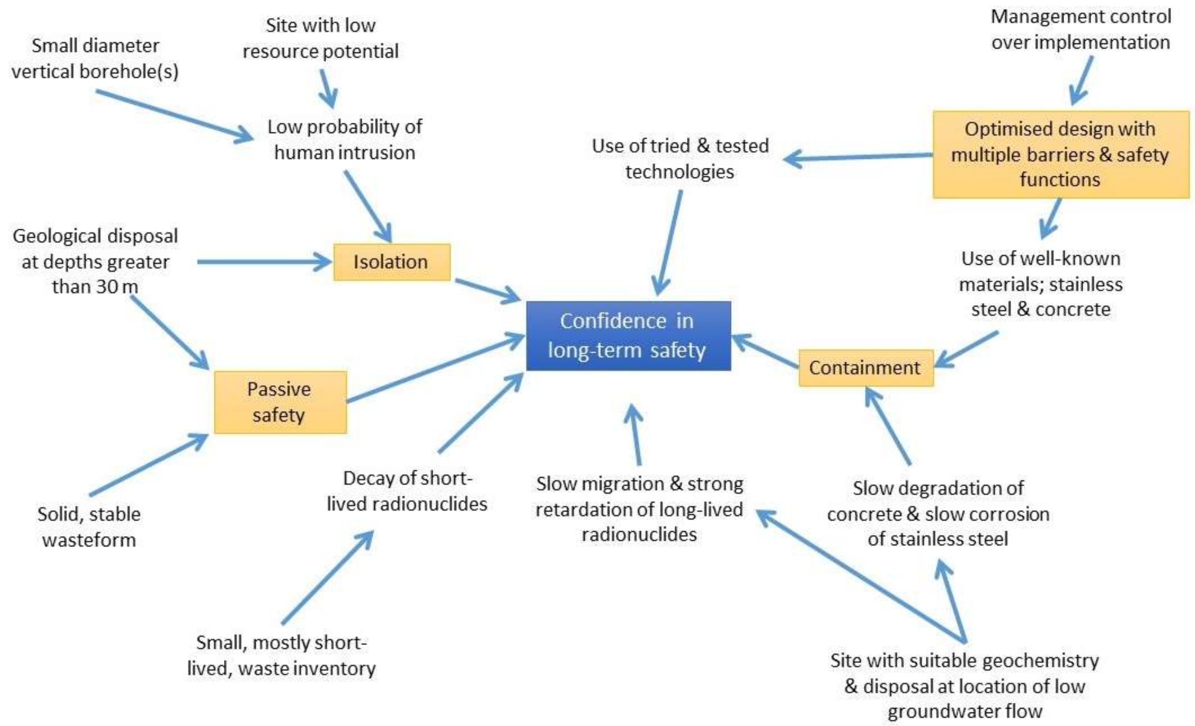

Figure 6. Factors that contribute to the long-term safety of borehole disposal.

2.5.4. Presentation of safety arguments

The integration of safety arguments within the safety case structure presented in Figure 1 is intended to provide a synthesis of the available evidence, arguments and analyses conducted, which leads to the conclusion that the proposed activities can be safely and securely managed.

Although not all, many of the safety arguments are generated and justified through the safety assessment. Simply showing that safety assessment results comply with quantitative regulatory criteria is not sufficient to demonstrate safety. A multiple lines of reasoning approach is required to integrate and synthesize available information, analyses and assessment results. It is also important to acknowledge any limitations of currently available information and how it influences the safety case. The objective of safety case development is, thus, to generate arguments for the safety of the BDS and its implementation. These arguments may be qualitative or quantitative in nature. The multiple lines of reasoning might include discussions of:

- The use of best available techniques;
- The history of design and optimization;
- Waste isolation and containment;
- Passive safety;

- Robustness and defence in depth;
- Management systems, quality assurance and peer review;
- Conservatism in safety assessment;
- The approach to the management of uncertainty through which any open questions and uncertainties will be addressed; and
- Application of limits, controls and conditions.

3. DESCRIPTION OF GSA SCENARIOS AND CONCEPTUAL MODELS

Generic (i.e. not site-specific) safety assessment is a tool that can be used in many aspects of a waste disposal programme. For example, at the concept development stage and in support of site screening, the GSA can be used:

- To help identify radionuclide inventories suitable for disposal;
- To help determine suitable levels of engineering;
- To help determine suitable site characteristics;
- To help determine the need for, and duration of, an institutional control period.

Even when a site has been chosen for investigation, the GSA may help in:

- Identifying the key parameters that need to be characterized for a site-specific assessment and the extent of site characterization required;
- Providing a basis, consistent with good practice, for any site-specific assessment that might be undertaken and helping to build confidence in that site-specific assessment.

A GSA for the disposal of DSRS is presented in Ref. [13]. The GSA conceptualized a disposal system comprising a single borehole containing DSRS that had been disposed of in a series of stainless steel and concrete engineered barriers.

The purpose of this section is to provide a high level overview of the design and alternative scenarios in Section 3.1, as well as of the associated conceptual models in Section 3.2 and Section 3.3 for assessing the disposal of DSRS in the generic borehole disposal facility considered, as implemented and used in the GSA [13].

3.1. SCENARIOS

3.1.1. Design scenario

The GSA uses a structured approach described in Section 4.1 of Ref. [13] to identify a so-called 'Design Scenario' in which the disposal facility is constructed and closed as designed and evolves in the expected way.

The scenario assumes that, due to the corrosion of the stainless steel disposal containers and the subsequent corrosion of the stainless steel capsules, water eventually contacts the sources. The subsequent migration of radionuclides released into groundwater is limited by decay and sorption of the radionuclides onto the cement grout in the near field. On leaving the near field, the radionuclides migrate through the geosphere and are subject to further decay and retardation due to sorption onto the rocks. The groundwater is assumed to be abstracted from the geosphere via a water abstraction borehole that is drilled at the end of the active institutional control period. The water abstraction borehole is assumed to be 100 m down the hydraulic gradient from the disposal borehole and used for domestic purposes (drinking) and for agricultural purposes (watering of cows and irrigation of root and green vegetables).

The failure of the waste disposal containers and capsules also allows radioactive gases to be released. These gases are assumed to migrate up the disposal borehole through the closure zone[2] into the biosphere. It is conservatively assumed that a dwelling is constructed on top of the disposal borehole (without intruding into the disposal zone of the disposal borehole) at the end of the active institutional control period, resulting in the gases migrating directly into the dwelling and being inhaled by the occupants.

In the assessment, it is conservatively assumed that the combination of the surface erosion rate and the depth of the disposal zone from the ground surface results in the waste being uncovered after 100 000 years, and so the Design Scenario for the GSA also considers potential releases due to erosion.

3.1.2 Alternative scenarios

The GSA also includes four alternative scenarios as part of its approach to the treatment of uncertainty, in order to build confidence in the safety of the disposal system.

- 'The Defect Scenario' – it is assumed that not all components of the near field perform as envisaged in the Design Scenario due to either defective manufacturing of waste packages (e.g. welding defects), or defective implementation in the disposal borehole (e.g. improper emplacement of cement grout). This results in the earlier release of radionuclides from the near field.

- 'The Unexpected Geological Characteristics Scenario' – it is assumed that the actual performance of the geosphere from a safety perspective is worse than the expected performance (e.g. the geosphere is subjected to an unexpected seismic event resulting in the reactivation of high permeability fractures and modification of associated sorption properties). The range of geosphere characteristics assessed in the GSA, combined with the additional geosphere parameter sensitivity analysis undertaken on the Design Scenario, allow the consequences of this scenario to be bounded.

- 'The Changing Environmental Conditions Scenario' – it is assumed that the disposal system is affected by climate change resulting in modifications to certain geosphere characteristics (e.g. groundwater recharge rates) and biosphere characteristics (e.g. water demand, surface erosion rates). The range of geosphere characteristics assessed in the GSA, combined with the additional geosphere and biosphere parameter sensitivity analysis undertaken on the Design Scenario, allow the consequences of this scenario to be bounded. Furthermore, such changes will not have significant impacts on the water uses considered in the GSA since they are applicable to a wide range of differing climatic conditions. Results from a previous GSA [12], which considered an environmental change scenario, further support the screening out of this scenario from more detailed consideration.

- 'The Borehole Disturbance Scenario' – it is assumed that drilling of a water abstraction borehole immediately adjacent to the disposal borehole results in the disturbance of the disposal borehole and the earlier release of radionuclides from the near field and

[2] The zone between the disposal zone and the ground surface.

subsequent exposure of humans to radionuclides (e.g. due to the use of contaminated water from the abstraction borehole). This scenario is screened out from more detailed consideration in the GSA on the grounds of its very low probability due to the disposal borehole's limited footprint, the depth of the disposal zone and the various engineered barriers.

3.2. CONCEPTUAL MODELS FOR RADIONUCLIDE RELEASE AND TRANSPORT: DESIGN SCENARIO

The conceptual models for the Design Scenario for radionuclide release and transport from a generic borehole disposal facility are described in detail in Section 4.2 and Section 5.2 of Ref. [13] and supporting appendices. The conceptual models are summarized below in Section 3.2.1 (Near Field), Section 3.2.2 (Geosphere) and Section 3.2.3 (Biosphere), respectively.

3.2.1. Near field

The near field comprises (see Figure 7):

- The source and its container within which the source material is contained;
- The stainless steel capsule which contains the source and its container;
- The cement containment barrier within the disposal container;
- The stainless steel disposal container;
- The disposal zone cement backfill used to separate disposal containers in the vertical dimension from one another, and in the horizontal dimension from the borehole casing;
- The high density polyethylene casing and carbon steel centralizers which are emplaced at time of drilling;
- The disturbed zone cement backfill used to fill the gap between the casing and the host rock and any voids/cracks in the host rock immediately adjacent to the borehole;
- The disposal zone cement plug at the base of the disposal borehole; and
- The closure zone backfill (assume that the first 5 m from the ground surface is native soil/crushed rock and the remainder is cement).

The conceptual models adopted in the GSA for the evolution of the near field and its various components are summarized in Table 1 and are illustrated in Figures 8 to 11 with particular focus on:

- The status of the near field components at differing times;
- The key assumptions adopted in the GSA;
- The key processes resulting in the failure of the near field components (with associated Features, Events and Processes (FEPs) number in parentheses, taken from Appendix V of Ref. [13]);

- The key processes affecting the release and migration of radionuclides (with associated FEPs number in parentheses, taken from Appendices IV and V of Ref. [13]).

Table 2 summarizes the stainless steel corrosion model included in the GSA for aerobic and anaerobic conditions, while further details on the degradation and corrosion models, and on the release and migration models are given in Section 3.2.1.1 and Section 3.2.1.2, respectively.

3.2.1.1. Degradation and corrosion

Four stages of cement degradation are considered in the GSA:

- Stage 1 – porewater pH is around 13.5, owing to the presence of significant sodium hydroxide, NaOH, and potassium hydroxide, KOH. Such high pH values can persist during flushing by about 100 pore volumes of water. It is assumed that the values for chemical and physical parameters such as sorption coefficient, porosity and hydraulic conductivity are comparable with those for undegraded cement grout.

- Stage 2 – porewater pH has fallen slightly to about 12.5, owing to buffering by Portlandite, $Ca(OH)_2$. This pH can persist during flushing by an additional 900 pore volumes. Although pH has declined slightly, it is assumed that the chemical and physical parameter values are the same as for Stage 1.

- Stage 3 – porewater pH diminishes steadily from 12.5 to about background groundwater pH, owing to buffering with calcium-silicate-hydrate (C-S-H) phases having progressively decreasing Ca/Si ratios. This stage can persist during flushing by approximately an additional 4000 to 9000 pore volumes. There is significant chemical and physical degradation of the cement grout resulting in changes in chemical and physical parameter values. It is assumed that there is a linear change, during Stage 3, in parameter values from the start value (i.e. value for undegraded conditions) to the end value (i.e. value for degraded conditions).

- Stage 4 – porewater pH in the borehole has returned to that of the groundwater surrounding the borehole, and the cement grout is fully degraded. The chemical and physical parameter values are the same as those at the end of Stage 3 (i.e. values for degraded conditions).

As discussed in Appendix VIII of Ref. [13], the exact duration of each stage depends on the composition of the groundwater (in particular groundwater pH) and the rate of groundwater flow (the higher the flow, the more rapid the pore flushes and the more rapid the degradation). As the degradation proceeds from the start to the end of Stage 3, the hydrogeological properties (hydraulic conductivity and porosity) of the cement increase linearly from undegraded values to degraded values (see Equations 60, 61 and 62 in Appendix XI.3.1.2 of Ref. [13]), while the sorption coefficient decreases linearly from undegraded to degraded values (see Equations 65 and 66 in Appendix XI.3.2 of Ref. [13]).

Corrosion of the stainless steel disposal container and capsule is assumed to start at the same time as the degradation of the surrounding cement. Initially, general corrosion rates are low but as the pH falls corrosion rates can become faster. Below some critical pH (depending on the type of steel), the possibility of localized corrosion occurs under aerobic conditions (see Table 2).

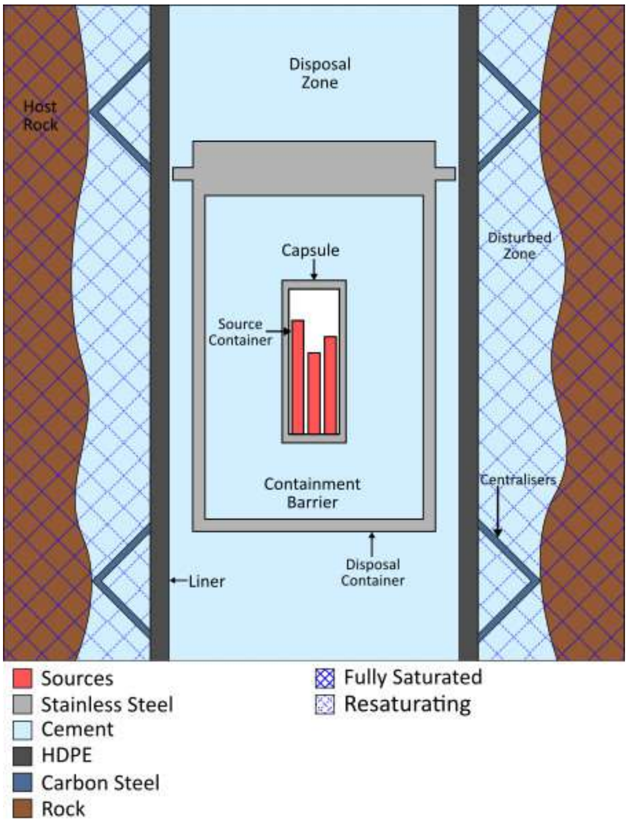

Figure 7. Schematic Representation of the Near field at Time of Borehole Closure (T_0) for Disposal in Saturated Conditions for the Design Scenario.

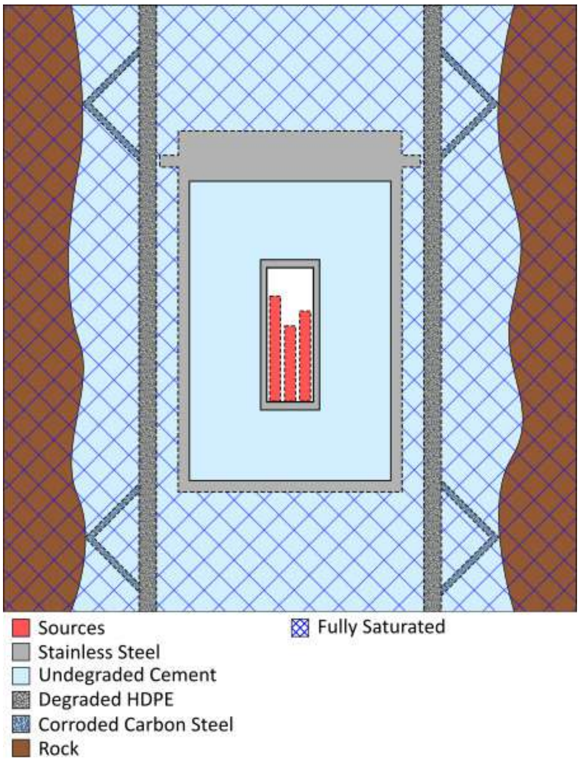

Figure 8. Schematic Representation of the Near field Conceptual Model at Time of Borehole Closure (T_0) for Disposal in Saturated Conditions for the Design Scenario. In comparison to the situation illustrated in Figure 7, for modelling purposes it is assumed that the HDPE liner is degraded, the cement in the disposal zone is already fully saturated at T_0 and the source container has already failed.

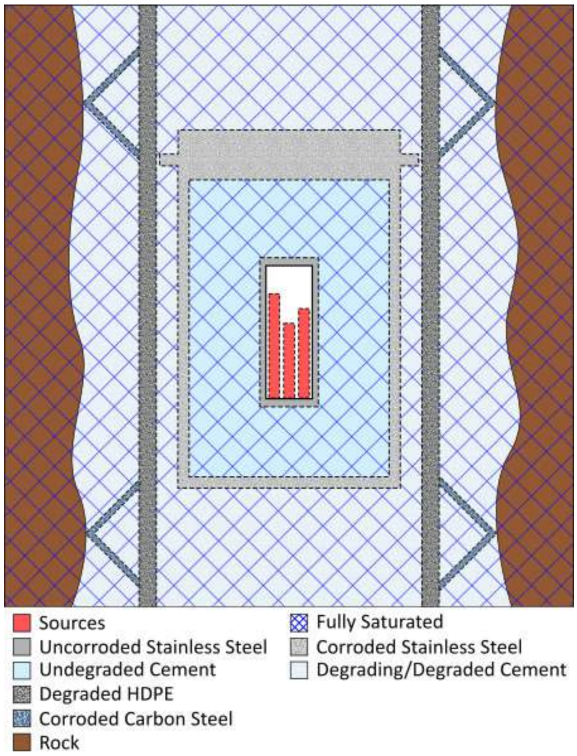

Figure 9. Schematic Representation of the Near field Conceptual Model at Time of Disposal Container Failure (T_I) for Disposal in Saturated Conditions for the Design Scenario.

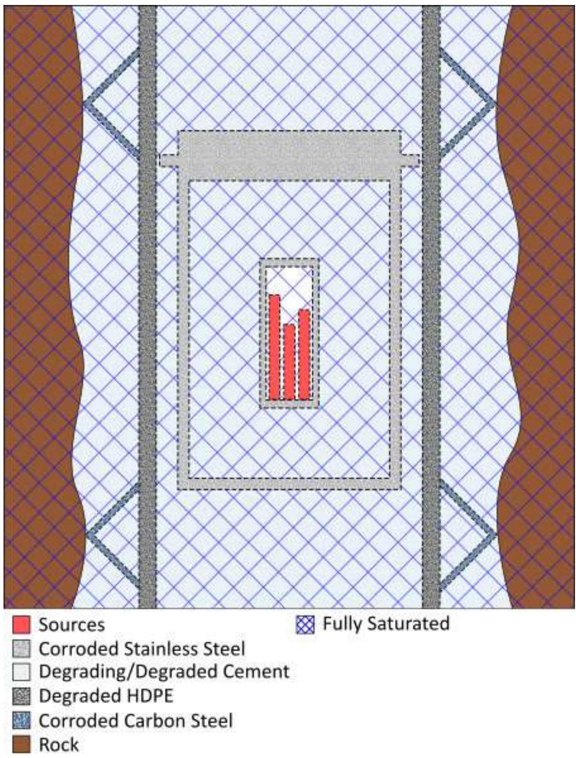

Figure 10. Schematic Representation of the Near field Conceptual Model at Time of Capsule Failure (T_R) for Disposal in Saturated Conditions for the Design Scenario.

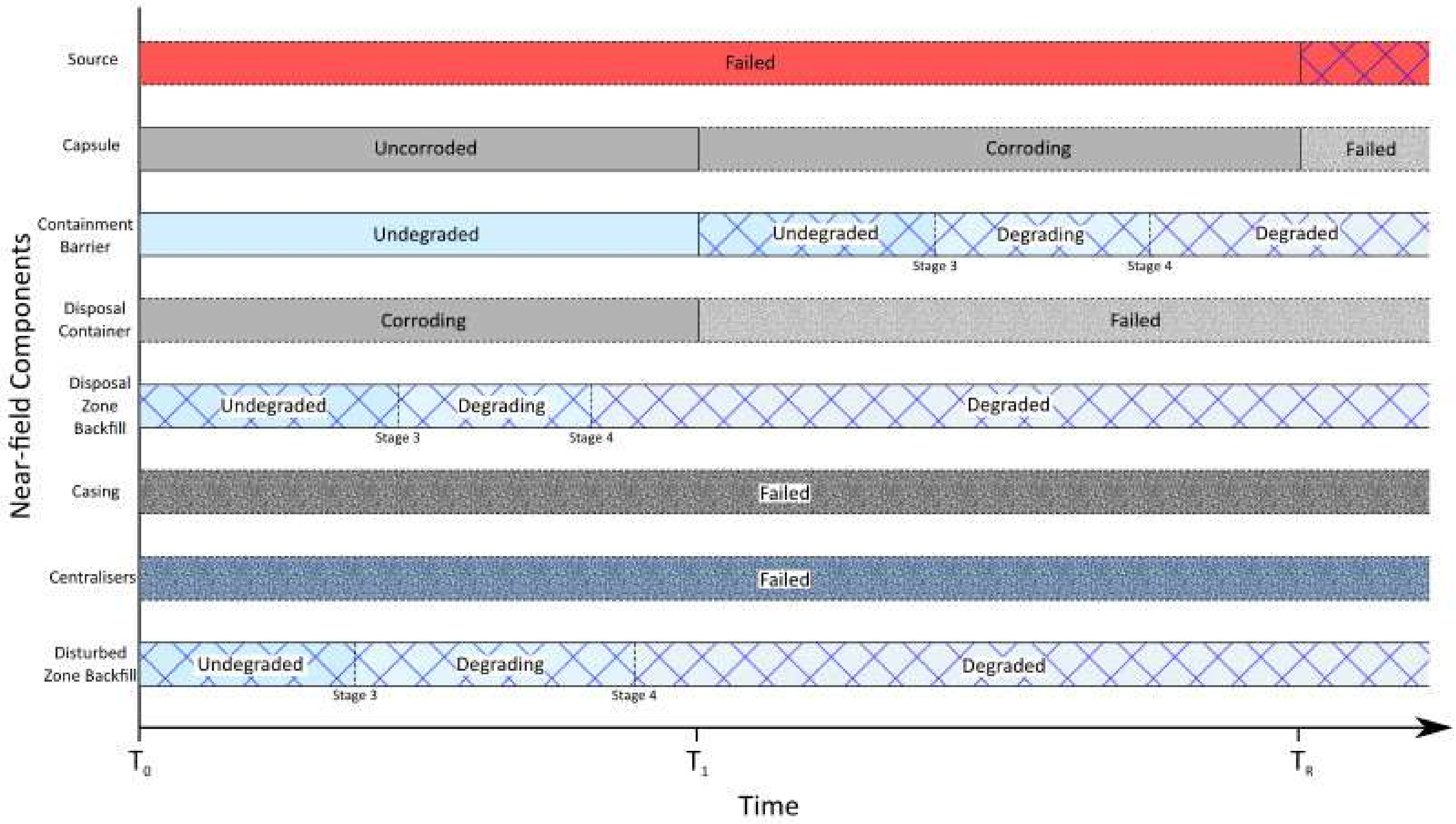

Figure 11. Schematic Representation of the Near field Conceptual Model's Temporal Evolution for Disposal in Saturated Conditions for the Design Scenario.

TABLE 1. SUMMARY OF NEAR FIELD EVOLUTION FOR THE DESIGN SCENARIO

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes Affecting Radionuclides
Increasing time →	Borehole closure (T_0)	Source	Failed	Will have failed since cannot guarantee longevity, but no radionuclides released since capsule still intact.	N/A (already failed) (2.1.3.1)	Decay and ingrowth (3.1.1) Gas generation (2.1.11.6, 3.1.7)
		Capsule	No corrosion	No corrosion since disposal container intact.	N/A (disposal container intact)	N/A (not represented in the transport model)
		Containment barrier	No degradation	No degradation since disposal container intact.	N/A (disposal container intact)	N/A (no releases from capsule)
		Disposal container	Start of corrosion	Due to the failed casing, water will start to infiltrate into the disposal zone and start the corrosion of the disposal container. Corrosion model is given in Section 3.2.1.1.	General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)	
		Backfill (disposal, disturbed & closure zones) & plug	Start of degradation	Due to failed casing, water will start to infiltrate into the disposal zone and start the degradation of the backfill. Degradation model is given in Section 3.2.1.1.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	
		Casing & centralizers	Failed	Fails immediately on closure (considerable uncertainty over casing lifetimes and centralizers have no long-term safety role).	N/A (already failed) (2.1.5.1 & 2.1.5.2)	
	Disposal container failure (T_1) – see Table 76	Source	Failed	Already failed at T_0.	N/A (already failed) (2.1.3.1)	Decay and ingrowth (3.1.1) Gas generation (2.1.11.6, 3.1.7)

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes Affecting Radionuclides
	in Appendix XII.3 of Ref. [13] for times for different illustrative systems	Capsule	Start of corrosion	Due to failed disposal container, water will start to infiltrate into the containment barrier and start the corrosion of the capsule. Corrosion model is given in Section 3.2.1.1.	General and localized corrosion (2.1.3.2 — 2.1.3.8)	N/A (not represented in the transport model)
		Containment barrier	Start of degradation	Due to failed disposal container, water will start to infiltrate into the containment barrier and start the degradation of the cement. Degradation model is given in Section 3.2.1.1.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	N/A (no releases from capsule)
		Disposal container	Failed	If only general corrosion occurs (see Table 2), T_1 occurs once 80% of disposal container wall/weld thickness is corroded. If localized corrosion can occur (see Table 2), T_1 occurs 100 years after the onset of localized corrosion.	General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)	
		Backfill (disposal, disturbed & closure zones) & plug	Degrading/ Degraded	Already started to degrade at T_0. Degradation timings will depend on site-specific conditions taking into account the cement degradation model in Section 3.2.1.1. See Table 76 in Appendix XII.3 of Ref. [13] for Stage 3 start and finish times for different illustrative systems.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	
		Casing & centralizers	Failed	Already failed at T_0.	N/A (already failed) (2.1.5.1)	
	Capsule failure and	Source	Failed	Already failed at T_0 but now that the capsule has failed, radionuclides can be	N/A (already failed) (2.1.3.1)	Decay and ingrowth (3.1.1)

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes Affecting Radionuclides
	radionuclide release (T_R) – see Table 76 in Appendix XII.3 of Ref. [13] for times for different illustrative systems			released due to gas migration and water ingress.		Gas generation (2.1.11.6, 2.1.12, 3.1.7) Gas release (2.1.2.5, 3.1.6) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Groundwater release (2.1.2.2 — 2.1.2.5) Advection, dispersion, diffusion (3.2.4) Precipitation/dissolution (2.1.8.7) Potential for erosive release (3.2.5) depending on depth and erosion rate
		Capsule	Failed	If only general corrosion occurs (see Table 2), T_R occurs once 80% of capsule wall/weld thickness is corroded. If localized corrosion can occur (see Table 2), T_R occurs 100 years after the onset of localized corrosion.	General and localized corrosion (2.1.3.2 — 2.1.3.8)	N/A (not represented in the transport model)
		Containment barrier	Degrading/ Degraded	Already started to degrade at T_1. Degradation timings will depend on site-specific conditions taking into account the cement degradation model in Section 3.2.1.1. See Table 76 in Appendix XII.3 of Ref. [13] for Stage 3 start and finish times for different illustrative systems.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	Decay and ingrowth (3.1.1) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Advection, dispersion, diffusion (3.2.4) Sorption (3.2.2)

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes Affecting Radionuclides
		Disposal container	Failed	Already failed at T_1.	General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)	N/A (not represented in the transport model)
		Backfill (disposal, disturbed & closure zones) & plug	Degrading/ Degraded	Already started to degrade at T_0. Degradation timings will depend on site-specific conditions taking into account the cement degradation model in Section 3.2.1.1. See Table 76 in Appendix XII.3 of Ref. [13] for Stage 3 start and finish times for different illustrative systems.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	Decay and ingrowth (3.1.1) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Advection, dispersion, diffusion (3.2.4) Precipitation/dissolution (2.1.8.7) Sorption (3.2.2)
		Casing & centralizers	Failed	Already failed at T_0.	N/A (already failed) (2.1.5.1)	N/A (not represented in the transport model)

Key

(green)	No degradation/corrosion
(light yellow)	Starting to degrade/corrode
(amber)	Degrading/degraded
(orange-red)	Failed

TABLE 2. SUMMARY OF THE MODEL FOR STAINLESS STEEL CORROSION INCLUDED IN THE GSA

pH Conditions for Different Corrosion Stages		Aerobic conditions	Anaerobic conditions
Stage 1	pH 13.5	General corrosion only	General corrosion only
Stage 2	pH 12.5	General corrosion only	General corrosion only
Stage 3(a)	$pH_{CRIT} < pH < 12.5$	General corrosion only	General corrosion only
Stage 3(b)	$pH_{GW} < pH \leq pH_{CRIT}$	General and localized corrosion	General corrosion only
Stage 4	pH_{GW}	General and localized corrosion	General corrosion only
Note: In the GSA, Ref. [13], pH_{CRIT} is defined as pH 10 for Type 316 stainless steel and pH 11 for Type 304. pH_{GW} is defined as the pH of the surrounding groundwater.			

In the GSA, the rate of general corrosion is assumed to be a time-dependent function of the pH (see above cement degradation model), Eh, and chloride content of the groundwater. Failure of the disposal container/capsule occurs when the total corrosion reaches 80% of its wall thickness, or the depth of the weld thickness, whichever is smaller.

Localized corrosion is assumed to occur if the following set of conditions is met:

- The conditions are aerobic (Eh > 0);
- The pH is below a critical value (10 for Type 316 stainless steel, 11 for Type 304 stainless steel); and
- The chloride concentration is high enough (depends on Eh conditions — see Figure 26b in Appendix IX of Ref. [13]).

If these conditions are met, localized corrosion is then assumed to take 100 years to fail the stainless steel once it initiates.

The timing of the degradation of the cement components and the corrosion of stainless steel components and can be calculated using the BDC Scoping Tool [14], which implements the conceptual models described above and takes into account the hydrogeological and geochemical characteristics of the system being modelled. The values used in the GSA for the different illustrative systems are given in Table 76 in Appendix XII.3 of Ref. [13].

3.2.1.2. Release and migration

Once the capsule is breached, it is possible for radionuclides to be subject to the following release mechanisms (see Table 3):

- Instantaneous release of gas for radionuclides that are in gaseous form (H-3 and Kr-85) or which have gas phase progeny (Rn-222 for Ra-226 and Pu-238);
- Instantaneous dissolution of radionuclides that are in a form (e.g. liquid, soluble solid, surface contamination) that would result in a release to water and then immediate release from the source container (H-3, Ni-63, Sr-90, Cs-137, Pb-210, Ra-226 and Am-241); or

- Congruent release of radionuclides that are in a form (e.g. solids with low solubility) that would result in a slow release to water (Co-60, Pu-238 and Pu-239).

TABLE 3. PHYSICAL/CHEMICAL FORMS OF RADIONUCLIDES AND ASSUMED RELEASE MECHANISM

Radionuclide	Physical/Chemical Form	Release Mechanism
H-3	Often tritium gas or liquid as H_2O	Instantaneous release to air or water (depending upon the form of H-3).
Co-60	Metallic form in thin discs or small cylindrical pellets. Very low solubility	Congruent release (slow rate of release due to low solubility and metallic form).
Ni-63	Solid, electrical deposition on metal foil	Conservatively assume instantaneous release to groundwater since the Ni-63 is on the surface of the metal foil.
Kr-85	Gas	Instantaneous release to air.
Sr-90	Oxide or titanate form Often silver plated for medical applications Ceramic or glass bead or rolled silver foil for other applications	The oxide is likely to be soluble in all waters. There is less information concerning the titanate form, but this is likely to be much less soluble than the oxide. Conservatively assume soluble and instantaneous release to groundwater.
Cs-137	Only used as a salt (often caesium chloride) Sometimes ceramic form for weak sources (very low solubility)	Conservatively assume instantaneous release to groundwater.
Pb-210	Solid, mainly carbonate, and sulphate	Depending upon the water composition the carbonate and sulphate could be very soluble or poorly soluble. Conservatively assume soluble and instantaneous release to groundwater.
Ra-226 (+ Rn gas)	Very reactive alkaline earth metal in form of salts (e.g. bromides, chlorides, sulphates or carbonates). All soluble	Instantaneous release to groundwater. Instantaneous release of radon gas to air.
Pu-238 (+ Rn gas)	Sources (used in Radioisotope Thermoelectric Generators (RTGs) and for neutron generators and calibration) typically have Pu oxide in ceramic	PuO_2 is only poorly soluble. Assume a congruent release with a slow rate of release to groundwater.
Pu-239		
Am-241	Chemical characteristics are similar to rare earth metals. Americium oxides normally used. For neutron sources, fine americium oxide powder used mixed with beryllium powder. Often in pellet form. Sometimes in sintered form	AmO_2 is very soluble. Consequently, it is conservative to assume instantaneous release to groundwater. Under high pH, oxidising conditions. Am-241 could have low solubility and so a congruent release with a slow rate of release to groundwater would be appropriate.

For radionuclides released into groundwater, transport from the source container through the near field can occur by advection, dispersion and diffusion. The relative importance of these processes depends upon the hydrogeological conditions at the site. Migration through the near field is limited by decay and ingrowth, and sorption of the radionuclides onto the cement. It is assumed that the migration is not solubility limited. As noted in Section 3.2.1.1, as the degradation of the cement barriers proceeds from the start to the end of Stage 3, the hydrogeological properties (hydraulic conductivity and porosity) of the cement are assumed to increase linearly from undegraded to degraded values. A linear decrease of the cement's sorption coefficients occurs from undegraded to degraded values.

For radionuclides released in the gas phase, it is assumed that, if the disposal zone is in the saturated zone, H-3 gas is dissolved in the groundwater, whilst Kr-85 and Rn-222 remain in the gas phase. It is conservatively assumed that Kr-85 will migrate directly up the borehole via the closure zone into the biosphere (see Figure 12). The very short half-life of Rn-222 (3.82 days) means that there is likely to be significant decay within the saturated closure zone, although some will reach the unsaturated closure zone and might eventually discharge into the biosphere via the closure zone.

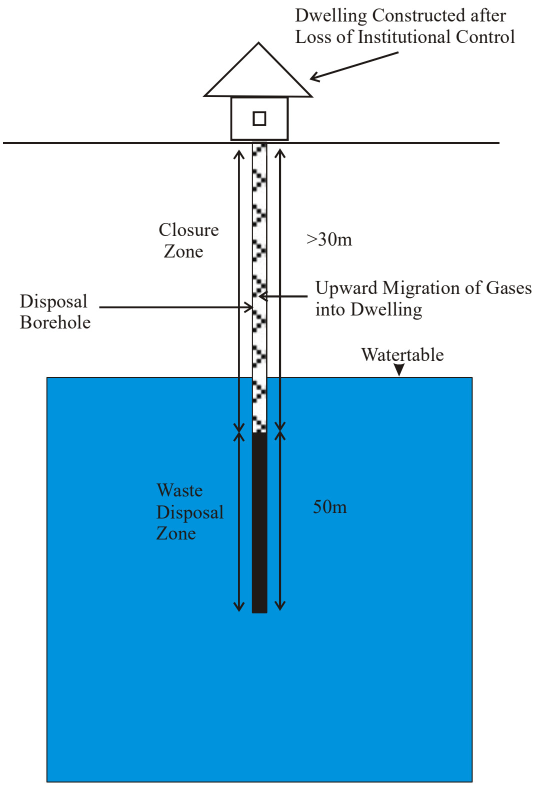

Figure 12. Gas Phase Releases for Disposal in the Saturated Zone for the Design Scenario.

If the disposal zone is in the unsaturated zone, it is assumed that the H-3, Kr-85 and Rn-222 gases remain in the gas phase and migrate up the borehole through the closure zone and into the biosphere (see Figure 13).

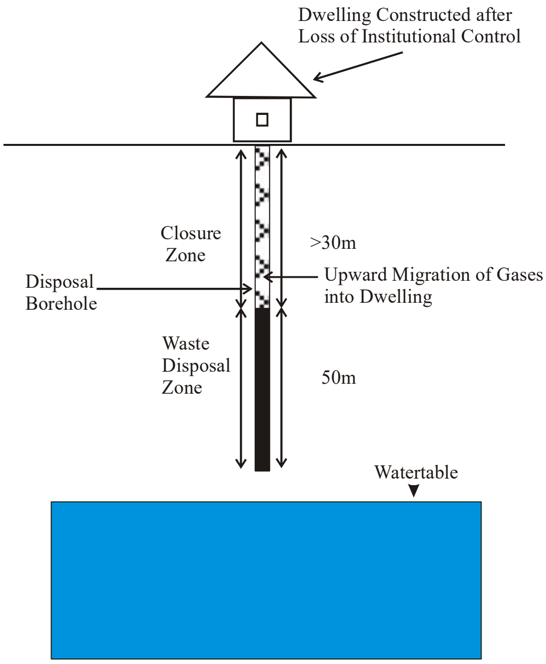

Figure 13. Gas Phase Releases for Disposal in the Unsaturated Zone for the Design Scenario.

There is potential for the release of radionuclides from the near field in the solid phase if the erosion rate of the material above the disposal zone results in the uncovering of the waste (see Figure 14)[3]. The time this would occur would be a function of the erosion rate and the depth of the disposal zone. The erosion rate and the depth of the disposal zone from the ground surface assumed in Ref. [13] results in the waste being uncovered after 100 000 years.

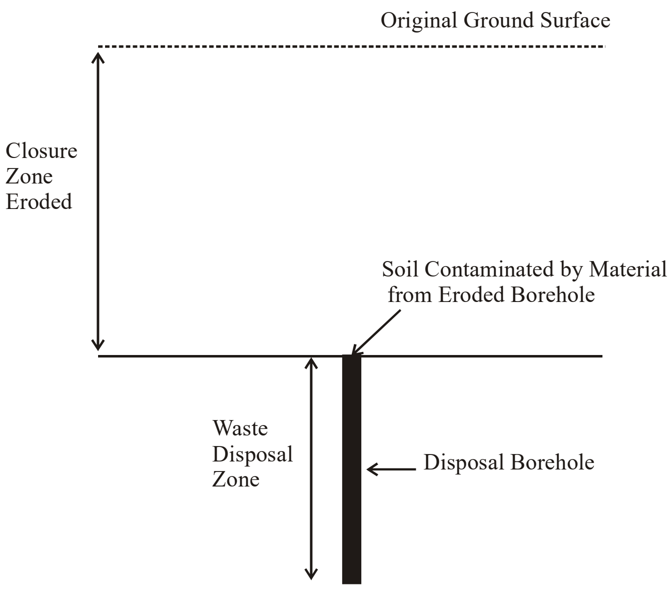

Figure 14. Solid Phase Releases for the Design Scenario.

[3] In reality, scenarios involving erosion of the disposed waste would not occur because the waste would always be disposed of deep enough to avoid this possibility. The erosion scenarios were considered for completeness and can be regarded as a type of hypothetical (zero probability) ‘what if’ scenarios.

3.2.2. Geosphere

For a site-specific assessment, site-specific data relating to the geosphere are collected and collated. However, for a generic assessment such as that undertaken in Ref. [13], it can be helpful to consider a range of 'synthesized' geospheres, which are not based on specific geospheres, but which are representative of potential conditions that might be found in reality. By considering more than one synthesized, time-invariant geosphere, the performance of the borehole disposal facility can be evaluated under a range of geosphere conditions, thereby, helping to define the envelope of site conditions for which the assessment and its results are applicable. In the GSA, the following conditions are considered:

- Disposal in either the unsaturated or saturated geosphere;
- The assumption of either low, medium or high flow rates in the saturated geosphere; and
- The assumption of either porous or fracture flow in the geosphere.

From these conditions, eight potential disposal systems can be identified (see Table 4). Details of each system are provided in Section 3 of Ref. [13].

On leaving the near field, the radionuclides in groundwater migrate through the geosphere by advection, dispersion and diffusion, and are subject to decay and ingrowth, as well as retardation due to sorption onto the rocks. Flow can be through pores or fractures, and diffusion can occur into stagnant water in the rock matrix — depending upon the characteristics of the geosphere. Again, the relative importance of these geosphere processes depends on the hydrogeological conditions at the site. The groundwater is assumed to be abstracted from the geosphere via an abstraction borehole that is located 100 m down the hydraulic gradient from the disposal borehole and the abstraction point is assumed to be at the same depth as the disposal zone. For disposal in the unsaturated zone, transport through the unsaturated zone also needs to be considered down to the underlying saturated zone (see Figure 15). For disposal in the saturated zone, transport through the unsaturated zone does not need to be considered (see Figure 16).

TABLE 4. DIFFERING CHARACTERISTICS OF DISPOSAL SYSTEMS CONSIDERED

<table>
<tr><th>Near Field</th><th colspan="2">Saturated Geosphere</th></tr>
<tr><td>Disposal Zone</td><td>Flow Rate</td><td>Flow Type</td></tr>
<tr><td rowspan="4">Unsaturated</td><td rowspan="2">High</td><td>Porous</td></tr>
<tr><td>Fractured</td></tr>
<tr><td>Medium</td><td>Porous</td></tr>
<tr><td>Low</td><td>Porous</td></tr>
<tr><td rowspan="4">Saturated</td><td rowspan="2">High</td><td>Porous</td></tr>
<tr><td>Fractured</td></tr>
<tr><td>Medium</td><td>Porous</td></tr>
<tr><td>Low</td><td>Porous</td></tr>
</table>

Radionuclides in the gas phase are assumed not to enter the geosphere; they migrate via the closure zone directly to the biosphere (see Figures 12 and 13). The release of radionuclides in the solid phase is also assumed not to enter the geosphere, but to occur directly to the biosphere once the overlying closure zone has been eroded (see Figure 14) – however, see also Footnote 2 in Section 3.2.1.2.

3.2.3. Biosphere

As with the geosphere, a stylized approach is adopted for the biosphere. For radionuclides in the groundwater, the groundwater abstraction borehole is assumed to be used for domestic purposes (drinking) and for agricultural purposes (watering of cows and irrigation of root and green vegetables). The irrigation water is applied to the soil which is used for growing vegetables.

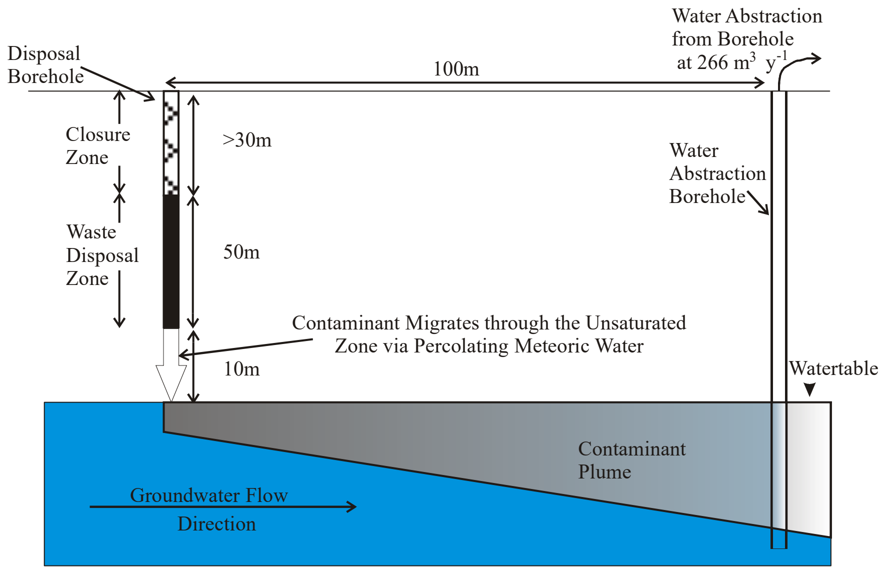

Figure 15. Groundwater Releases for Disposal in the Unsaturated Zone for the Design Scenario.

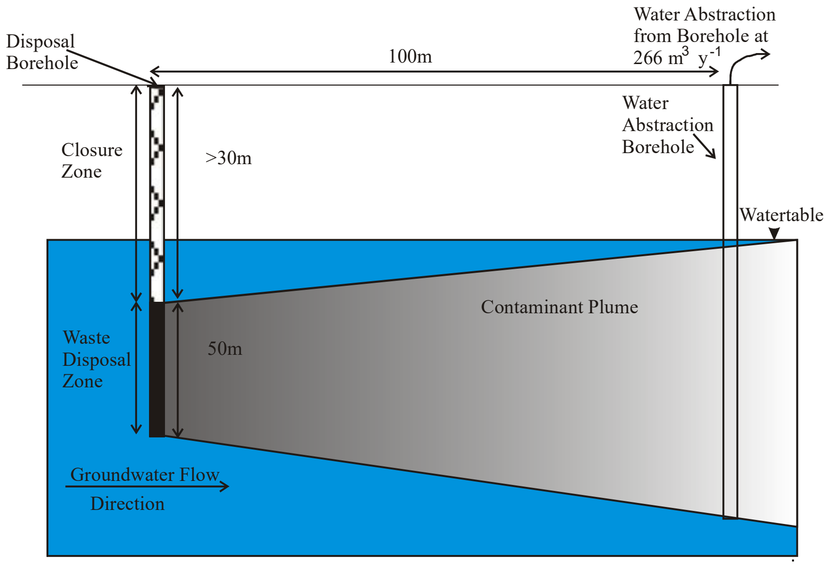

Figure 16. Groundwater Releases for Disposal in the Saturated Zone for the Design Scenario.

Key processes affecting the subsequent migration of radionuclides in the biosphere are:

- Decay and ingrowth;
- Sorption on the soil;
- Percolation through the soil;
- Erosion of the soil;
- Suspension from the soil into the atmosphere;
- Deposition from the irrigation water and the atmosphere onto the vegetables; and
- Root uptake by the vegetables.

Humans may be exposed via ingestion of water, animal products and crops, inadvertent ingestion of soil, external irradiation from the soil, and inhalation of dust.

For radionuclides released from the disposal borehole in the gas phase, it is assumed that a dwelling is constructed on top of the disposal borehole (without intruding into the disposal zone) once institution controls are no longer effective. Radioactive gases are assumed to migrate directly into the dwelling where they may be inhaled by the occupants.

For radionuclides released in the solid phase due to erosion of the closure zone, it is assumed that the contaminated soil is used for growing root and green vegetables by a site dweller. Key processes affecting the subsequent migration of radionuclides through the biosphere are:

- Decay and ingrowth;
- Sorption on the soil;
- Percolation through the soil;
- Erosion of the soil;
- Suspension from the soil into the atmosphere;
- Deposition from the atmosphere onto the vegetables; and
- Root uptake by the vegetables.

Humans are exposed via ingestion of vegetables, inadvertent ingestion of soil, external irradiation from the soil, and inhalation of dust.

3.3. CONCEPTUAL MODELS FOR RADIONUCLIDE RELEASE AND TRANSPORT: DEFECT SCENARIO

The GSA Defect Scenario and its conceptual models for disposal in the saturated zone are described in detail in Section 4.3 and Section 5.2 of Ref. [13] and their supporting appendices, respectively. The GSA assumes that not all components of the near field perform as envisaged in the Design Scenario, resulting in the earlier than expected release of radionuclides from the near field. A range of possible defects involving one or more of four near field barriers (i.e.

capsule, disposal container, disposal zone backfill, and disturbed zone backfill) are considered, hereinafter referred to as D1 to D4:

- D1: all welds are satisfactory due to appropriate quality assurance/quality control (QA/QC) except for the closure weld in one disposal container. All other near field barriers as per the Design Scenario (see Figure 17).

- D2: all welds are satisfactory due to QA/QC except for the closure weld in one capsule. All other near field barriers as per the Design Scenario (see Figure 18).

- D3: degrading/incomplete disposal and disturbed zone cement grout. All other near field barriers as per the Design Scenario (see Figure 19).

- D4: all welds are satisfactory due to QA/QC except for the closure weld in one disposal container and one capsule. The faulty capsule is in the faulty disposal container. All other near field barriers as per the Design Scenario (see Figure 20).

3.3.1. Near field

The evolution of the near field and its various components are summarized below in Tables 5 to 8 with focus on:

- The status of the near field components at differing times;

- The key assumptions adopted in the GSA;

- The key processes resulting in the failure of the near field components (with associated FEPs number in parentheses, taken from Appendix V of Ref. [13]);

- The key processes affecting the release and migration of radionuclides (with associated FEPs number in parentheses, taken from Appendices IV and V of Ref. [13]).

Changes to the conditions assumed for the Design Scenario are identified using a ***bold, italic font***. Further details on the conceptual models for degradation and corrosion, and for release and migration for the Defect Scenario are given in Section 3.3.1.1 and Section 3.3.1.2, respectively.

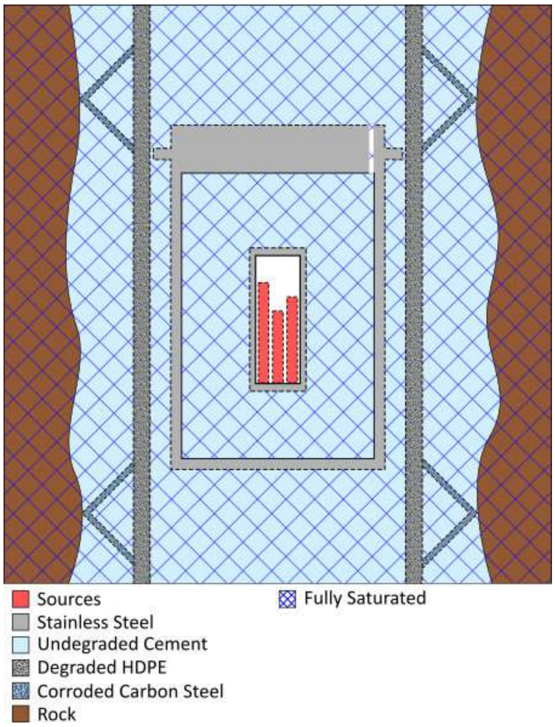

Figure 17. Schematic Representation of the Near field Conceptual Model at Time of Borehole Closure (T_0) for Disposal in Saturated Conditions for the Defect Scenario D1.

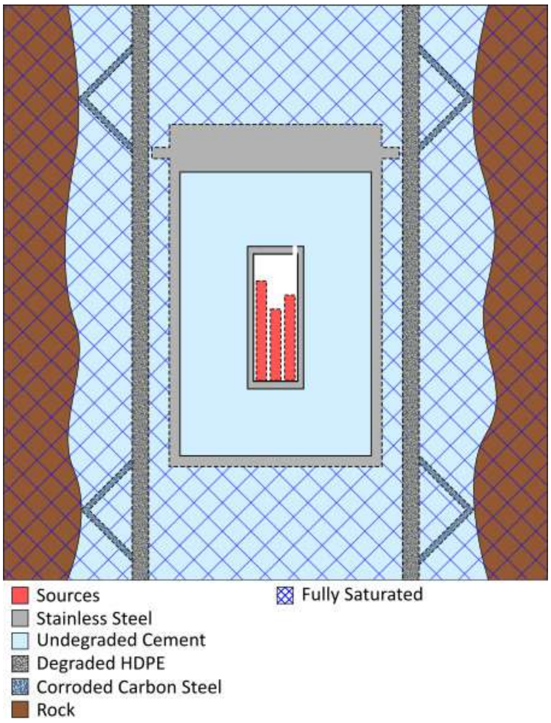

Figure 18. Schematic Representation of the Near field Conceptual Model at Time of Borehole Closure (T_0) for Disposal in Saturated Conditions for the Defect Scenario D2.

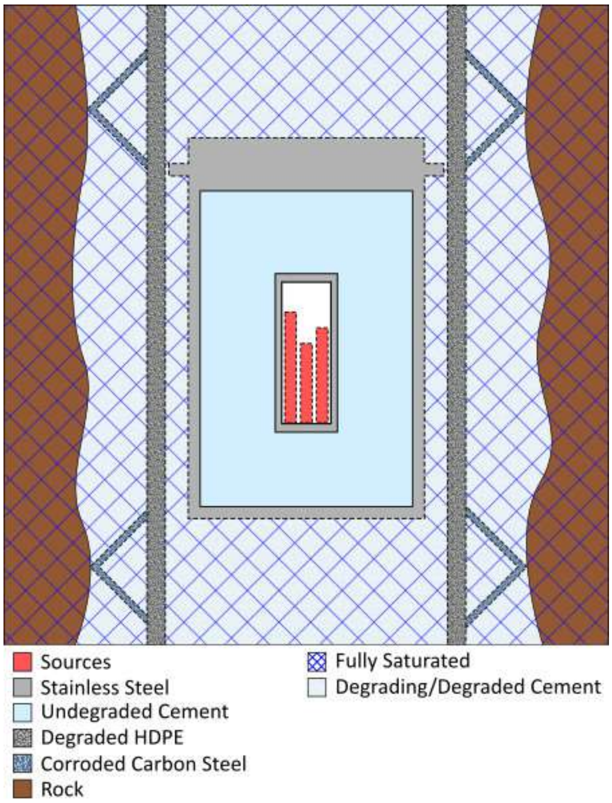

Figure 19. Schematic Representation of the Near field Conceptual Model at Time of Borehole Closure (T_0) for Disposal in Saturated Conditions for the Defect Scenario D3.

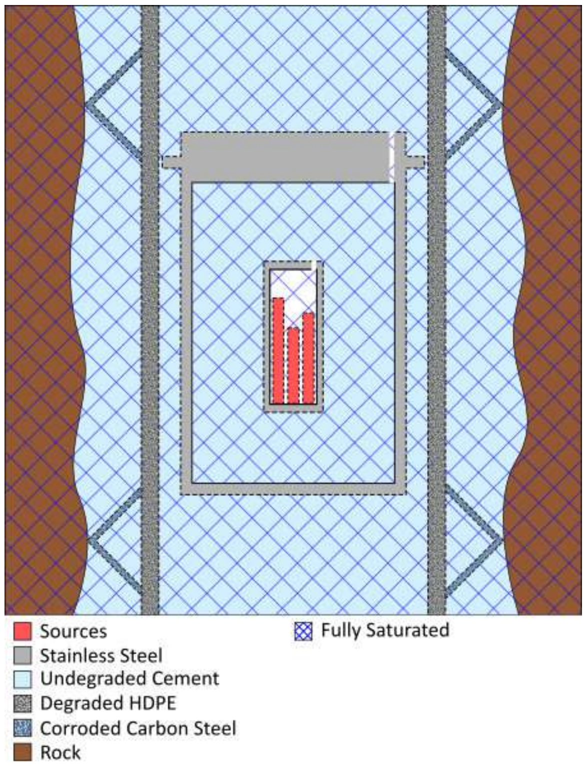

Figure 20. Schematic Representation of the Near field Conceptual Model at Time of Borehole Closure (T_0) for Disposal in Saturated Conditions for the Defect Scenario D4.

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TABLE 5. SUMMARY OF NEAR FIELD EVOLUTION FOR THE DEFECT SCENARIO D1

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes Affecting Radionuclides
Increasing time →	Borehole closure (T_0)	Source	Failed	Will have failed since cannot guarantee longevity, but no radionuclides released since capsule still intact.	N/A (already failed) (2.1.3.1)	Decay and ingrowth (3.1.1) Gas generation (2.1.11.6, 3.1.7)
		Capsule (in defective disposal container)	***Start of corrosion***	***Due to defective weld in disposal container, water will start to infiltrate into the containment barrier and start the corrosion of the capsule. Corrosion model is given in Section 3.3.1.1.***	***General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)***	N/A (not represented in the transport model)
		Containment barrier (in defective disposal container)	***Start of degradation***	***Due to defective weld in disposal container, water will start to infiltrate into the containment barrier and start the degradation of the cement. Degradation model is given in Section 3.3.1.1.***	***Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)***	N/A (no releases from capsule)
		Disposal container	***Partly failed***	***The disposal container has an undetected defective weld. Due to failed casing, water will start to infiltrate into the disposal zone and start the corrosion of the disposal container, especially around the weld. Corrosion model is given in Section 3.3.1.1.***	***Manufacturing fault General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)***	
		Backfill (disposal, disturbed & closure zones) & plug	Start of degradation	Due to failed casing, water will start to infiltrate into the disposal zone and start the degradation of the backfill. Degradation model is given in Section 3.3.1.1.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	
		Casing & centralizers	Failed	Fails immediately on closure (considerable uncertainty over casing lifetimes and centralizers have no long-term safety role).	N/A (already failed) (2.1.5.1 & 2.1.5.2)	
	Disposal container totally failed (T_1) – see Table 77 in Appendix	Source	Failed	Already failed at T_0.	N/A (already failed) (2.1.3.1)	Decay and ingrowth (3.1.1) Gas generation (2.1.11.6, 3.1.7)
		Capsule (in defective	***Corroding***	***Already started to corrode at T_0 due to ingress of water through defective weld in the disposal container. Time of failure will depend on site-***	***General and localized corrosion (2.1.3.2-2.1.3.8)***	N/A (not represented in the transport model)

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes Affecting Radionuclides
	XII.3 of Ref. [13] for times for different illustrative systems	***disposal container)***		***specific conditions taking into account corrosion model in Section 3.3.1.1. See Table 77 in Appendix XII.3 of Ref. [13] for times for different illustrative systems.***		
		Containment barrier (in defective disposal container)	***Degrading***	***Already started to degrade at T_0. Degradation timings will depend on site-specific conditions taking into account the cement degradation model in Section 3.3.1.1. See Table 77 in Appendix XII.3 of Ref. [13] for Stage 3 start and finish times for different illustrative systems.***	***Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)***	N/A (no releases from capsule)
		Disposal container	Failed	If only general corrosion occurs (see Table 2), T_1 occurs once 80% of disposal container wall/weld thickness is corroded. If localized corrosion can occur (see Table 2), T_1 occurs 100 years after the onset of localized corrosion.	General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)	
		Backfill (disposal, disturbed & closure zones) & plug	Degrading/ Degraded	Already started to degrade at T_0. Degradation timings will depend on site-specific conditions taking into account the cement degradation model in Section 3.3.1.1. See Table 77 in Appendix XII.3 of Ref. [13] for Stage 3 start and finish times for different illustrative systems.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	
		Casing & centralizers	Failed	Already failed at T_0.	N/A (already failed) (2.1.5.1)	
	Capsule failure and radionuclide release (T_R) – see Table 77 of Ref. [13] for times for different illustrative systems	Source	Failed	Already failed at T_0 but now that the capsule has failed, radionuclides can be released due to gas migration and water ingress.	N/A (already failed) (2.1.3.1)	Decay and ingrowth (3.1.1) Gas generation (2.1.11.6, 2.1.12, 3.1.7) Gas release (2.1.2.5, 3.1.6) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Groundwater release (2.1.2.2 — 2.1.2.5) Advection, dispersion, diffusion (3.2.4) Precipitation/dissolution (2.1.8.7)

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes Affecting Radionuclides
						Potential for erosive release (3.2.5) depending on depth and erosion rate
		Capsule (in defective disposal container)	Failed	If only general corrosion occurs (see Table 2), T_R occurs once 80% of capsule wall/weld thickness is corroded. If localized corrosion can occur (see Table 2), T_R occurs 100 years after the onset of localized corrosion.	General and localized corrosion (2.1.3.2 — 2.1.3.8)	N/A (not represented in the transport model)
		Containment barrier (in defective disposal container)	Degrading/ Degraded	Already started to degrade at T_0. Degradation timings will depend on site-specific conditions taking into account the cement degradation model in Section 3.3.1.1. See Table 77 in Appendix XII.3 of Ref. [13] for Stage 3 start and finish times for different illustrative systems.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	Decay and ingrowth (3.1.1) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Advection, dispersion, diffusion (3.2.4) Sorption (3.2.2)
		Disposal container	Failed	Already failed at T_1.	General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)	N/A (not represented in the transport model)
		Backfill (disposal, disturbed & closure zones) & plug	Degrading/ Degraded	Already started to degrade at T_0. Degradation timings will depend on site-specific conditions taking into account the cement degradation model in Section 3.3.1.1. See Table 77 in Appendix XII.3 of Ref. [13] for Stage 3 start and finish times for different illustrative systems.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	Decay and ingrowth (3.1.1) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Advection, dispersion, diffusion (3.2.4) Precipitation/dissolution (2.1.8.7) Sorption (3.2.2)
		Casing & centralizers	Failed	Already failed at T_0.	N/A (already failed) (2.1.5.1)	N/A (not represented in the transport model)

Key

(green)	No degradation/corrosion
(light yellow)	Starting to degrade/corrode
(amber)	Degrading/degraded
(salmon)	Partly failed/failed

TABLE 6. SUMMARY OF NEAR FIELD EVOLUTION FOR THE DEFECT SCENARIO D2

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes affecting Radionuclides
Increasing time →	Borehole closure (T_0)	Source	Failed	Will have failed since cannot guarantee longevity, but no radionuclides released since disposal container still intact.	N/A (already failed) (2.1.3.1)	Decay and ingrowth (3.1.1) Gas generation (2.1.11.6, 3.1.7) Gas release (2.1.2.5, 3.1.6) Gas migration (2.1.12)
		Defective capsule	***Partly failed***	***The capsule has an undetected defective weld. Despite defective weld in the capsule, there is no corrosion or radionuclide release since disposal container intact and no water can enter the disposal container.***	***Manufacturing fault***	N/A (not represented in the transport model)
		Containment barrier	No degradation	No degradation corrosion since disposal container intact.	N/A (disposal container intact)	Decay and ingrowth of radioactive gases (3.1.1) Gas migration (2.1.12)
		Disposal container	Start of corrosion	Due to the failed casing, water will start to infiltrate into the disposal zone and start the corrosion of the disposal container. Corrosion model is given in Section 3.3.1.1.	General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)	N/A (no releases from disposal container)
		Backfill (disposal, disturbed & closure zones) & plug	Start of degradation	Due to failed casing, water will start to infiltrate into the disposal zone and start the degradation of the backfill. Degradation model is given in Section 3.3.1.1.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	
		Casing & centralisers	Failed	Fails immediately on closure (considerable uncertainty over casing lifetimes and centralisers have no long-term safety role).	N/A (already failed) (2.1.5.1 & 2.1.5.2)	
	Disposal container failure (T_1) and radionuclide release (T_R) – see Table 77 in	Source	Failed	Already failed at T_0 but now that the disposal container has failed, radionuclides can be released due to gas migration and water ingress via the failed disposal container and defective capsule.	N/A (already failed) (2.1.3.1)	Decay and ingrowth (3.1.1) Gas generation (2.1.11.6, 2.1.12, 3.1.7) Gas release (2.1.2.5, 3.1.6) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Groundwater release (2.1.2.2 — 2.1.2.5)

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes affecting Radionuclides
	Appendix XII.3 of Ref. [13] for times for different illustrative systems					Advection, dispersion, diffusion (3.2.4) Precipitation/dissolution (2.1.8.7) Potential for erosive release (3.2.5) depending on depth and erosion rate
		Defective capsule	***Partly failed***	***Due to failed disposal container, water will start to infiltrate into the containment barrier and start the corrosion of the capsule, especially around the defective weld. Corrosion model is given in Section 3.3.3.1.*** ***If only general corrosion occurs (see Table 2), total failure occurs once 80% of capsule wall/weld thickness is corroded. If localized corrosion can occur (see Table 2), total failure occurs 100 years after the onset of localized corrosion.*** ***A fraction of waste available for release increases linearly (Section 3.3.1.1).***	***General and localized corrosion (2.1.3.2 — 2.1.3.8)***	N/A (not represented in the transport model)
		Containment barrier	Start of degradation	Due to failed disposal container, water will start to infiltrate into the containment barrier and start the degradation of the cement. Degradation model is given in Section 3.3.1.1.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	Decay and ingrowth (3.1.1) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Advection, dispersion, diffusion (3.2.4) Sorption (3.2.2)
		Disposal container	Failed	If only general corrosion occurs (see Table 2), T_1 occurs once 80% of disposal container wall/weld thickness is corroded. If localized corrosion can occur (see Table 2), T_1 occurs 100 years after the onset of localized corrosion.	General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)	N/A (not represented in the transport model)

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes affecting Radionuclides
		Backfill (disposal, disturbed & closure zones) & plug	Degrading/ Degraded	Already started to degrade at T_0. Degradation timings will depend on site-specific conditions taking into account the cement degradation model in Section 3.3.1.1. See Table 77 in Appendix XII.3 of Ref. [13] for Stage 3 start and finish times for different illustrative systems.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	Decay and ingrowth (3.1.1) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Advection, dispersion, diffusion (3.2.4) Precipitation/dissolution (2.1.8.7) Sorption (3.2.2)
		Casing & centralizers	Failed	Already failed at T_0.	N/A (already failed) (2.1.5.1)	N/A (not represented in the transport model)

Key

	No degradation/corrosion
	Starting to degrade/corrode
	Degrading/degraded
	Partly failed/failed

TABLE 7. SUMMARY OF NEAR FIELD EVOLUTION FOR THE DEFECT SCENARIO D3

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes Affecting Radionuclides
Increasing time →	Borehole closure (T_0)	Source	Failed	Will have failed since cannot guarantee longevity, but no radionuclides released since capsule still intact.	N/A (already failed) (2.1.3.1)	Decay and ingrowth (3.1.1) Gas generation (2.1.11.6, 3.1.7)
		Capsule	No corrosion	No corrosion since disposal container intact.	N/A (disposal container intact)	N/A (not represented in the transport model)
		Containment barrier	No degradation	No degradation since disposal container intact.	N/A (disposal container intact)	N/A (no releases from capsule)
		Disposal container	Start of corrosion	Due to the failed casing, water will start to infiltrate into the disposal zone and start the corrosion of the disposal container. Corrosion model is given in Section 3.3.1.1.	General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)	
		Backfill (disposal, disturbed & closure zones) & plug	***Degrading/ Incomplete***	***Faster degradation than Design Scenario – see Table 77 in Appendix XII.3 of Ref. [13] for Stage 3 start and finish times for different illustrative systems. Degradation model is given in Section 3.3.1.1.***	***Increased degradation of backfill/ incomplete backfilling of disposal borehole Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)***	
		Casing & centralisers	Failed	Fails immediately on closure (considerable uncertainty over casing lifetimes and centralisers have no long-term safety role).	N/A (already failed) (2.1.5.1 & 2.1.5.2)	
	Disposal container failure (T_1) – see Table 77 in Appendix XII.3 of Ref.	Source	Failed	Already failed at T_0.	N/A (already failed) (2.1.3.1)	Decay and ingrowth (3.1.1) Gas generation (2.1.11.6, 3.1.7)
		Capsule	Start of corrosion	Due to failed disposal container, water will start to infiltrate into the containment barrier and start the corrosion of the	General and localized corrosion (2.1.3.2 — 2.1.3.8)	N/A (not represented in the transport model)

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes Affecting Radionuclides
	[13] for times for different illustrative systems			capsule. Corrosion model is given in Section 3.3.1.1.		
		Containment barrier	Start of degradation	Degradation timings are faster than Design Scenario and will depend on site-specific conditions taking into account the cement degradation model in Section 3.3.1.1. See Table 77 in Appendix XII.3 of Ref. [13] for Stage 3 start and finish times for different illustrative systems.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	N/A (no releases from capsule)
		Disposal container	Failed	If only general corrosion occurs (see Table 2), T_1 occurs once 80% of disposal container wall/weld thickness is corroded. If localized corrosion can occur (see Table 2), T_1 occurs 100 years after the onset of localized corrosion.	General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)	
		Backfill (disposal, disturbed & closure zones) & plug	Degrading/ Degraded	Degradation timings are faster than Design Scenario and will depend on site-specific conditions taking into account the cement degradation model in Section 3.3.1.1. See Table 77 in Appendix XII.3 of Ref. [13] for Stage 3 start and finish times for different illustrative systems.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	
		Casing & centralisers	Failed	Already failed at T_0.	N/A (already failed) (2.1.5.1)	
	Capsule failure and radionuclide release (T_R) – see Table 77 in Appendix XII.3 of Ref. [13] for times for different	Source	Failed	Already failed at T_0 but now that the capsule has failed, radionuclides can be released due to gas migration and water ingress.	N/A (already failed) (2.1.3.1)	Decay and ingrowth (3.1.1) Gas generation (2.1.11.6, 2.1.12, 3.1.7) Gas release (2.1.2.5, 3.1.6) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Groundwater release (2.1.2.2 — 2.1.2.5) Advection, dispersion, diffusion (3.2.4) Precipitation/dissolution (2.1.8.7)

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes Affecting Radionuclides
	illustrative systems					Potential for erosive release (3.2.5) depending on depth and erosion rate
		Capsule	Failed	If only general corrosion occurs (see Table 2), T_R occurs once 80% of capsule wall/weld thickness is corroded. If localized corrosion can occur (see Table 2), T_R occurs 100 years after the onset of localized corrosion.	General and localized corrosion (2.1.3.2 — 2.1.3.8)	N/A (not represented in the transport model)
		Containment barrier	Degrading/ Degraded	Degradation timings are faster than Design Scenario and will depend on site-specific conditions taking into account the cement degradation model in Section 3.3.1.1. See Table 77 of Ref. [13] for Stage 3 start and finish times for different illustrative systems.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	Decay and ingrowth (3.1.1) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Advection, dispersion, diffusion (3.2.4) Sorption (3.2.2)
		Disposal container	Failed	Already failed at T_1.	General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)	N/A (not represented in the transport model)
		Backfill (disposal, disturbed & closure zones) & plug	Degarding/ Degraded	Degradation timings are faster than Design Scenario and will depend on site-specific conditions taking into account the cement degradation model in Section 3.3.1.1. See Table 77 in Appendix XII.3 of Ref. [13] for Stage 3 start and finish times for different illustrative systems.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	Decay and ingrowth (3.1.1) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Advection, dispersion, diffusion (3.2.4) Precipitation/dissolution (2.1.8.7) Sorption (3.2.2)
		Casing & centralisers	Failed	Already failed at T_0.	N/A (already failed) (2.1.5.1)	N/A (not represented in the transport model)

Key

(green)	No degradation/corrosion
(yellow)	Starting to degrade/corrode
(orange)	Degrading/degraded
(salmon)	Failed

TABLE 8. SUMMARY OF NEAR FIELD EVOLUTION FOR THE DEFECT SCENARIO D4

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes affecting Radionuclides
Increasing time →	Borehole closure (T_0), disposal container failure (T_1) and radionuclide release (T_R) – see Table 77 in Appendix XII.3 of Ref. [13] for times for different illustrative systems	***Source***	***Failed***	***Will have failed since cannot guarantee longevity. Since the defective capsule is in the defective disposal container, water can ingress into the capsule and the source resulting in the release of radionuclides.***	***N/A (already failed) (2.1.3.1)***	Decay and ingrowth (3.1.1) Gas generation (2.1.11.6, 2.1.12, 3.1.7) Gas release (2.1.2.5, 3.1.6) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Groundwater release (2.1.2.2 — 2.1.2.5) Advection, dispersion, diffusion (3.2.4) Precipitation/dissolution (2.1.8.7) Erosive release (3.2.5) will only occur after source is exposed
		Defective capsule	***Partly failed***	***The capsule has an undetected defective weld. Since the defective capsule is in the defective disposal container, water can ingress into the capsule and there is radionuclide release. Corrosion of the capsule starts, especially around the defective weld. Corrosion model is given in Section 3.3.1.1. If only general corrosion occurs (see Table 2), total failure occurs once 80% of capsule wall/weld thickness is corroded. If localized corrosion can occur (see Table 2), total failure occurs 100 years after the onset of localized corrosion. A fraction of waste available for release increases linearly (Section 3.3.1.1).***	***Manufacturing fault General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)***	N/A (not represented in the transport model)

	Time	Near field Component	Status	Key Assumptions	Key Failure Processes	Key Processes affecting Radionuclides
		Containment barrier (in defective disposal container)	***Start of degradation***	***Due to defective weld in disposal container, water will start to infiltrate into the containment barrier and start the degradation of the cement. Degradation model is given in Section 3.3.1.1.***	***Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)***	Decay and ingrowth (3.1.1) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Advection, dispersion, diffusion (3.2.4) Sorption (3.2.2)
		Disposal container	***Partly failed***	***The disposal container has an undetected defective weld. Corrosion of the disposal container starts, especially around the defective weld. Corrosion model is given in Section 3.3.1.1. If only general corrosion occurs (see Table 2), total failure occurs once 80% of capsule wall/weld thickness is corroded. If localized corrosion can occur (see Table 2), total failure occurs 100 years after the onset of localized corrosion.***	***Manufacturing fault General corrosion (2.1.3.2), localized corrosion (2.1.3.3), weld attack (2.1.3.8)***	N/A (not represented in the transport model)
		Backfill (disposal, disturbed & closure zones) & plug	Start of degradation	Due to failed casing, water will start to infiltrate into the disposal zone and start the degradation of the backfill. Degradation model is given in Section 3.3.1.1.	Chlorine attack (2.1.4.2), sulphate attack (2.1.4.3), carbonation (2.1.4.4), fracturing (2.1.6.3), mineralization (2.1.8.4)	Decay and ingrowth (3.1.1) Gas migration (2.1.7.4, 2.1.12, 3.2.7) Advection, dispersion, diffusion (3.2.4) Precipitation/dissolution (2.1.8.7) Sorption (3.2.2)
		Casing & centralisers	Failed	Fails immediately on closure (considerable uncertainty over casing lifetimes and centralisers have no long-term safety role).	N/A (already failed) (2.1.5.1 & 2.1.5.2)	N/A (not represented in the transport model)

Key

(yellow)	Starting to degrade/corrode
(orange)	Partly failed/failed

3.3.1.1. Degradation and corrosion

The same conceptual models of cement degradation and stainless steel corrosion are used as for the Design Scenario, with the following modifications:

- Degradation model — shorter degradation stages are assumed for the Defect Scenario D3 due to the incomplete or degraded cement grout (see Table 77 in Appendix XII.3 of Ref. [13] for times for different illustrative systems);

- Corrosion model — the lifetimes of the affected disposal container(s)/capsule(s) will be reduced due to the earlier onset of corrosion, although the processes will be the same as the Design Scenario (see Table 77 in Appendix XII.3 of Ref. [13] for times for different illustrative systems). Defect Scenario D3 (incomplete or degraded cement grout) compromizes the ability of the cement grout to condition the near field pH and can result in substantially shorter lifetimes in aerobic environments because of the possibility of rapid localized corrosion failure of the disposal container.

3.3.1.2. Release and migration

The processes considered for the Defect Scenario are the same as those for the Design Scenario since the faster degradation rates, earlier failure times, and faster radionuclide migration times of the Defect Scenario can be accounted for by modifying the associated parameters in the mathematical model (e.g. container degradation rates) rather than considering different processes.

For release into groundwater, it is assumed that the defective capsule/waste container is at the base of the disposal zone for disposal in the unsaturated zone, thereby minimizing the travel distance to the saturated zone. For disposal in the saturated zone, the position of the defective capsule/container is not important since the flow from the disposal borehole to the abstraction borehole is assumed to be horizontal. For the Defect Scenarios D1 and D3, all the waste available for release is assumed to be released as soon as the capsule has failed. For the Defect Scenarios D2 and D4, the fraction of the waste available for release is assumed to be 10%, starting from the time of initial water ingress into the capsule up to the time when the capsule is fully failed, i.e. when all of the waste is assumed to be available (see Equation 56 in Appendix XI.3.1.1 of Ref. [13]).

For gas releases, it is assumed that the defective capsule/waste container is at the top of the disposal zone.

3.3.2. Geosphere

The same conceptual model is used as for the Design Scenario (see Section 3.3.1).

3.3.3. Biosphere

The same conceptual model is used as for the Design Scenario (see Section 3.3.1).

4. HIERARCHY OF CONCEPTUAL AND MATHEMATICAL MODELS

4.1. GENERAL

Section 3 described the scenarios and conceptual models that were defined and used in the GSA [13]. These scenarios and conceptual models provide the basis for an in depth evaluation of the potential exposures which people living near a borehole disposal facility might receive.

As discussed in Section 2, in accordance with the graded approach, there may be circumstances (e.g. where the amount of waste to be disposed of is small, or where a particular site has favourable properties) in which it is possible to use less complex safety assessment models and, based on these, still develop sufficient confidence that the disposal system will be safe. The application of simpler models would be easier, may require less site-specific input data, and could be easier to communicate to interested parties.

The application of such simpler models could have different purposes:

- A simpler model can be used for scoping purposes to identify relevant radionuclides and/or relevant site parameters and processes to be investigated further in subsequent, more detailed modelling steps.

- The confidence in the results of detailed site-specific models may be increased by comparison with results from simpler models, which are – in general – easier to understand for a broader audience.

- If a relatively simple model can be shown to be conservative and demonstrates that exposures from the facility will be well below regulatory criteria (e.g. dose limits and constraints), there may be no need for any further detailed modelling and the safety of the facility can be demonstrated based on this relatively simple model.

The purpose of this section is to provide a detailed description of the hierarchy of conceptual and mathematical models discussed in this publication that can be used in a graded approach to support the safety assessment and safety case development process for the BDS. For this purpose, Section 4.2 defines the assessment tiers used in this publication, while Section 4.3 provides a more detailed description of the conceptual models used for these tiers. Section 4.4 describes the mathematical models defined for this purpose.

4.2. DEFINITION OF ASSESSMENT TIERS

In view of the above mentioned considerations, the conceptual models described in Section 3 were reviewed to identify possible simplifications concerning the assumptions made and the needs for site-specific data.

The result has been the description of a hierarchy of models at different levels of complexity. This hierarchy of models provides the possibility for States or organisations considering borehole disposal of DSRS to follow a graded approach to safety assessment modelling. Such States or organisations could begin by making simple, conservative safety assessments and they would then only progress to developing and using more realistic, complex models as needed, according to the level of risk posed by the disposed waste.

The hierarchy of models is described in terms of a number of levels (tiers) of increasing model complexity (from Tier 1 — the simplest model, up to Tier 5 — the most complex model), as defined below and described more comprehensively in Section 4.3:

- Tier 1: Assessment of toxicity of radionuclide inventory

  The Tier 1 assessment is a simple approach which assumes direct ingestion of the waste is applied to assess the radiotoxicity of the radionuclide inventory to be disposed of. This simple assessment does not intend to describe any real exposure situation, but just serves the purpose of identifying which radionuclides in the waste can be of any potential concern.

  An additional or alternative Tier 1 assessment option would be to carry out a first screening of hypothetical exposures arising from the borehole disposal facility under very simple and extremely conservative assumptions.

  These approaches are discussed in Appendix I of Ref. [13].

- Tier 2: Comparison with predefined values

  The Tier 2 assessment is based on values which have been derived elsewhere to get further insight into the relevance of the inventory to be disposed of. To this end, the significance of radionuclides can be assessed against the clearance values established in IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [31].

  For radionuclides with relevant radionuclide concentrations, a comparison against the waste acceptance criteria (WAC) derived in the GSA [13] for different environments can be made. Based on this comparison, radionuclides of high concern (reaching or exceeding the GSA WAC) and those of lesser concern (leading to average borehole concentrations near or below the clearance levels) can be identified.

- Tier 3: Use of BDC Scoping Tool

  The BDC Scoping Tool [14] (see Section 1.1.3) provides a simple-to-use option for some scoping dose calculations for the groundwater pathway. These are based on the degradation model for the near field barriers described in Section 3 and on a very simple geosphere and biosphere model. Key hydrogeological and hydrogeochemical site parameters (e.g., saturated vs. unsaturated conditions, pH, Eh, SO_4, Cl, flow rate) can be provided as a basis for the degradation modelling. The geosphere model is very simplified and does not, for example, take into consideration the retardation of radionuclides by sorption.

  In addition to the groundwater model, the BDC Scoping Tool contains a model which addresses exposures due to gas releases from the disposal facility. This model is identical to the model described in Section 3.

  The solid release exposure mechanism is not explicitly addressed by the BDC Scoping Tool but can be easily assessed separately using the simple conceptual model provided in Section 3.

On this basis, a very conservative upper bound of potential doses is calculated. This also holds for cases in which no groundwater abstraction is assumed based on site conditions, but instead, the release of the contaminants into a local river and exposure from this river water is dominating because there will be a substantial additional dilution taking place between groundwater and river water.

- Tier 4: Use of screening model

  The BDC Scoping Tool [14] allows estimation of failure times for the near field barriers. In Tier 4, the failure time of the last barrier from the BDC Scoping Tool is used as input to a groundwater model which, unlike the model from the BDC Scoping Tool, takes account of contaminant retardation by sorption. Sorption coefficients (K_d values) as well as hydrogeological parameters are provided as default values but can also be modified if site-specific data are available.

  In addition to modelling exposure to water from a groundwater abstraction borehole, the screening model also allows the evaluation of a variant calculation which considers the possibility that radionuclides are released into a river. Potential exposures are estimated for use of the river water as drinking water or for irrigation purposes. In addition, potential exposures from the consumption of fish which take up radionuclides from the river water can be calculated.

  In order to provide a screening of all possible exposures, it is also necessary to provide simple models for the other possible exposure mechanisms, i.e. the transport of radionuclides in the gas phase and the erosion of the soil above the disposal borehole and the subsequent solid release. However, the conceptual models described in Section 3 for these exposure mechanisms are relatively simple and the site-specific data required are very limited. Therefore, there is no sense in trying to simplify these models further for the Tier 4 screening assessment. Instead, models as described in Section 3 can be applied.

  Dose estimates derived on the basis of these screening assessments are still conservative but, in contrast to the results from the simpler models in Tier 1 to Tier 3, will not be overly conservative because of deliberately neglecting important aspects or processes, such as sorption.

- Tier 5: Application of GSA models

  Tier 5 involves detailed modelling according to the approach described in the GSA [13] with full application of the conceptual models described in Section 3. Modelling on this basis will address all events and processes covered by the conceptual models in Section 3 and will require the data described in the GSA. In particular, the Tier 5 model has a more detailed representation of the near field. In the case of disposal of Category 1 and Category 2 DSRS, it will be necessary to also take into account the effects of heating and radiolysis as discussed in Ref. [15].

From the description above, it is apparent that the Tier 1 and Tier 2 assessments only identify which radionuclides are potentially relevant. This information may serve as a basis for the Tier 3 to Tier 5 assessments.

Tier 3 addresses the groundwater gaseous pathways in a scoping manner. Solid releases can also be estimated. The results from these assessments are designed to be highly conservative and will be mainly used to get an indication of the possible exposures and to identify which site-specific data may be relevant.

Tier 4 comprizes complete screening models of the potential exposure which could arize from the borehole disposal facility. These models make limited use of site-specific data but provide, based on conservative assumptions, a more realistic upper bound estimate of potential exposure. If these estimates in Tier 4 are well below the applicable regulatory compliance criteria (e.g. by an order of magnitude or more below the dose constraints), the safety of the facility has been demonstrated already at this assessment step. A continuation of Tier 5 calculations, including the use of further site-specific data and the use of the more complex GSA models, may not be necessary in this case.

The conceptual models for Tier 4 and Tier 5 consider several Defect Scenarios, as defined in the GSA in addition to the Design Scenario. These Defect Scenarios address situations in which more pessimistic assumptions as compared to the Design Scenario are made for the conditions determining the release of radionuclides from the near field and their migration into the geosphere. For the simple comparisons and the very conservative modelling in Tier 1 to Tier 3, these scenarios do not play any role. However, the screening in Tier 4 and Tier 5 does have to consider these scenarios.

The relationship between the five assessment tiers is illustrated in Figure 21.

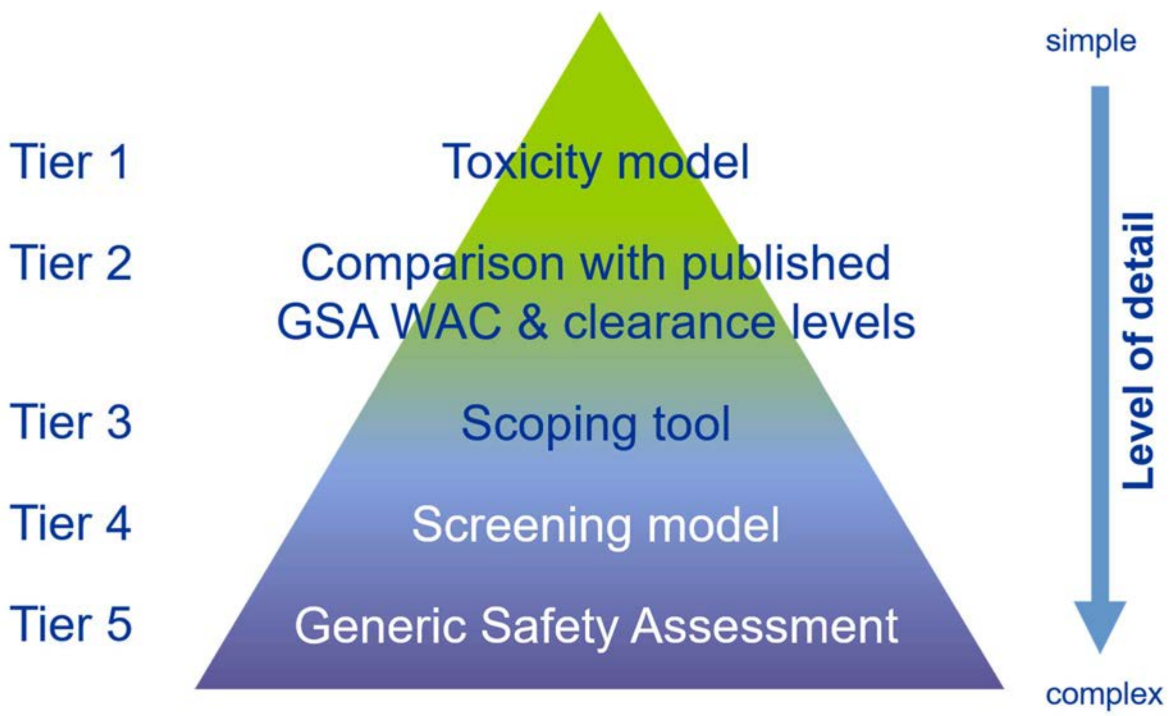

Figure 21. Illustration of the graded approach comprising five assessment tiers.

4.3. DESCRIPTION OF ASSESSMENT TIERS

This section describes the conceptual models used for the assessments in Tier 1 to Tier 5. The conceptual models described in Section 3 corresponding to Tier 5 are used as a basis for the discussion of simplifications made in Tier 1 to Tier 4, as applicable.

4.3.1. Tier 1

Tier 1 consists of two separate assessment steps which are described in Appendix I of the GSA [13]. Both steps intend to identify those radionuclides in the inventory which do not represent a relevant hazard and can be screened out from the further analysis, i.e. do not require further consideration in safety assessment:

The first screening step involves an assessment of radiotoxicity of the inventory in question. This is performed on the basis of the unrealistic assumption that somebody is exposed to the whole radionuclide inventory by ingestion or inhalation. Calculation of the hypothetically resulting ingestion and inhalation doses can be done by multiplying the inventory for each radionuclide by the dose coefficients.

This calculation of hypothetical doses can be performed under the assumption that ingestion or inhalation occurs immediately after the closure of the disposal facility or after a certain period of time. Appendix I of the GSA chooses a period of 30 years (the assumed period of institutional control) after the waste has been disposed of. In this case, the assessment is based on a combination of radiotoxicity and half-life and will provide a less restrictive screening of the radionuclides in the inventory. This will, however, requires a justification for the choice of the time period between facility closure and the hypothetical exposure.

If, under these hypothetical assumptions, the exposure arising from a radionuclide in the inventory is below the applicable dose constraint, this radionuclide is of no concern and does not have to be considered in the following assessment steps.

In the second screening step, a similar highly conservative approach is to evaluate the hypothetical exposures arising from a borehole disposal facility assuming that all the engineered barriers fail, and that the entire radionuclide inventory is released into the groundwater. The radionuclides then migrate directly into a water abstraction borehole, without any dilution or retardation. Details of a simple model developed to carry out this assessment are presented in Appendix I of the GSA.

As in the first step, the option exists to assume that the exposure takes place immediately after closure of the facility or to assume a failure of the engineered barriers only at a later point in time. Appendix I of the GSA again uses a period of 30 years based on the assumption that an institutional control period of this duration will be in effect after the closure of the facility. If the justification of such a time period between the closure of the facility and the hypothetical exposure is difficult, immediate exposure may be assumed at this screening step.

As in the first step, radionuclides in the inventory can be screened out from the further assessment steps if the hypothetical doses are below the applicable dose constraint.

Based on the inventory of radionuclides in a specific case, the Tier 1 assessments may be more or less useful. The use of the second screening step may not add much value to the overall assessment if the subsequent assessments at Tier 2 and Tier 3 will also be carried out. If, however, for some first scoping study only very limited site-specific data are available, applying both Tier 1 steps above may be useful.

It is obvious, that these first screening steps only serve the purpose of narrowing down the further assessments to radionuclides of potential concern. The results of the dose calculation at Tier 1, which can be extremely high, do not provide any assessment of the safety of the facility because the results are derived under hypothetical exposure conditions which cannot realistically take place.

4.3.2. Tier 2

A first indication of the significance of the inventory to be disposed of can be obtained by comparison of the inventory with the WAC in the GSA. The WAC, provided in Section 6 of the GSA [13], can be used as an indication as to whether the inventory to be disposed of will be suitable for a borehole disposal facility given certain site-specific conditions.

Radionuclides occurring in concentrations reaching the WAC or even exceeding these values will require particular attention in subsequent assessments, while radionuclides with much lower activity concentrations than the WAC will likely not lead to any significant exposures. Even for the latter radionuclides, this comparison will not substitute for a site-specific assessment, but the comparison can be used to direct assessment efforts to the potentially more important radionuclides and may contribute to building confidence in the results of the site-specific modelling.

A second assessment step in this tier uses the clearance levels provided in GSR Part 3 [31] as benchmarks for the radionuclide concentration in the disposal borehole as a function of time. The radionuclide concentration in the borehole is calculated by dividing the inventory of radionuclides by the total mass of the near field components in the disposal zone. For simplicity, it is assumed that the whole disposal zone is filled with concrete and the mass of this concrete cylinder (length = length of disposal zone; diameter = diameter of borehole casing) is used as the basis of the volume estimate.

The average radionuclide concentration in this concrete cylinder as a function of time is divided by the clearance level for each radionuclide. The sum of these quotients provides an indication of the hazard presented by the inventory in the disposal borehole as a function of time.

The comparison also allows identification of the radionuclides which dominate the hazard at different points in time. Radionuclides which have average concentrations in the disposal borehole below the clearance levels at the time of disposal or after a short decay period will most likely not represent substantial hazards and, therefore, will not require specific focus in subsequent safety assessments.

As discussed already for the Tier 1 assessment, the Tier 2 assessments may be more or less useful for a specific disposal programme, depending on the intended purpose of the assessment and the availability of data.

Note that the simple assessments at Tier 2 cannot substitute for more detailed modelling of possible exposures because, in reality, radionuclides are not distributed homogeneously throughout the disposal zone. However, meeting clearance levels in this averaged sense for certain (or even all) radionuclides at some point in time will provide an indication that the borehole inventory is no longer hazardous beyond this time. This type of indication can be used in the multiple lines of reasoning included in the safety case as a supporting argument for the results of more detailed assessment modelling undertaken as part of the following tiers.

4.3.3. Tier 3

A description of the conceptual models implemented in the BDC Scoping Tool is provided in the User Guide [14] which explains that the BDC Scoping Tool performs two assessment steps:

- The degradation of the near field components (backfill, disposal container, containment barrier and capsule) is modelled to estimate failure times for each of these components. Failure of the capsule is then used to define the point in time at which contaminants are assumed to be released into the geosphere. Depending on disposal conditions, this release either takes place in the unsaturated zone or in saturated disposal conditions. Conservatively, it is assumed that all radionuclides are released from the near field to the geosphere in one year.

- For the geosphere and biosphere, simplified assumptions are used as compared to the GSA models described in Section 3. The groundwater and gas pathways are taken into account. The results of this assessment are exposure estimates which are, in general, very conservative.

Modelling of the degradation of the near field components is based on some key input data concerning the hydraulic and geochemical groundwater conditions. The modelling uses the same approach as outlined in the GSA, i.e. there is no difference between modelling approaches used in Tier 3 to Tier 5 for estimating near field barrier degradation. An overview of the conceptual models applied and of the required input data is provided in Section 3.

The exposure models for the groundwater and gas pathways are more simplified and are described fully in Ref. [14]. Their key elements are summarized below.

The geosphere and biosphere models provide a very simplified representation of the mechanisms by which the radionuclides released from the near field after the failure of the capsule lead to exposures.

The geosphere model distinguishes between disposal in the unsaturated and the saturated zone, as shown in Figures 15 and 16. After the radionuclides have been released from the near field, it is assumed that these radionuclides migrate within the saturated zone towards a water abstraction borehole. It is assumed that the water abstraction borehole is located at a nominal distance of 100 m downstream of the disposal borehole. The saturated zone is represented by some key properties (hydraulic conductivity, hydraulic head, water-filled porosity). No retardation of the radionuclides by sorption is taken into account.

The only mechanism which leads to a dilution of the contaminant plume in the saturated zone is dispersion. The way in which this effect is treated depends on the disposal conditions:

- For disposal in unsaturated conditions, it is assumed that the radionuclides enter the saturated zone directly below the disposal borehole. This essentially corresponds to a point source. Dispersion then leads to a horizontal and vertical spreading of the contamination, which defines the cross-section of the resulting plume as a function of distance from the disposal zone.

- If disposal conditions are saturated, the whole length of the disposal zone is used as the vertical length of the plume and the widening of the plume is only considered in the horizontal direction. The assumption for the length of the disposal zone is one meter per waste package so that for the default number of 50 waste packages a length of the disposal zone of 50 m results.

Using the default dispersion parameter values in the BDC Scoping Tool, the above assumptions about the source geometry lead to a plume cross-section at the abstraction borehole of 1 m^2 for unsaturated disposal conditions and of 50 m^2 for saturated disposal conditions and 50 waste packages. This difference needs to be taken into account when analyzing the assessment results.

The biosphere model considers four hypothetical adults who obtain all their drinking water from the abstraction borehole, which is used to abstract water directly from the plume of radionuclides in the saturated zone downstream of the facility.

The actual radionuclide concentration in the drinking water depends in this model on the flow rate in the saturated zone — if the contaminant plume supplies enough or more water to the borehole within one year to cover the drinking water consumption of the four adults, the concentration of radionuclides in the plume is used directly as basis for the dose calculation. If, for low flow rates in the saturated zone, the amount of water supplied by the abstraction borehole from the plume is smaller than the drinking water consumption, then it is assumed that the whole inventory in the plume is ingested by the four adults.

The assessment results are obviously very conservative since an immediate release of all contaminants after the failure of the capsule is assumed and all the released radionuclides, or at least a large portion of them, are assumed to be ingested. Furthermore, no sorption of radionuclides in the near field or geosphere is considered.

Solid releases could only take place if the soil and rock layers above the disposal zone erode completely and the waste were to become exposed. This possibility would be excluded by appropriate selection of the disposal depth at a site, and an assessment may not be necessary. If, however, for specific conditions the possible effect of erosion needs to be considered, this would have to be made as part of Tier 4 or Tier 5 modelling since there is no consideration of solid releases in the BDC Scoping Tool. Nevertheless, the assessment of whether a complete erosion of the layers above the waste disposal zone is possible at a specific site can be made within Tier 3 based on knowledge of site conditions (e.g. local erosion rate, site topography and lithology). If this assessment results in excluding the possibility that waste packages could become exposed by erosion, the solid release scenario could be either disregarded completely in the detailed Tier 4 and Tier 5 assessments or at least be treated as a ‘what if’ scenario.

4.3.4. Tier 4

For application in Tier 4, a new screening model has been developed that addresses the groundwater and gas pathways. The basic approach in developing the Tier 4 screening model was to include all processes that are included in the GSA model at Tier 5 but in a further simplified manner. Simplifications were made to reduce the requirements for site-specific input data, and to make the models as understandable as possible. The Tier 4 modelling provides conservative estimates of the potential exposures arising from the facility.

Figures 22 and 23 provide conceptual representations of the Tier 4 model for disposal in the saturated and unsaturated zone, respectively. The approach adopted for the groundwater pathway in this model is similar to that used in Tier 3, but there are some differences:

- The failure of the capsule and disposal container, and the start of radionuclide release is identical to Tier 3, i.e. it is based on the BDC Scoping Tool. The release of contaminants from the waste package is modelled by assuming a period over which the total inventory is released from the waste container into the surrounding backfill and from there into the geosphere. The transport of radionuclides in the backfill by advection and diffusion, as well as their retardation by sorption are considered.

- The models for the geosphere are kept simple and conservative but consider the retardation of radionuclide migration by sorption.

- The biosphere model is a simplified version of the Tier 5 model in the GSA and considers exposure by ingestion of water abstracted from a borehole, irrigated crops, and animal products.

In addition to the assumption that the radionuclides may lead to exposures through drinking of water abstracted from a borehole, a variant case can also be considered in which radionuclides are released into a river. In this case, an additional exposure pathway, the ingestion of fish, is considered.

Gas releases are covered in an identical manner as in the BDC Scoping Tool and the GSA. It is assumed that gaseous radionuclides released from the waste containers enter the interior of a building above the borehole and lead to exposures of the inhabitants of this building.

Since site-specific data (e.g. for hydrogeological parameters, K_d values) can be entered for the Tier 4 screening model, it is possible to derive conservative estimates of exposures from a borehole disposal facility at a specific site and with a given inventory.

This means that if conservative estimates obtained with the Tier 4 screening model result in doses that are substantially (e.g. by an order of magnitude or more) below the applicable regulatory compliance criteria (e.g. dose constraints), it might not be necessary to undertake detailed site-specific modelling at Tier 5. In this case, the screening tool can be considered as a site-specific safety assessment tool with a sufficient level of detail and depth to make sound and defendable judgments about the safety of the planned disposal facility.

The conceptual models used in the different elements of the Tier 4 model are described in more detail in the following sections. These descriptions are related to the groundwater and gas pathways. The application of the Tier 4 model for different scenarios is also discussed.

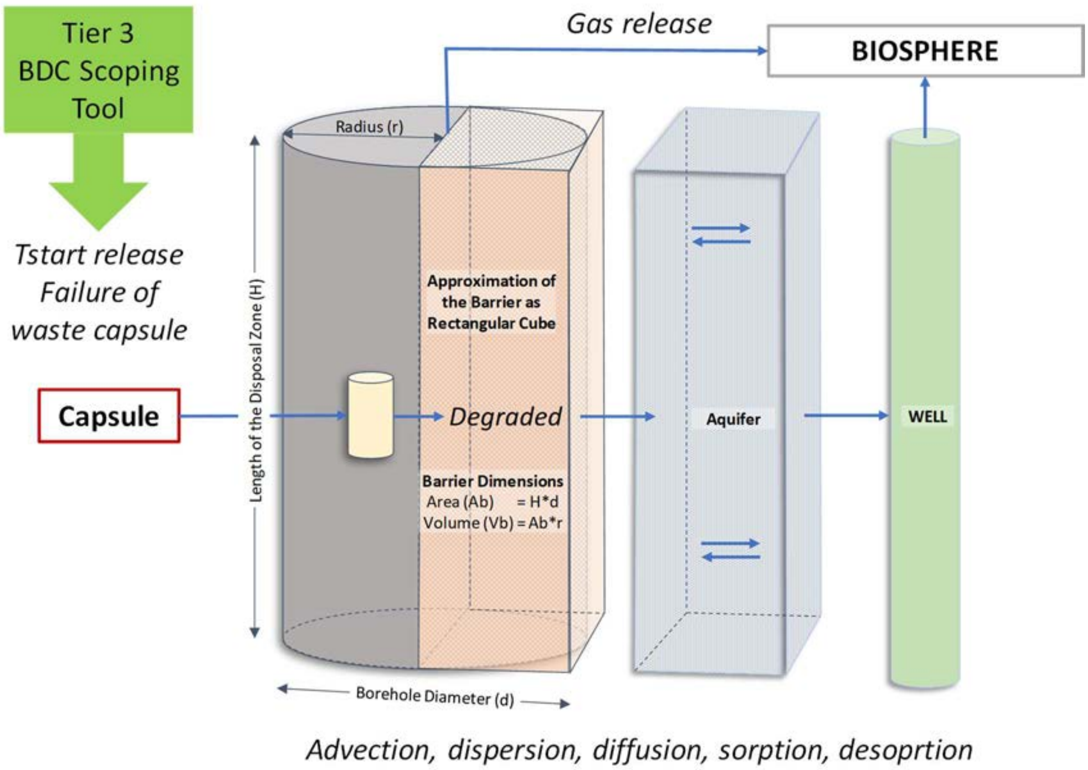

Figure 22. Schematic representation of the implementation of the Tier 4 model for disposal in the saturated zone.

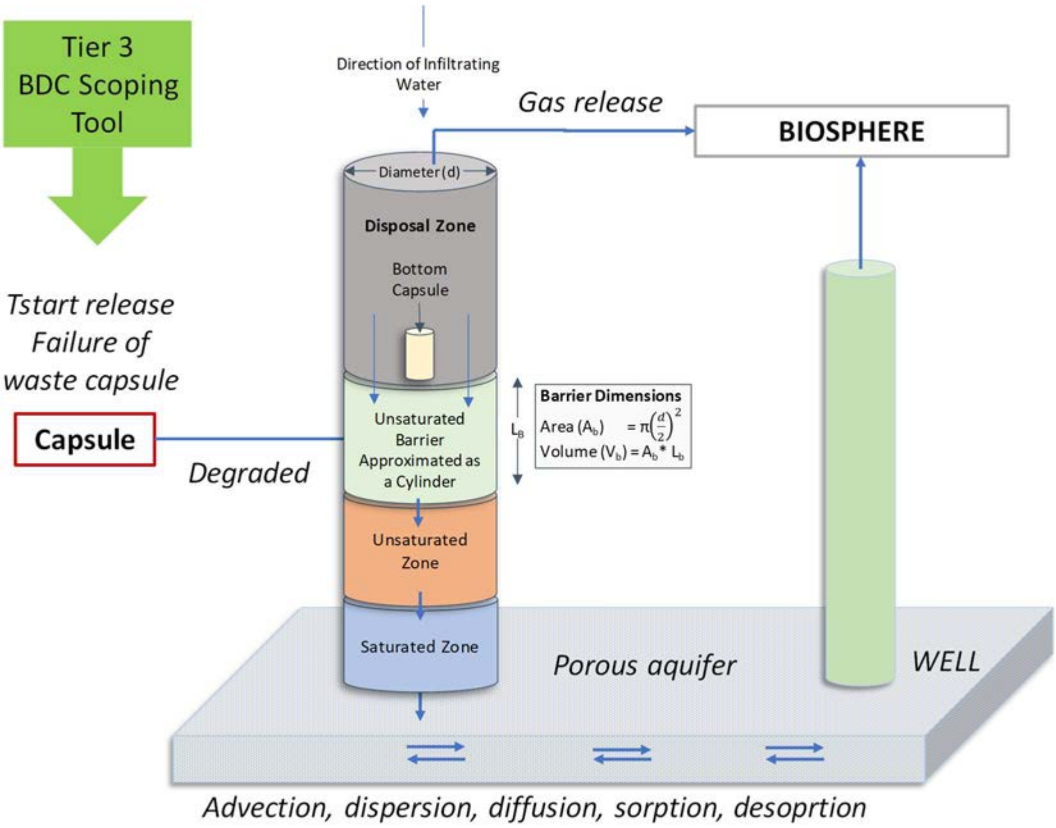

Figure 23. Schematic representation of the implementation of the Tier 4 model for disposal in the unsaturated zone.

4.3.4.1. Near field source term

The start of the release is calculated by applying the BDC Scoping Tool, in the same way as in Tier 3, by specifying some key hydrogeological and geochemical parameters. Two important differences exist, however, regarding the assumed releases mechanisms:

- The release of the radionuclide inventory from the waste container can be assumed to take place instantaneously as in the BDC Scoping Tool, but it is also possible to define a period over which the last barrier (usually the capsule) fails and gradually releases its radionuclide content.

- Instead of assuming an instantaneous release of the radionuclide inventory from the near field barriers to the geosphere as in the BDC Scoping Tool, the radionuclides released from the capsules are transported and retarded in the backfill (assumed to be degraded concrete) before they are released into the geosphere. The migration of radionuclides through the degraded backfill compartments is modelled by representing advective and diffusive transport, taking account of the sorption capacity of the degraded concrete.

4.3.4.2. Release period

The period in which the radionuclides are released from the capsule into the first backfill compartment is one of the key parameters because there is a linear dependence between the calculated exposures and activities released annually.

An instantaneous release, as used in the BDC Scoping Tool, is unlikely but cannot be completely ruled out. This could occur in cases in which the radionuclides are already mobilized through some ingress of water into the capsule such that the activity would then be released within a short time period after failing of the last barrier. A gradual release of the activity during the degradation of the barriers appears more likely — possible timescales are decades, centuries or even longer. To remain conservative, however, the release period ought not to be set too long.

A reasonable assumption could be a period of 70 years between the start of the release and the eventual availability of the whole inventory for transport in the first degraded backfill compartment because 70 years would correspond approximately to the potential exposure duration for a person, living at or close to the disposal site, who was exposed to radioactivity in water abstracted from a borehole for most of their lifetime. In such a case, the important quantity determining the lifetime risk to the individual would be the cumulative dose received by the person over their lifetime. The assumed release period of 70 years would still lead to conservative estimates of lifetime risk because, even if there were no significant attenuation in the geosphere or radionuclide decay, a more rapid release of the inventory would not increase the estimated lifetime dose — the same total dose would just be incurred over a shorter time interval[4]. Radionuclide decay or attenuation in the geosphere would reduce the dose received.

[4] Acute exposures and deterministic effects are not credible for radionuclide release and transport via the groundwater pathway.

4.3.4.3. Transport through degraded near field compartments

The near field backfill will gradually degrade, but even when degraded, it will still restrict the hydraulic flows to some extent and provide some sorption capacity for the radionuclides. This will retard the release of radionuclides from the near field into the geosphere.

The near field in the Tier 4 model consists of several compartments. In each compartment, the activity concentration is assumed to be homogeneously distributed and the decay and ingrowth of radionuclides are considered. Transport of radionuclides between the compartments occurs by advection and diffusion. The modelling of these processes depends on the disposal conditions:

- For disposal in the unsaturated zone, migration of radionuclides down into an underlying saturated zone is modelled on the basis of assumed hydraulic properties of the zone (i.e. of the degraded concrete) and on the net infiltration rate of water. This contaminant migration is assumed to be retarded by sorption.

- If the waste has been disposed of in the saturated zone, the driving force for contaminant migration is the flow of the groundwater. Depending on the hydraulic head difference between the beginning and the end of the near field in the direction of the groundwater flow, the flux of radionuclides leaving the degraded backfill will depend on the hydraulic properties and sorption coefficients in the near field. The total length of these compartments in the direction of the flow is taken as the thickness of the backfill and disposal container surrounding the capsules.

4.3.4.4. Geosphere

The conceptual model for radionuclide migration into the geosphere is similar to the model used in GSA for transport in porous rock. It is assumed that the radionuclides migrate in a plume which originates from a point source that contains the entire disposed inventory. This source will either be at the bottom of the disposal zone for disposal in the unsaturated zone, or at any location within the disposal zone for disposal in the saturated zone, since in this case, the position of the source does not affect the transport resistance.

A widening of the plume by horizontal dispersion is considered. With the default parameters, the width of the plume will be 1 m at the location of the assumed water abstraction borehole, as assumed in the BDC Scoping Tool. This assumption is very conservative. But this conservatism will not play a major role because the water abstraction borehole assumed in the screening model has a substantially higher capacity so that it will draw water laterally from some distance. As long as this distance is larger than the lateral dimension of the plume, the assumptions for this parameter are not relevant because the borehole would abstract all released contaminants, regardless of the horizontal dispersion of the plume.

For the vertical thickness of the plume, the same assumptions as in the BDC Scoping Tool are used. This means that for disposal in the unsaturated zone, the height of the contaminated plume at a water abstraction borehole 100 m from the disposal borehole will be 1 m using the default dispersivity. For disposal in the saturated zone, the height of the plume will correspond to the height of the disposal zone.

In contrast to the BDC Scoping Tool, the screening model considers the retardation of contaminants by sorption in the geosphere, giving more time for radioactive decay and ingrowth. Retardation is modelled using a K_d concept and affects the radionuclide concentration in the groundwater.

The dilution of the water in the borehole with uncontaminated water not affected by the disposal facility is modelled in analogy to the BDC Scoping Tool. If the amount of water abstracted from the borehole is lower than the yield of contaminated water from the plume, no dilution with uncontaminated water is assumed. If the abstracted volume of water is higher than the amount which can be abstracted from the plume, a corresponding dilution with clean water is considered.

A case where radionuclides in the contaminant plume are discharged into a river can also be considered. The flux of the radionuclides in the groundwater and the flow rate of the river will determine the radionuclide concentration in the river water.

4.3.4.5. Biosphere

The conceptual model for the biosphere is a simplified version of the GSA model. The borehole water is assumed to be used for drinking, for the irrigation of agricultural areas and the watering of cattle as is done in the GSA.

The capacity of the water abstraction borehole is an input parameter and is chosen as the value that would be sufficient to provide drinking water to a family and to allow the uses of water for irrigation and watering of cattle.

If a river is considered, the assessments proceed in an identical way, using the concentration of radionuclides in the river water as a basis for estimating exposure. In addition to assessing doses from using the river water for drinking and irrigation, the consumption of contaminated fish is also considered.

The biosphere model used for the groundwater pathway is based on Safety Reports Series No. 19 [32] as updated by Ref. [33].

4.3.4.6. Gaseous release pathway

The conceptual model for gaseous pathway is the same as in the GSA. The need for consideration of the gas pathway will depend on whether gaseous releases would be credible given the inventory in question.

4.3.4.7. Alternative scenarios

The GSA defines and assesses several alternative scenarios:

- Defect Scenarios assume that not all components of the near field perform as envisaged in the Design Scenario due to either defective manufacturing of waste packages (e.g. welding defects), or defective implementation in the borehole (e.g. improper emplacement of cement grout). This results in the earlier release of radionuclides from the near field.

- Further alternative scenarios address unexpected geological characteristics, changes of environmental conditions, and a disturbance of the borehole caused by the drilling of a water abstraction borehole immediately adjacent to the disposal borehole. These alternative scenarios are either screened out in the GSA because of a very low likelihood of occurrence, or modelled implicitly because the corresponding assumptions that would be made to model them are already covered within the sensitivity studies and ranges of variation considered in the Design Scenario. For a specific site, however, the safety assessment may need to reconsider such alternative scenarios in order to investigate the impact of variations of site parameters and/or of possible human activities.

The main difference between the Defect Scenarios and the Design Scenario lies in the assumption of earlier failure of the engineered barriers within the defect scenarios and, therefore, an earlier release of radionuclides. This can be modelled in the Tier 4 screening model by changing the assumptions on the failure time of the engineered barriers:

- Defect Scenarios D1 and D2 assume that either one waste container or one capsule is already defective (e.g. because of imperfect welding) at the time of waste emplacement. This would lead to an earlier release of radionuclides into the near field from the affected waste container and can be treated analogously to the GSA.

- Defect Scenario D3 assumes that the cement grout in the disposal zone or the disturbed zone surrounding it is degraded or incomplete. In terms of modelling, the effects are identical to the Scenarios D1 and D2 in that the capsule and container would fail at an earlier point in time. Since the backfill in the Tier 4 model is always considered to be degraded after the failure of the waste packages, no further differences between these scenarios arise.

- Defect Scenario D4 assumes that one waste container and the capsule in this container are defective at the time of waste emplacement. This would result in the possibility of an immediate release of radionuclides into the near field compartments representing the barriers outside the disposal container. In the GSA, this release is treated on the basis of assumptions relating to the part of the inventory which is available for release as a diffusive process. The screening model, however, uses the same assumption as in the Design Scenario by defining a release period. Since the release mechanisms in the case of a defective capsule and a defective container are unclear, the same release period as in the Design Scenario is used as default value.

The other alternative scenarios result in different characteristics for the contaminant migration such as higher hydraulic conductivity. This can also be considered in the screening model.

The actual treatment of these defect scenarios in the screening model proceeds by considering two waste containers. One of them is assumed to be intact, the other is assumed to have a defect according to one of the defect scenarios described above. This configuration allows the simulation of situations in which both defective and intact waste containers are present in a disposal borehole.

Without more detailed knowledge of the actual distribution of the radionuclides in the disposal containers, it is conservatively assumed that the failed container contains the entire inventory of one radionuclide. The screening calculation, therefore, includes as a default assumption that the entire inventory of the radionuclide is present in the failed waste container. This then allows estimation of the highest possible exposure for each radionuclide.

If specific information is available for the planned distribution of radionuclides in the waste containers, less conservative assumptions could be used. If, for example, the planning is to distribute certain sources (e.g. containing Am-241) over ten disposal containers, the assumption for the respective activity inventory in the defective container could be reduced to 10% of the total inventory of this radionuclide.

4.3.5. Tier 5

The conceptual models used in the Tier 5 assessment, i.e. the application of the GSA, have already been summarized in Section 3 and are fully documented in Ref. [13].

4.3.6 Comparison of Tier 4 and Tier 5 conceptual models

The description of the Tier 4 model shows that it represents a simplified version of the site-specific model used in Tier 5. To facilitate decisions about the application of models in Tier 4 and Tier 5, the main differences of the conceptual models used for these two tiers are summarized as follows:

- Release mechanism

  Tier 5 considers different failure times for the near field components capsule, containment barrier, container, and backfill. The successive process of failure of these barriers is simplified in Tier 4 in that a single failure time for the capsule is considered. After the failure of the capsule, Tier 4 assumes that its radionuclide inventory is released over 70 years. There is no further modelling of the release processes as performed in Tier 5. Note that the model used at Tier 5 for the release of certain relatively insoluble radionuclides (e.g. Co-60) from the failed capsule, assumes that their release is congruent rather than instantaneous [13].

- Near field

  After the radionuclides have been released from the capsule, the Tier 4 model only assumes one barrier in the near field which comprizes degraded concrete. The Tier 5 model allows more complex modelling of the release from the near field, taking account of separate compartments for the different near field components.

- Geosphere

  Transport in the geosphere is based only on flow through porous media in Tier 4. The option in Tier 5 to also consider fracture flow including the process of matrix diffusion does not exist in Tier 4. If the groundwater flow regime is dominated by fracture flow, this only could be simulated in Tier 4 by assuming a very high conductivity.

- Biosphere

  The assumptions made for the biosphere at Tier 4 are simplified compared to the assumptions made at Tier 5 in that they do not consider soil erosion, external irradiation from soil and inhalation of dust.

  Note that the description for the Tier 4 conceptual model provided above makes provision for modelling the potential doses resulting from radionuclide discharge into a river as a variant to abstraction and use of waster from a borehole. The biosphere model used for the groundwater pathway at Tier 4 is based on Safety Reports Series No. 19 [32] as updated by Ref. [33]. The Tier 5 model in Ref. [13] does not consider a river pathway since it was argued that the use of waster abstracted from a borehole would be more conservative. However, on a site-specific basis, the GSA model has been revised to include modelling of the potential exposures resulting from radionuclide discharge into a river. Therefore, a river can be included or excluded at Tier 4 and Tier 5 as a variant to the water abstraction borehole.

- Gas pathway

  The gas pathway is treated in an identical way in Tiers 4 and 5.

The comparison shows that Tier 4 models are simpler and, therefore, require less input data and less modelling effort. Within their range of applicability results obtained within Tier 4 can be seen to provide a conservative envelope for the Tier 5 results. The more detailed Tier 5 models provide less conservative results and can also be used to represent additional FEPs, such as the effects of heating and radiolysis on steel corrosion, enabling more detailed assessment of the safety of disposing of Category 1 and Category 2 DSRS – see Refs [15] and [17].

4.4. MATHEMATICAL MODELS

Mathematical models have been developed based on the conceptual models described in Section 4.3 and are discussed in the following sections.

4.4.1. Tier 1

The mathematical model for the screening of radiotoxicity has been implemented in an Excel spreadsheet which is available from the IAEA[5].

Radiation doses are calculated for inhalation, ingestion and external exposure based on the assumption that a person is exposed to the whole inventory after a decay period (default 30 years). For external exposure, a point source at 1 m distance is assumed.

The mathematical model for the second part of Tier 1 which involves the application of a simplified exposure model is identical to the model in Tier 5 with several simplifications made as described in Section 4.3. The model is available from the IAEA.

4.4.2. Tier 2

In Tier 2 the activity inventories are compared against numerical criteria. The applied criteria are the WAC derived in Section 6 of the GSA [13] and the general clearance criteria stipulated in GSR Part 3 [31]. For the comparison with the general clearance criteria, it is necessary to divide the activity inventory by the volume of the disposal zone (see Section 4.3.2). Further mathematical models are not needed unless it is decided to take account of a period for radioactive decay and calculate the Hazard Quotient (HQ) – see Appendix I.

[5] Contact the Radioactive Waste and Spent Fuel Management Unit of the Division of Radiation Transport and Waste Safety.

4.4.3. Tier 3

The mathematical models used in the BDC Scoping Tool are described in the User Guide for the tool [14]. The tool is available from the IAEA.

4.4.4. Tier 4

Details of the mathematical model for the Tier 4 screening model are provided in Appendix III. The tool is also available from the IAEA.

4.4.5. Tier 5

The mathematical models used in the Tier 5 model are described in the GSA [13].

5. USING A HIERARCHY OF MODELS

5.1. GENERAL

The purpose of this section is to describe how the hierarchy of conceptual and mathematical models described in Section 4 may be used within a graded approach for post-closure safety assessment of a BDS for DSRS. The characterization of the disposal system and the application of the five tiers of models is discussed in the following regards:

- The graded approach (see Section 5.2) aims to optimize the use of resources for the safety assessment process, which includes site characterization and modelling. Resources ought only to be invested up to a level which is commensurate with the potential hazards represented by the facility. The application of this approach based on the hierarchy of models is discussed further in Section 5.3.

- The requirements for disposal system characterization as an outcome of following a graded approach to post-closure safety assessment using the different model tiers are discussed in Section 5.3. This includes descriptions and specifications of the components of the disposal system as well as information on the characterization of the site.

- Apart from directing efforts by applying the graded approach, another important role of using the model hierarchy is to increase confidence in safety by providing redundancies in the modelling approaches. The use of several models implemented in different software improves confidence that each model is providing sensible results. It also improves understanding of the factors that control the performance of the disposal system and can, therefore, be used to reduce uncertainties about the system behaviour. In addition, this allows the establishment of multiple lines of reasoning in support of the safety demonstration for the facility. These aspects play an important role in the presentation of safety arguments to build confidence in the assessment results. These aspects are discussed in Section 5.4.

5.2. GRADED APPROACH

The safety assessment for a disposal facility could, in principle, exclusively be based on the application of the most sophisticated model available which, by definition, does address all potentially relevant FEPs. For the BDS, this would be the Tier 5 GSA model possibly supported by appropriate research level models such as those used in Ref. [15]. Taking this approach would provide as complete an understanding as possible of the disposal system to draw conclusions about the safety of the planned facility. Simpler models would only be required to the extent that these might be needed to provide input or boundary conditions for the Tier 5 model (e.g. the BDC Scoping Tool providing estimates of the failure times of the system components used as input to the Tier 5 model).

However, such an approach would disregard important advantages which can be achieved by starting the analysis with the application of simpler early tier models. For example:

- The scoping models of Tier 1 to Tier 3 allow for the rapid and easy identification of relevant radionuclides and/or relevant site parameters and processes to be investigated further in subsequent more detailed modelling steps. In addition to focussing further modelling, the amount and extent of site-specific data collected can be limited only to those that are required for the safety demonstration. Both aspects are direct applications of the graded approach and reduce the necessity to expend resources to a reasonable level.

- If a relatively simple and conservative screening model demonstrates that the potential doses from the facility will be well below regulatory criteria (e.g. by an order of magnitude or more below the dose limits and constraints), then one could argue that there is no need for any further detailed modelling, since the safety of the facility can be demonstrated and considered on the basis of this relatively simple model.

- Moreover, the application of a simpler model within Tier 4 may enhance understanding of the disposal system and may also help to understand and build confidence in the results of the more complex models.

The application of the graded approach when using the models within the different tiers is discussed in the following subsections. Generally, decisions about the application of models within the different tiers can be based on the following considerations:

- The need for waste, site and design data increase from Tier 1 to Tier 5. The application of the models within these tiers is dependent on the availability of data.

- The usefulness of specific models depends on the overall scope of the assessments. For a first scoping study covering only the first two tiers, for example, the very conservative estimate of the hypothetical exposure and the comparison with generic WAC from the GSA may be useful. If, on the other hand, the scope of the assessment reaches up to Tier 3 or later, very simplified scoping models may only be of limited value because more realistic and site-specific models will be applied.

5.2.1. Tier 1

The main aim of the Tier 1 models is to identify radionuclides of potential concern. Radionuclides in the inventory can be screened out from the further assessment steps if worst case calculated doses fall below the applicable dose constraint. Two model options were presented in Section 4.2 and Section 4.3.1. Either one or both can be used for the purpose of the Tier 1 assessment.

Obviously, the results of the dose calculations in Tier 1 can be extremely high; however, they do not provide an assessment of the safety of the borehole disposal facility because the facility is not considered in the calculations and because the calculated results are for entirely hypothetical exposure conditions which cannot realistically take place.

5.2.2. Tier 2

The comparison with WAC and with clearance levels in Tier 2 has basically the same purpose as the Tier 1 models, namely to identify radionuclides which dominate the hazard arising from the disposal facility at different points in time. As in Tier 1, the WAC and clearance levels are two model options. Either one or both can be used for the purpose of the Tier 2 assessment.

As in Tier 1, this simple assessment cannot substitute for a more detailed modelling of possible exposures, but an indication of the level of hazard imposed by different radionuclides is provided. This type of indication can direct further modelling and data acquisition efforts. It also can be used in the multiple lines of reasoning of the safety case as supporting arguments for the results of more detailed exposure modelling undertaken as part of the later tiers.

5.2.3. Tier 3

As discussed in Section 4.3.3, the results of applying the BDC Scoping Tool are very conservative because an instantaneous release of all contaminants after the failure of the last engineered barrier and very unfavourable exposure conditions are assumed. Therefore, the BDC Scoping Tool is not meant to provide realistic estimates of exposures.

Results from the BDC Scoping Tool could, however, be used to obtain a first indication of upper bounds of exposures. These results could be used, for example, in feasibility or siting studies. They can also be useful in the disposal concept design of site characterization studies.

In addition to providing upper bounds for exposures at a specific site, the BDC Scoping Tool provides estimates of the failure times for the near field barriers. These failure times are calculated for different hydrogeological and geochemical environments. The results are used as input to the Tier 4 and Tier 5 models.

There is no consideration of solid releases in the BDC Scoping Tool. Therefore, if this pathway could be relevant at a specific site, the assessment would need to be performed as part of Tier 4 or Tier 5 modelling. However, it may be possible, through appropriate design of the disposal system based on knowledge of site conditions (e.g. local erosion rate, site topography and lithology), to ensure that a further consideration of solid releases is not needed or that this scenario is very unlikely and can be treated as a 'what if' scenario.

5.2.4. Tier 4

The Tier 4 screening model includes in a simplified manner all principle aspects which are taken account of in the GSA model applied in Tier 5. However, despite these simplifications, the Tier 4 modelling still provides conservative estimates of the potential exposures arising from the facility. Since site-specific data can be used, it is possible to derive conservative estimates of potential exposures from a borehole disposal facility at a specific site and with a given inventory.

The description of the Tier 4 conceptual model makes provision for discharge into a river and subsequent exposure as a variant to abstraction and use of borehole water. Depending on site-specific conditions, either one or both of these options may be used.

If these estimates from the Tier 4 model are well below the applicable dose constraints (e.g. by an order of magnitude or more), the screening tool can be considered as a site-specific safety assessment tool of sufficient level of detail and depth to make sound and defendable judgments about the safety of the planned disposal facility. A continuation of the safety assessment work at Tier 5, including the establishment of a more detailed site-specific data basis and the involvement of complex GSA models, may not be necessary in this case.

If the potential exposure estimates obtained using the Tier 4 models approach or exceed the applicable dose constraint, this does not necessarily mean that the facility would not be safe; the only conclusion to be drawn in this case is that more detailed modelling, using more detailed site-specific data and information, needs to be undertaken. This more detailed modelling may then demonstrate the safety of the facility by considering site-specific factors and processes in more detail than in the simpler models of Tier 4. These more detailed considerations may result in a reduction of the calculated potential exposures such that the more detailed modelling at Tier 5 indicates that the facility would comply with the applicable regulatory (e.g. dose) criteria.

5.2.5. Tier 5

The application of the Tier 5 model is described in detail in the GSA [13]. Setting up the model and defining the input parameters can be facilitated on the basis of the results of the earlier tiers. Results from simpler models can also be used to improve understanding of the results of the GSA model. It is important to remember that, even though the Tier 5 models are more detailed, they are still based on various conservative assumptions, in accordance with normal safety assessment practice for radioactive waste disposal.

Extensions have been made to the Tier 5 model described in the GSA [13] so that it can represent additional FEPs, such as the effects of heating and radiolysis on steel corrosion, and be used for more detailed assessment of the safety of disposing of Category 1 and Category 2 DSRS – see Refs [15] and [17].

Also, note that the Tier 5 model is based on the models implemented in the GSA where a river as the geosphere–biosphere interface was not included. However, as with Tier 4, if a river is more appropriate on a site-specific basis, then the Tier 5 model can be extended to include a river and the associated exposure pathways.

5.3. CHARACTERIZATION OF THE DISPOSAL SYSTEM

5.3.1. General

Characterization of the disposal system is an integral part of the safety case development process and serves as a basis for the supporting safety assessment work (which is in turn used to help guide further characterization). The need to apply the graded approach to disposal system characterization is well understood, and the level of characterization that is required will depend on a number of factors including the nature of the waste to be disposed, the specific phase in the development of the safety case and disposal facility, and whether the disposal system information is required to support a generic or site-specific safety assessment. As discussed in Section 1.1.2, generic safety assessments are useful tools in many aspects of a waste disposal programme, including disposal system characterization with the purpose of identifying the key parameters that need detailed characterization in support of a site-specific safety assessment and the extent of site characterization required [8].

Consistent with the purpose and objectives of this publication, this section discusses system characterization for a borehole disposal facility within a graded approach to illustrate the variation in data and parameter values required for the hierarchy of models.

5.3.2. System definition

As for any type of disposal facility, it is important to develop an adequate understanding of the FEPs that may influence the disposal system and its evolution, and to develop sufficient confidence in the system’s safety. This understanding will evolve as more system-specific information is accumulated and as scientific knowledge is gained from the waste and site characterization programmes. Within the safety assessment framework, the *system description* step includes information on:

- The waste (e.g. activities, physical and chemical characteristics) — this information is gained from source and waste characterization work;

- The engineered near field components (e.g. capsules, disposal containers, backfill and borehole liner) — this information is gained from the research, design and development work that supports the engineering design specification;

- The geosphere (e.g. geology, hydrogeology, geochemistry, tectonics and seismicity) — this information is gained from a literature review and site investigation; and

- The biosphere (e.g. surface processes, water bodies, meteorology, climate and human activities) — this information is gained from a literature review and site investigation.

This system-specific information is used to inform the development and justification of scenarios and the associated conceptual and mathematical models (see for example Section 4 and Section 5 of Ref. [13]). Scenario development and justification may require a more detailed and wider understanding of the system than for the associated conceptual and mathematical model development.

5.3.3. Waste characterization

Sealed radioactive sources contain various radionuclides. The choice of the source depends on the application and the device used. Sealed radioactive sources can have very different physical forms and may contain very different quantities of radioactive material. As a result, they exhibit a wide range of physical, chemical and radiological properties. For example, some sealed radioactive sources produce significant amounts of heat.

For safety assessment purposes, it would be ideal to have a detailed description of each source, of the total DSRS inventory and of all the associated physical, chemical and radiological characteristics. However, because DSRS are sometimes abandoned and are often collected for storage from many different locations and are then stored for a considerable period by persons or organisations other than those that used the sources, it is not uncommon for detailed information on the sources to be lost or to be patchy and of variable quality. This could be especially true if the State does not have a national register of radioactive sources, as recommended in IAEA Safety Standards Series No. RS-G-1.9, Categorization of Radioactive Sources [34].

As a minimum, a list of the radionuclides in the DSRS inventory would be required, together with the current activities of the DSRS. In the absence of such system-specific information, generic information such as provided in IAEA Nuclear Security Series No. 5, Identification of Radioactive Sources and Devices [35], can be used. In addition, the following information would help in developing the waste inventory for a site-specific disposal safety assessment:

- The radionuclide(s) and their current activity (or activity at a known date) for each individual DSRS. This information assists in allocating each DSRS to a particular capsule and waste container;

- The physical characteristics of the DSRS (e.g., shape, dimensions and physical form). This information also helps in the allocation of the DSRS to capsules;

- The chemical form of the DSRS. This information helps in understanding the likely rate of the dissolution of the radioactive material when water infiltrates the failed capsules after disposal; and

- Additional information about each DSRS, such as the device manufacturer, the manufacturer's device model identification number, the device model name, date of manufacturing, the DSRS Serial Number issued by the manufacturer or recycler, and the DSRS model number which could be used to obtain any missing information relating to the DSRS.

The IAEA Source Inventory Management for Borehole Disposal (SIMBOD) software [36] is a useful inventory tracking database tool that is available to support Member States managing DSRS information. It is intended to support predisposal management of radioactive waste (e.g. conditioning), safety assessment and the borehole design for disposal of DSRS, by tracking the DSRS inventory through the various necessary stages of analysis and calculating parameters that will be of value in decision making. SIMBOD is also intended to play an important part of the audit trail for Quality Management purposes when submitting waste inventory information to the regulator. The output from SIMBOD for a country-specific inventory includes the following:

- The number of boreholes that are required for the disposal of the DSRS, as well as the length of the disposal zone in each disposal borehole. These data collectively define the total length of the disposal zone required for the country-specific inventory;

- The total number of waste packages that is required and a database of the suggested content of each waste package (i.e., capsule size and list of the DSRS in each capsule, including the radionuclides and their activities); and

- A summary of the total inventory that includes the radionuclides and total activity in each waste package, the total inventory in each disposal borehole, and the total inventory in all the boreholes (in the case that more than one borehole is used for disposal of the inventory).

5.3.4. Site characterization

The overall aim of site characterization is to gain sufficient understanding of the disposal system, including the characteristics of the biosphere and the geosphere, in terms of regional setting, past evolution and likely future evolution over the timescales of concern. This will include investigating the site characteristics, especially with respect to geological and geomorphological stability, geochemistry and hydrogeology.

The post-closure safety of a borehole disposal facility relies on isolation of the wastes from humans and the containment of radionuclides in the disposal system. Isolation is provided by ensuring waste emplacement at sufficient depth, as well as the narrow diameter of the disposal borehole and the measures taken to reduce the probability of human intrusion. Containment is provided by the waste containers and the other engineered barriers, but also by the geosphere, which provides mechanical stability for the borehole and suitable hydrogeochemical conditions (e.g. for retardation of radionuclides by sorption).

Water is usually the most important medium for the transport of radionuclides from a disposal facility after waste disposal, and so the hydrogeology of the site will always need to be appropriately understood. Because certain types of groundwater could promote enhanced corrosion and degradation of the engineered barriers and may also affect radionuclide sorption (particularly in the geosphere), geochemistry is also a potentially critical determinant of site suitability.

Consistent with guidelines for, and best practice in, site characterization, the site characterization programme may include, but not necessarily be limited to, some or all of the following activities which would provide a basis for the disposal system description that underpins the post-closure safety assessment and safety case:

- The geology and geological evolution of the area. This may involve various surface based and underground activities such as geophysical and borehole drilling investigations, the collection of rock samples for examination and characterization. Investigatory drilling may help to establish drill penetration rates, determine the presence of resources, and establish the presence of faults or other geological features that may influence the performance of the borehole disposal facility;

- The geomorphology and geomorphological evolution of the area. This may involve various studies to examine erosive processes and past land movements (landslips, faults, earthquakes, volcanism);

- The meteorological conditions at present and the possible effects of future climate states on landform development and site conditions (see for example Ref. [37]);

- The geochemistry of the disposal system, in particular the oxidation and reduction (redox) potential and the chemical composition of the groundwater and its speciation;

- The hydrological conditions of surface water bodies in the area in relation to the local topography and the behaviour of these surface water bodies in relation to local meteorological conditions (e.g. precipitation), including adverse conditions (e.g., extreme rainfall events and flooding);

- The hydrogeological conditions of the disposal system, including the general groundwater conditions, the depth to the water table (i.e., the saturated–unsaturated zone interface), the piezometric surface, the results from tests to determine hydraulic parameters (e.g., storativity, hydraulic conductivity, hydraulic gradient), and the soil moisture properties for the unsaturated zone; and

- The biosphere, land use and human behavioural conditions (e.g., foodstuffs and consumption rates and sources of drinking water) for past and present conditions.

Note, however, that it is important that the scope of the site characterization programme for a borehole disposal facility is consistent with the graded approach. The BDS has been shown to be a safe disposal solution for suitably small inventories of DSRS under a wide range of site conditions [13]. This is because the disposal system has been designed to provide a high level of isolation and containment. This reduces the level to which the safety of the facility relies on site characteristics, compared to other disposal facility types. As a consequence, the needs for site characterization are less extensive than for a large near surface or geological disposal facility.

This characteristic of the BDS needs to be reflected in the definition of the site characterization programme. The collection of site-specific data can focus on data that are relevant for the assessment models used. Other site-specific data and information may be collected for additional confidence building purposes; these data may not be absolutely necessary for demonstrating the safety of the borehole disposal facility but can nevertheless be useful, for example in helping to develop multiple lines of reasoning in the safety case.

5.3.5. Engineered near field

The reference design for the engineered near field of the BDS is well defined and is described in detail in Ref. [13]. On a site-specific basis, the design is not expected to change significantly, especially in terms of the materials used for the capsules, waste containers and containment barriers. This means that, in principle, very little additional characterization would be required for the near field components of the system. However, if a decision is made to use an alternative design, additional characterization would be required for the near field components in terms of their physical, chemical, radiological and thermal properties, and the potential interactions of the components with their environment.

Although the behaviour of the near field components included in the GSA reference design was evaluated for a range of generic conditions, additional information would be required to evaluate their degradation performance under site-specific conditions. Factors that have the potential to influence the degradation of the near field components include:

- The radiological properties of the DSRS and associated radionuclides. To assess the potential effects, it could be necessary to characterize the influence of thermal, radiolysis and criticality effects on the performance of the near field components;
- The geochemical and material properties of the engineered near field components. To assess the potential effects, it could be necessary to characterize the chemical behaviour and interactions between the near field components and the groundwater (e.g., chemical composition, pH and redox conditions); and
- The hydrogeological properties of the near field and its surroundings. To assess the potential effects, it could be necessary to characterize the hydrological behaviour and interactions between the near field components and the groundwater (e.g., percolation rate for unsaturated conditions and the groundwater flow regime for saturated conditions).

Most of these parameters would be characterized as part of the site characterization programme for use in the system-specific safety assessment.

Although the engineered components of the near field may remain broadly as defined in the GSA reference design, it may still be necessary to adjust the design of certain aspects (such as capsule/waste package dimensions and containment barrier/backfill thickness) to account for system-specific characteristics (such as DSRS dimensions and inventories).

Other aspects are likely to vary on a site-specific basis, such as what material to use as the casing, the spacing between disposal packages in the borehole, the depth of disposal and the associated lengths of the disposal zone and the closure zone. To a large extent, these aspects will be informed by the outcome of the site characterization programme.

5.3.6. Graded approach to system characterization

5.3.6.1. General

Differing levels of system-specific information will be required for the different tiers of models, with the simplest model (Tier 1) requiring the least information and the most complex model (Tier 5) requiring the most.

The amount of data required from the site characterization programme can be reduced to some extent by using generic (e.g. literature) data and by following a reference biosphere approach to represent the surface system [38]. The use of generic data is common practice especially during early iterations of a safety assessment when limited system-specific information is likely to be available because system characterization programmes might not have been fully implemented. The use of a reference biosphere approach is particularly relevant for the Tier 4 and Tier 5 models.

As waste and site characterization programmes progress, further system-specific information is likely to become available and will need to be incorporated into the safety assessment, thereby reducing uncertainties associated with the use of generic information. It is important to ensure that this new information is incorporated appropriately into the safety assessment as it becomes available. A formal data clearance and data freeze process can be used to help ensure this. This involves approving data from the disposal system characterization programme for use in the safety assessment and introducing the data into the next iteration of the safety assessment in a transparent, auditable manner.

SSG-23 [22] recommends that the site characterization programme reflects the conclusions from previous safety assessments on the need for information. Note that the results of safety assessment can be used at any stage to inform the characterization programme for the disposal system by identifying the parameters which have the greatest impact on the calculated doses. Uncertainties associated with these parameters can be investigated through sensitivity analyses and then, if required, reduced through further system characterization or design work. This ‘risk management’ approach fits well into the overall safety assessment framework.

The following sections discuss the disposal system-specific information requirements of the hierarchy of models and relate these to the characterization programme for the disposal system. The purpose is to identify and focus on the key parameters that could influence decisions on the post-closure safety of the BDS. Table 9 summarizes the complete list of system-specific information requirements for the Tier 1 to Tier 5 models discussed below.

5.3.6.2. Tier 1

The disposal system-specific information requirements for performing the Tier 1 scoping calculation described in Section 4.3.1 are limited to the radionuclide inventory. The 31 radionuclides typically found in DSRS are listed in Table 10, together with the maximum activities that can be expected for each radionuclide in a specific country as documented in Ref. [13]. This allows for a direct comparison between the inventory of radionuclides and the maximum expected activities as a reality check.

The second Tier 1 scoping calculation introduced in Section 4.3.1 includes the geosphere and biosphere components of the system, in addition to the inventory information. However, values for the applicable parameters are hardcoded in the model and consequently are not required to be defined by the user.

TABLE 9. SUMMARY OF POTENTIAL SYSTEM-SPECIFIC INFORMATION FOR TIER 1 TO TIER 5 MODELS

Tier	Near Field	Geosphere	Biosphere
1	Radionuclide inventory	-	-
2	Radionuclide inventory Borehole disposal zone: • inner diameter • vertical length	-	-
3	Radionuclide inventory Capsule/disposal container[b]: • outer diameter • vertical length • wall thickness • weld thickness Containment barrier[b]: • vertical length • gap thickness	Hydrogeology: • Percolation rate[a] • Degree of saturation[a] • Total porosity[a] • Hydraulic conductivity • Hydraulic gradient • Water-filled porosity Geochemistry: • pH • Eh • Chloride concentration • Sulphate concentration • Total inorganic carbon concentration	-
4	Radionuclide inventory Diffusion coefficients Sorption coefficients Percolation rate[a] Degree of saturation [a] Total porosity [a] Grain density Hydraulic conductivity Hydraulic gradient Water-filled porosity Failure times for capsule	Diffusion coefficients Sorption coefficients Percolation rate[a] Degree of saturation[a] Total porosity[a] Grain density Hydraulic conductivity Hydraulic gradient Water-filled porosity A fraction of water demand supplied by contaminated water	Concentration factors House dimensions House ventilation rate Soil total porosity Soil degree of saturation Percolation rate Ingestion rates Inhalation rates House occupancy rate Irrigation rates Crop yields
5	Radionuclide inventory Diffusion coefficients Sorption coefficients Percolation rate[a] Degree of saturation[a] Total porosity[a] Grain density Hydraulic conductivity Hydraulic gradient Water-filled porosity Failure/degradation times for near field components	Diffusion coefficients Sorption coefficients Percolation rate[a] Degree of saturation[a] Total porosity[a] Grain density Hydraulic conductivity Hydraulic gradient Water-filled porosity A fraction of water demand supplied by contaminated water	Concentration factors Garden dimensions House dimensions House ventilation rate Soil total porosity Soil degree of saturation Percolation rate Inhalable dust concentration Erosion rate Ingestion rates Inhalation rates Occupancy rates Irrigation rates Crop yields

Notes:

[a] Only required if the disposal zone is in the unsaturated zone.
[b] Expected to be broadly similar for different systems.

TABLE 10. SUMMARY OF RADIONUCLIDES ASSOCIATED WITH DSRS AND THEIR MAXIMUM EXPECTED ACTIVITY [13]

Radionuclide	Maximum Expected Activity (Bq)	Radionuclide	Maximum Expected Activity (Bq)	Radionuclide	Maximum Expected Activity (Bq)
Am-241	1.20E+13	H-3	2.80E+14	Pu-239	6.70E+11
Au-195	4.00E+07	Hg-203	1.50E+06	Ra-226	2.60E+13
Ba-133	3.30E+08	Ir-192	9.30E+14	Se-75	3.00E+11
Cd-109	3.10E+09	Kr-85	6.30E+11	Sm-151	7.80E+09
Cf-252	4.10E+09	Mn-54	1.00E+05	Sr-90	5.30E+11
Co-57	1.70E+10	Na-22	3.40E+06	Tl-204	5.00E+08
Co-60	2.90E+15	Ni-63	2.10E+10	Y-88	1.00E+05
Cs-137	7.50E+14	Pb-210	1.00E+07	Yb-169	2.20E+11
Eu-152	4.00E+08	Pm-147	2.70E+11	Zn-65	3.70E+05
Fe-55	2.70E+00	Po-210	1.00E+10		
Gd-153	1.50E+11	Pu-238	2.00E+10		

5.3.6.3. Tier 2

The disposal system-specific information requirements needed to perform the Tier 2 scoping calculations introduced in Section 4.3.2 are also limited to the radionuclide inventory.

The basis for the first Tier 2 model are the radionuclide-specific WAC that were derived in Ref. [13] for a range of site conditions. The GSA provides activity limits per waste package for 11 radionuclides which were selected for detailed calculations following screening (i.e., H-3, Co-60, Ni-63, Kr-85, Sr-90, Cs-137, Pb-210, Ra-226, Pu-238, Pu-239 and Am-241) and total activity limits for the 31 radionuclides listed in Table 10.

The clearance levels specified in Table I.2 in Schedule I of GSR Part 3 [31] include all the radionuclides listed in Table 10 and provide the basis for the second Tier 2 model. Clearance levels represent a value expressed in terms of activity concentration and/or total activity, at or below which regulatory control may be removed from a source of radiation [6]. To express the total activity on a mass basis, it is assumed that the borehole disposal zone is filled with concrete and that the activity is equally distributed throughout this concrete cylinder. For this purpose, it is necessary to make assumptions about the dimensions of the disposal zone (i.e., its length and diameter).

5.3.6.4. Tier 3

The disposal system-specific information requirements needed to perform the Tier 3 calculations using the BDC Scoping Tool introduced in Section 4.3.3 are greater than for the Tier 1 or Tier 2 models. The information needed relates to the near field and geosphere components of the system.

For the near field component, the radionuclide inventory is required, similar to Tier 1 and Tier 2. The current version of the BDC Scoping Tool makes provision for the following 16 radionuclides: H-3, Co-60, Ni-63, Kr-85, Sr-90, Ba-133, Cs-137, Pm-147, Sm-151, Eu-152, Pb-210, Ra-226, Pu-238, Pu-239, Am-241 and Cf-252. The inventory is divided into the number of capsules that are specified. The BDC Scoping Tool makes provision for this information to be imported from SIMBOD.

In addition, some further information is required about the waste package that will be used for the disposal of the DSRS. This includes the following:

- The material types used for the capsule and the waste container. The current reference design of the BDS uses 316/L stainless steel, but this material is replaced by superduplex stainless steel for Category 1 and Category 2 Co-60 sources, consistent with the recommendations of Ref. [17][6];

- The geometry parameter values for a capsule and waste container, including the outer diameter, the vertical length, the wall thickness, and the weld thickness;

- The geometry parameter values for the containment barrier, including the vertical length and the gap thickness.

In representing the geosphere, a distinction is made between hydrogeological and geochemical conditions. Provision is made for disposal in saturated or unsaturated hydrogeological conditions. The system-specific parameter values that are required include the following:

- Hydrogeological parameters: the vertical percolation rate, degree of saturation and total porosity for disposal in unsaturated conditions, and the hydraulic conductivity, hydraulic gradient and water-filled porosity for saturated disposal conditions;

- Geochemical parameters: the pH and Eh, as well as the chloride, sulphate and inorganic carbon concentrations of the percolating water (unsaturated conditions) or groundwater (saturated conditions).

Note that the current version of the BDC Scoping Tool allows the user to select pre-populated values for the waste package, hydrogeology and geochemistry characteristics that are consistent with the generic values defined in Ref. [13]. These values can be replaced with system-specific values as appropriate.

5.3.6.5. Tier 4

The disposal system-specific information requirements needed to perform the Tier 4 screening calculations introduced in Section 4.3.4 are greater than for the Tier 3 model and relate to the near field, geosphere and biosphere components of the system.

For the near field component, the radionuclide inventory is needed, similar to the Tier 1 to Tier 3 models. However, there is no limitation on the number of radionuclides that can be entered into the Tier 4 model. The Tier 1 to Tier 3 models can be used to reduce the list of radionuclides to only those that potentially have an influence on the safety of the disposal facility.

[6] Note that the use of 304 stainless steel has been dropped from the reference design of the BDS and is no longer recommended.

The Tier 4 model needs some input from the Tier 3 model. The Tier 3 model is used to calculate the degradation of near field components, which requires the parameter values listed in Section 5.3.6.4. The containment period for the near field barriers calculated with the Tier 3 model is an input parameter for the Tier 4 model.

As described in Section 4.3.4, the Tier 4 model represents the degraded near field components of the system (i.e., after the containment period) as a near field barrier consisting of degradation/corrosion products of whatever materials were included in the near field (e.g., capsule, cement containment barrier, waste container, and cement backfill). For this near field barrier, system-specific values are required for the following parameters:

- The vertical percolation rate, degree of saturation, and total porosity for disposal in unsaturated conditions;
- The hydraulic conductivity, hydraulic gradient and water-filled porosity for disposal in saturated conditions;
- The diffusion coefficients, sorption coefficients (K_d values) and grain density of the near field barrier material;
- The dimensions of the near field barrier, including the cross-sectional area of the barrier and the length of the barrier in the direction of flow for saturated or unsaturated conditions.

For the geosphere, the disposal system-specific parameters for the Tier 4 model are similar to those required for the Tier 4 near field barrier. These include:

- The vertical percolation rate, degree of saturation, and total porosity for disposal in unsaturated conditions;
- The hydraulic conductivity, hydraulic gradient and water-filled porosity in the saturated geosphere component;
- The diffusion coefficients, sorption coefficients (K_d values) and grain density of the saturated or unsaturated geosphere component;
- The dimensions of the saturated or unsaturated geosphere component, including the cross-sectional area of the component and the length of the component in the direction of flow for saturated or unsaturated conditions.

The lengths of the geosphere required are the distance from the bottom of the disposal zone to the water table (for disposal in the unsaturated zone), and the horizontal distance from the disposal facility to the abstraction borehole or river (for disposal in the saturated zone).

For the biosphere, the disposal system-specific parameter values that are required depend on the biosphere model and associated detail considered in the Tier 4 model. As a minimum, the water abstraction rate from either the water abstraction borehole or the river is required. A value is also needed for the water ingestion rate; in most cases, a generic value can be used for this parameter. In addition, values for the following parameters may be required for the representation of groundwater pathway:

- Radionuclide specific parameters, such as concentration factors in soil and crops;
- Soil and crop-specific parameters, such as soil total porosity, grain density, degree of saturation, percolation rate of irrigating water through the soil, crop yields and consumption rates;
- Gardening-specific parameters, such as the irrigation rate.

The model included at Tier 4 for the gas pathway is the same as the model included in Tier 5. Some of the soil specific parameters required for the groundwater pathway are also needed for the gas pathway, such as the grain density and the degree of saturation. In addition, parameter values are required for the following:

- House occupancy parameters, such as occupancy rates (indoor and outdoor) and inhalation rates for dust and radon gas;
- House-specific parameters, such as the dimensions of the house (i.e., cross-section and height of the house to calculate the total volume) and the air ventilation rate through the house;
- Contaminant-specific parameters, such as the inhalable dust concentration, the diffusion length of Rn-222 in the closure zone of the disposal facility, and the radon emanation fraction; and
- The thickness of the closure zone.

5.3.6.6. Tier 5

The disposal system-specific information requirements needed to perform the Tier 5 calculations are similar to — but more extensive than — those required for Tier 4. As introduced in Section 4.3.5, the main difference between the two tiers relates to the conceptual representation of different components of the system.

The most significant difference between the Tier 4 and Tier 5 models lies in the near field, with a more facility-specific representation of the near field in Tier 5. Instead of a single near field component that represents the degraded near field components in the Tier 4 model, the following parameters values are required for each near field component in the Tier 5 model:

- The vertical percolation rate, degree of saturation, and total porosity for disposal in unsaturated conditions;
- The hydraulic conductivity, hydraulic gradient and water-filled porosity for disposal in saturated conditions;
- The diffusion coefficients, sorption coefficients (K_d values) and grain density of the near field barrier material;
- The dimensions of the near field barrier that includes the cross-sectional area of the barrier and the length of the barrier in the direction of flow for saturated or unsaturated conditions.

The geosphere and biosphere parameter values required for the Tier 5 model are largely the same as those required for the Tier 4 model (see Table 9), with a few additional parameters required for the biosphere model (e.g., garden dimensions, inhalable dust concentration erosion rates, and garden occupancy rate).

5.4. CONFIDENCE BUILDING AND PRESENTATION OF SAFETY ARGUMENTS

As discussed in Section 2.3, the safety case provides a synthesis of the available evidence, arguments and analyses, explaining how relevant data and information have been considered, how models have been tested, and how a rational and systematic assessment procedure has been followed.

The tiered safety assessment approach outlined in Section 4 provides a systematic basis for developing this synthesis of arguments and results. Model results from different tiers can be related and compared to each other. Discussions can be provided which explain and support model results from the Tier 4 and Tier 5 models based on results of the earlier tiers. Similarly, the screening model results can be used to provide additional insight into the processes and results implemented into the Tier 5 GSA model.

An important aspect of safety case development and assessment concerns the treatment of uncertainties, and this is often largely approached through the development of scenarios.

Scenario development is usually an iterative process. Results from the earlier assessment tiers can be used to help define a set of scenarios relevant to site conditions for application in Tier 4 and Tier 5. These results can also be used to justify why certain scenarios can be excluded from further analysis or treated as low probability ‘what if’ scenarios.

Paragraph 3.34 of SSR-5 [10] states: “Adequate defence in depth has to be ensured by demonstrating that there are multiple safety functions, that the fulfilment of individual safety functions is robust and that the performance of the various physical components of the disposal system and the safety functions they fulfil can be relied upon, as assumed in the safety case and supporting safety assessment.”

Simple models, which do not take account of certain engineered or natural barriers, can be used to demonstrate that safety does not depend unduly on individual safety functions. Such arguments may also be fostered by comparing model results from different tiers which take account of safety functions to different degrees.

Paragraph 3.48 of SSR-5 [10] states: “For geological disposal and for the disposal of intermediate level radioactive waste, the passive safety features (barriers) have to be sufficiently robust so as not to require repair or upgrading. The long-term safety of a disposal facility for radioactive waste is required not to be dependent on active institutional control.”

The individual barriers, as well as the whole disposal system, have to be able to withstand perturbations by internal processes and external impacts as required by safety considerations. Combining and comparing results from different modelling tiers can be useful to identify the barriers or the combinations of barriers which are relevant in this regard and for which robustness needs to be demonstrated.

A further element of the safety case in which results from the different modelling tiers can be used is the management of uncertainties:

- Information from scoping assessments on the relevance of certain design or site parameters, and model assumptions, can be very useful for determining which parameters and model uncertainties need to be addressed further, for example in uncertainty and sensitivity analyses using the Tier 4 and Tier 5 models.

- The requirement for robustness does not only apply to the disposal system; it also applies to the safety assessment. The results of the safety assessment cannot depend unduly on a single parameter or model assumption. While the dependency on parameter values can be tested in a straightforward and systematic way through sensitivity analysis, the impact of model assumptions is often more difficult to evaluate. Here, the use of different conceptual and mathematical models in the different assessment tiers can provide valuable insight into the impact of certain model assumptions. Confidence can be gained from the fact that the different assessment tiers and their associated models discussed in this publication, implemented by different experts using different software (e.g., spreadsheets, the BDC Scoping Tool, Ecolego, AMBER), give sensible and comparable results whose differences can be understood and relate to the different degrees of conservatism in the different assessment tiers and models (see Appendix I).

The approach, outlined above, of systematically assessing disposal system performance and managing uncertainties by applying models at a level of detail and complexity relevant to the question under consideration enhances confidence in the safety case. This approach also helps to provide multiple lines of reasoning for demonstrating the safety of the disposal facility. The approach enhances the safety case by providing a range of different arguments that, together, build confidence in certain data, assumptions and results. For example, the confidence in model results achieved numerically in Tier 4 and Tier 5 may be enhanced by comparison with simpler spreadsheet calculations obtained in the earlier tiers. The correctness, or validity, of such simpler models can often be demonstrated independently from numerical calculation tools, and this can make the arguments for safety more easily communicated, understood and accepted.

In this regard, it is important to note that certain arguments may be more meaningful and convincing than others to specific audiences.

6. CONCLUSIONS

Disposal provides the only permanent solution for managing DSRS containing long-lived isotopes or high activities of short-lived radionuclides such as Cs-137 that cannot be re-used or recycled. Of the possible disposal options, the BDS has some potential advantages over near surface and geological disposal systems: it has the potential to safely accept a wider range of DSRS and be more secure than near surface disposal particularly for high activity sources, it requires a limited land area and infrastructure to implement, and it can be constructed, operated and closed in a short period. Borehole disposal may be particularly suitable for relatively small inventories of DSRS, whereas geological disposal may be more suitable for large DSRS inventories.

The Safety Fundamentals [3] state: “Safety has to be assessed for all facilities and activities, consistent with a graded approach”. This publication addresses the question of how the graded approach can be applied to safety assessment for borehole disposal facilities by using a set of safety assessment models which has been developed and tested at five different levels (tiers) of complexity and conservatism:

- Tier 1 (less complex, extremely conservative): Assessment of toxicity of radionuclide inventory;
- Tier 2: Comparison of activity concentrations with predefined WAC and clearance level values;
- Tier 3: Use of the BDC Scoping Tool;
- Tier 4: Use of a newly developed screening model;
- Tier 5 (more complex, more realistic): Use of the GSA model.

These models can be used to assess the safety of BDS in accordance with the graded approach by working systematically from the simpler, more conservative models (Tier 1) to the more complex, more realistic models (Tier 5).

An important role of the relatively simple assessments in the earlier assessment tiers lies in the fact that they can be used to exclude some radionuclides from further analysis. If the models in Tier 1 and Tier 2 show that the potential radiation exposures from certain radionuclides are insignificant, even under very conservative assumptions, then there is no need to take those radionuclides into consideration in later more detailed modelling.

The use of relatively simple models can also provide important indications regarding the ease of making a convincing safety case for the disposal of a certain waste inventory at a particular site. Such models can also help to identify which properties of the disposal system and which uncertainties are likely to be the most important to safety, and which of these may need to be addressed and quantified in greater detail.

During this study, a new model was developed at Tier 4. This fills a gap between the relatively simple BDC Scoping Tool (Tier 3) and the more complex GSA model at Tier 5. The Tier 4 model avoids some of the overly conservative assumptions of the BDC Scoping Tool and describes the relevant processes (release of radionuclides from the waste forms, migration in the geosphere and exposure mechanisms in the biosphere) using simplified, but not unrealistic assumptions and models. The Tier 4 model includes a complete set of screening models with which to evaluate the potential exposures that could arise from a borehole disposal facility.

By comparison, the Tier 4 model is simpler than the Tier 5 model and, therefore, requires less input data and less modelling effort. Based on slightly more conservative assumptions, the Tier 4 model provides a conservative upper bound for the Tier 5 results. If the exposure estimates made at Tier 4 are well below the applicable regulatory criteria (e.g. by an order of magnitude or more below the dose constraints), then the post-closure safety of the facility has been demonstrated and it may not be necessary to undertake a more detailed post-closure safety assessment at Tier 5. In such cases, and because the Tier 4 model requires less site-specific data, it may also be possible to conduct a less extensive and more focused site characterization programme.

If the post-closure safety assessment work does not stop at Tier 4, but continues with the Tier 5 model, then the use of the different, independent models implemented in different software with differing levels of complexity can help verify and build confidence in the ‘correctness’ of each of the models. This may help ultimately to make the safety case for disposal more convincing to interested parties.

The overall result from the work described in this publication is an approach by which programmes considering the disposal of DSRS in boreholes can systematically assess the post-closure safety of the BDS, starting with simple methods and working as needed to increasingly detailed and more complex safety assessments, in order to develop a well supported safety case in accordance with the IAEA safety standards.

The application of this approach has been demonstrated using an example that considers a hypothetical, but realistic situation. The results illustrate the advantages of following the graded approach and focusing facility design, system characterization and safety assessment effort on those aspects which are most significant to the level of risk.

The work has also considered the potential needs for the collection of data with which to characterize the disposal system. In terms of site characterization, enough work needs to be done so that, at a minimum, there is sufficient understanding of the geography, geology and hydrogeology of the site to provide confidence in: its geological stability; the identification, characterization, location and depth of a suitable host rock for the waste disposals; and the groundwater flow pattern and rates (e.g., location of the water table, groundwater flow direction and hydraulic head gradient). There also needs to be sufficient information on the geochemistry of groundwater at the site (e.g., major elements, pH and redox conditions), and on the potential pathways through which radiological exposures might potentially occur. The amount of site characterization and the level of detail in the characterization programme can be guided by the safety assessment work, consistent with the graded approach.

APPENDIX I

THE APPLICATION OF THE GRADED APPROACH TO POST-CLOSURE SAFETY ASSESSMENT: AN EXAMPLE

GENERAL

The purpose of this appendix is to provide further information on how to follow the graded approach when performing post-closure safety assessment to support a site-specific safety case for implementation of a BDS.

This appendix presents a worked example of a hypothetical country-specific implementation of the BDS, the decision making process followed, and how the hierarchy of models presented in Section 4 can be used to perform the post-closure safety assessment using the graded approach.

EXAMPLE BACKGROUND

A hypothetical Member State uses sealed radioactive sources (SRS) for medical, industrial and research purposes. With time, some of these SRS become disused, which result in a substantial national DSRS inventory. A Radioactive Waste Management Organization (RWMO) established in the country is tasked to find a long-term management solution for their national DSRS inventory that consists of several high activity Category 1 and 2 DSRS, as well as a wide spectrum of Category 3 to 5 DSRS.

Table 11 summarizes the DSRS inventory that the Member State has to manage. It consists of a spectrum of long-lived and short-lived radionuclides.

TABLE 11. DSRS INVENTORY THAT A HYPOTHETICAL MEMBER STATE INTENDS TO DISPOSE OF USING THE BDS

Radionuclide	Inventory (Bq)	Half-Life (y)	Radionuclide	Inventory (Bq)	Half-Life (y)
Am-241	1.70E+12	4.32E+02	Mn-54	1.00E+05	8.56E-01
Au-195	3.00E+07	5.01E-01	Na-22	2.40E+06	2.60E+00
Ba-133	2.30E+08	1.07E+01	Pm-147	1.70E+11	2.62E+00
Cd-109	2.10E+09	1.27E+00	Po-210	1.00E+10	3.79E-01
Co-57	1.00E+10	7.42E-01	Ra-226	3.00E+11	1.60E+03
Co-60	1.11E+14	5.27E+00	Se-75	2.00E+11	3.28E-01
Cs-137	5.10E+11	3.00E+01	Sm-151	6.80E+09	9.00E+01
Eu-152	3.00E+08	1.33E+01	Tl-204	4.00E+08	3.78E+00
Fe-55	9.00E+09	2.70E+00	Y-88	1.00E+05	2.92E-01
Gd-153	1.00E+11	6.63E-01	Yb-169	1.20E+11	8.76E-02
Hg-203	1.00E+06	1.28E-01	Zn-65	2.70E+05	6.68E-01
Ir-192	8.30E+14	2.03E-01			

The Member State has no nuclear programme at present. However, there is a prospect that a limited nuclear programme with the focus on research will be developed in future. It is foreseen that a near surface disposal facility will also be developed in the future for the management of the radioactive waste generated in the process.

The national policy and strategy recognize the need to manage the DSRS inventory as radioactive waste. However, the strategy is not specific on which disposal concept to use for the long-term management of the DSRS inventory. The Member State is aware of the BDS and the advantages of this concept as an option for the long-term management of the DSRS inventory. However, national legislation still requires the consideration of different alternatives. The RWMO is, therefore, required to argue the suitability of the different disposal concepts available for the long-term management of the DSRS inventory as part of their submissions to the national regulator.

The RMWI has access to the hierarchy of models presented in this publication. These models are used below to illustrate their application as part of the post-closure safety assessment and safety case development process followed to support the selection of the BDS as the preferred alternative, using the hypothetical system description as a basis.

DISPOSAL CONCEPT SELECTION

General

The RWMO evaluates the available management options for DSRS, which include: return to supplier, decay storage, and disposal. After establishing that decay storage and return to supplier are not feasible management options for the Member State, the RWMO embarks on a process to identify a suitable disposal concept for the DSRS inventory between the 3 main categories, namely near surface disposal facilities, geological disposal facilities, and borehole disposal facilities.

Factors taken into consideration include the risks posed by the inventory of DSRS and the disposal facility, the costs involved in developing and implementing the disposal concept, the benefits that the concept would provide in terms of long-term safety, the timescales required to implement the different disposal concepts, and the views of interested parties.

Near surface disposal

The development of a near surface disposal facility that could be used for the disposal of some of the DSRS in the Member State is a prospect for the near future. This means that the expected lead time to develop and implement a near surface disposal facility is less than for a geological disposal facility. However, it will still be longer than the time required to develop a borehole disposal facility.

While still not insignificant, the cost to implement a near surface disposal facility is also less than for a geological disposal facility. However, a near surface disposal facility has other challenges.

The disposal of high activity Category 1 and Category 2 DSRS in a near surface disposal facility would pose a safety and security risk (e.g. related to an intrusion event). Even some of the Category 3 to 5 DSRS may pose a risk for some scenarios (e.g. following erosion).

A particular issue would arise from the long-lived alpha emitters (e.g., Ra-226 and Am-241) in the inventory. The activity concentration of these radionuclides is considerably higher than the levels which normally can be disposed of safely in a near surface disposal facility. Therefore, it is likely that the total inventory cannot be managed within this disposal option.

In summary, the disposal of the inventory of DSRS in a near surface disposal facility would pose safety and security challenges and, consequently, is not a suitable concept for the long-term management of the inventory of DSRS in the hypothetical Member State.

Geological disposal

Viewed from a long-term safety and security perspective, a geological disposal facility would provide a very high and sufficient level of security, safety, isolation and containment. However, the high costs for developing a geological disposal facility might not be justified for just a small inventory of DSRS.

As there are no immediate plans by the Member State to develop a geological disposal facility for the disposal of high level waste, it is likely that the DSRS would have to remain in storage for a long and undefined period which is not desirable. This would not be in accordance with IAEA safety standards and there would be ongoing costs and needs for security measures at the store.

In summary, the disposal of the inventory of DSRS in a geological disposal facility would provide the necessary safety and security. However, the associated cost for the development of such a facility, and the uncertainty in timescales when and if a geological disposal facility will be implemented, means that such a facility is not regarded as a suitable concept for the long-term management of the inventory of DSRS in the hypothetical Member State.

Borehole disposal

The advantages of using the BDS as a long-term management solution for DSRS is well known to the Member State. It not only addresses the safety and security concerns due to deeper than near surface disposal, the cost to implement the disposal facility is expected to be significantly less than either near surface or geological disposal. The lead time to develop and implement the borehole disposal facility is relatively short, which means that the facility can be sited, constructed, operated and closed in a relatively short period of time (~1 year after regulatory approval). Borehole disposal has the further advantage that the concept has been subject to extensive international evaluations and reviews for a range of site conditions.

Conclusion

With the discussion above as a basis, the Member State concludes that borehole disposal would be the most logical strategy to adopt as the long-term management option for the inventory of DSRS presented in Table 11.

CHARACTERIZATION OF DSRS

The inventory provided in Table 11 is the starting point for the implementation of the BDS. Detailed characterization of the inventory in terms of its physical, chemical and radiological properties does, therefore, contribute to the effective decision making process. Here the use of SIMBOD (Source Inventory Management for Borehole Disposal) as an inventory tracking database tool is encouraged. The output from SIMBOD summarizes the total inventory that includes the radionuclides and total activity in each waste package, the total inventory in each borehole, and the total inventory in all the boreholes (in the case that more than one borehole is used).

The RWMO uses SIMBOD to manage the inventory and to derive management information for the design of the borehole disposal facility. It is determined that one disposal borehole is sufficient for the disposal of the DSRS inventory. Each radionuclide in the inventory is allocated to a single capsule, resulting in a total of 24 capsules and associated disposal containers. Note that, except under very specific conditions, only one capsule is allocated to a disposal container. However, SIMBOD is not prescriptive in terms of the number of DSRS that can be allocated to a capsule. Instead, the following criteria are used to decide the number of DSRS that are allocated to a capsule:

- The physical dimensions of the source(s) relative to the volume available in the capsule;
- The total activity of the DSRS allocated to the capsule are below the WAC per waste package bounds; and
- The total thermal output generated by the DSRS allocated to the capsule.

This means that the RWMO may have allocated several DSRS containing the same radionuclides to a specific capsule, to the maximum activity listed in Table 11.

SITE SELECTION

Depending on national legislation, the Member State can follow different approaches to identify a suitable site for the implementation of the BDS, i.e., finding a site that meets the requirements for operational as well as long-term safety and security of the DSRS. Given that the BDS was designed to tolerate a wide range of geological and climatic conditions, the expectation would be that many sites will meet the safety requirement. Site suitability is then likely to depend on issues such as transport distances, accessibility, local acceptability, and proximity to important cultural and religious sites. Land ownership is often an important factor with preference being given to government-owned land.

As far as technical issues are concerned, the site needs to have a stable geology and be reasonably accessible. In addition, the site cannot be subject to rapid erosion or flooding and cannot have natural resources (e.g. mineral, oil or gas reserves). If aquifers with potable water are present, these cannot be significantly affected by the disposals. Although the BDC can safely accommodate a wide range of hydrogeological conditions, a site with a low groundwater flow rate at depth would be preferred. Also, geochemical conditions which lead to a low level of corrosion of the waste containers would be favoured.

The RWMO conducts a site selection process on behalf of the Member State and establishes a shortlist of three candidate sites potentially suitable for the implementation of the BDS. Some site characterization studies were performed at these sites, which led to the identification of a preferred site. More detailed site characterization studies were performed, with the aim of gaining a general understanding of the preferred site in terms of its regional setting, its past evolution and future natural evolution.

SITE CHARACTERISTICS

General

In the hypothetical example, the preferred site selected for the implementation of the BDS is located in a relatively sparsely populated rural area of the Member State within the site boundary of the DSRS conditioning facility operated by the RWMO, which means that it is government-owned land. The proximity of the conditioning facility was a major consideration in the site selection process, but the site is considered suitable even if the conditioning facility had been located elsewhere.

The area is characterized by an undulating topography, with a general topographical gradient towards the low lying drainage lines in the area. Geomorphologically, the area is stable with low erosion rates. Generally, surface flooding is not associated with the undulating plains.

Geology and hydrogeology

The site geology comprizes weathered crystalline rocks to a depth of 200 m, with unweathered bedrock at depths of 200 m and deeper. The site is geologically stable, with no evidence of large scale tectonic or volcanic activity. Core drilling revealed no major faults that may serve as preferential flow paths.

The water table in the area is at 65 m, with a hydraulic gradient of 1% in the direction of groundwater flow associated with the low lying areas. A regional hydrocensus confirmed a good correlation (on the order of 86%) between the groundwater levels in the area and the topography.

The hydraulic conductivity of the weathered rock is determined to be in the order of 1000 m/y, on average. The percolation rate through the unsaturated zone is determined through detailed modelling and is found to be in the order of 0.05 m/y.

The depth to the water table means that the disposal zone for the BDS may be located in the unsaturated zone or the saturated zone. It is still uncertain which option to use as the disposal zone. Therefore, both options are evaluated below as part of the safety assessment to define the optimal disposal depth from a long-term safety perspective.

Geochemistry

The geochemistry of the site, as determined through sampling and analysis conducted as part of the hydrocensus, is characterized by low pH meteoric water in the unsaturated zone that is often associated with acid rainwater conditions (pH of 4.1 and Eh of 996 mV). In the saturated zone, the geochemistry is typical of what one would expect in granitic rock with a moderate alkaline composition (pH of 8.46 and Eh of -281 mV).

Biosphere

The biosphere conditions are typical of what one would expect in a dry temperate rural area with a mean annual precipitation on the order of 450 mm. This is not sufficient for domestic and agricultural water needs and, therefore, the local population are dependent on additional water sources.

Those members of the public that reside in the vicinity of the site abstract groundwater mainly for personal use (e.g., bathing and drinking water) but also to sustain a small scale subsistence farm consisting of crops (green and root vegetables) and animals (cattle for meat and cows for milk). Small scale subsistence farming is the main land use condition in the area. The crops are under irrigation, whereas the cattle and cows roam the natural pasture. However, the abstracted groundwater is also used as drinking water for the animals.

Typical groundwater abstraction boreholes in the area extend down to depths of 100 m to 150 m into the weathered rock, with the average water level at 65 m below surface.

DISPOSAL FACILITY CHARACTERISTICS

The Member State makes the decision to adopt the BDS developed through support provided by the IAEA. This means that the waste disposal packages are consistent with the description considered in Ref. [16], which in turn is consistent with the configuration provided in the GSA [13] (see Figure 24). However, the disposal container and capsule are updated to the SDC-250 design as shown in Figure 25. In the case of the Category 1 and Category 2 Co-60 sources, the 316/L stainless steel is replaced with superduplex stainless steel, consistent with the recommendations of Ref. [17].

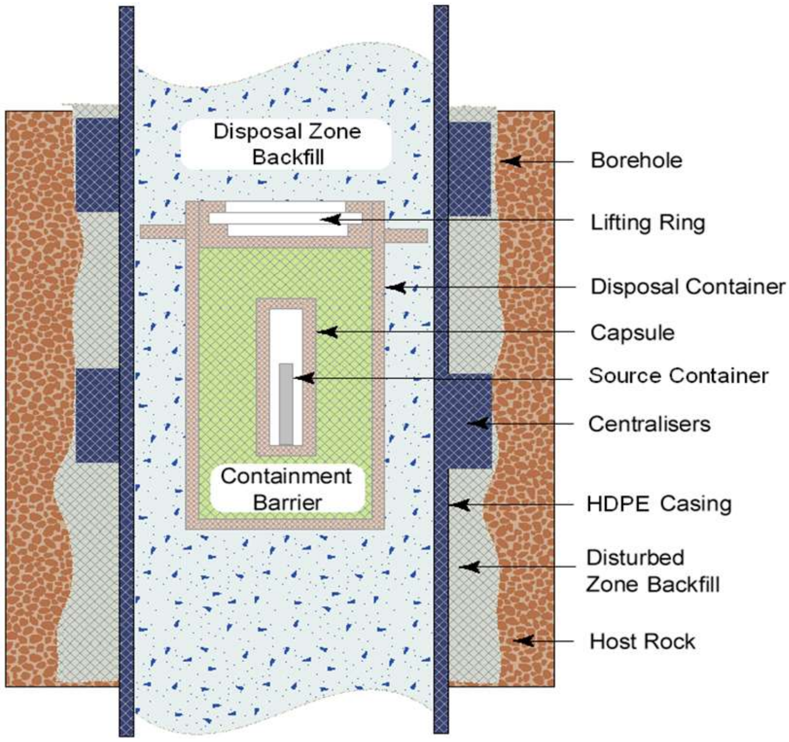

Figure 24. Cross-section through the disposal borehole.

The disposal borehole is 260 mm in diameter and will be fitted with high density polyethylene casing. The inner and outer diameters of the casing are 140 mm and 160 mm, respectively, giving a casing thickness of 10 mm. The borehole backfill is assumed to be sulphate resistant Ordinary Portland cement grout. A pitch height of 1 m per waste package will be used. The 50 mm gap between the casing and the borehole wall is backfilled with the borehole backfill. There is a 0.5 m plug of backfill at the base of the disposal borehole.

The number of waste packages required and the design of the remainder of the disposal borehole (e.g., depth of disposal and the length of the disposal zone) are a function of the DSRS inventory. Through the use of SIMBOD, the RWMO determines that 24 capsules and waste containers are required for the inventory listed in Table 11. Table 12 summarizes the resulting disposal facility dimensions and configurations for disposal in the saturated and unsaturated zones, respectively.

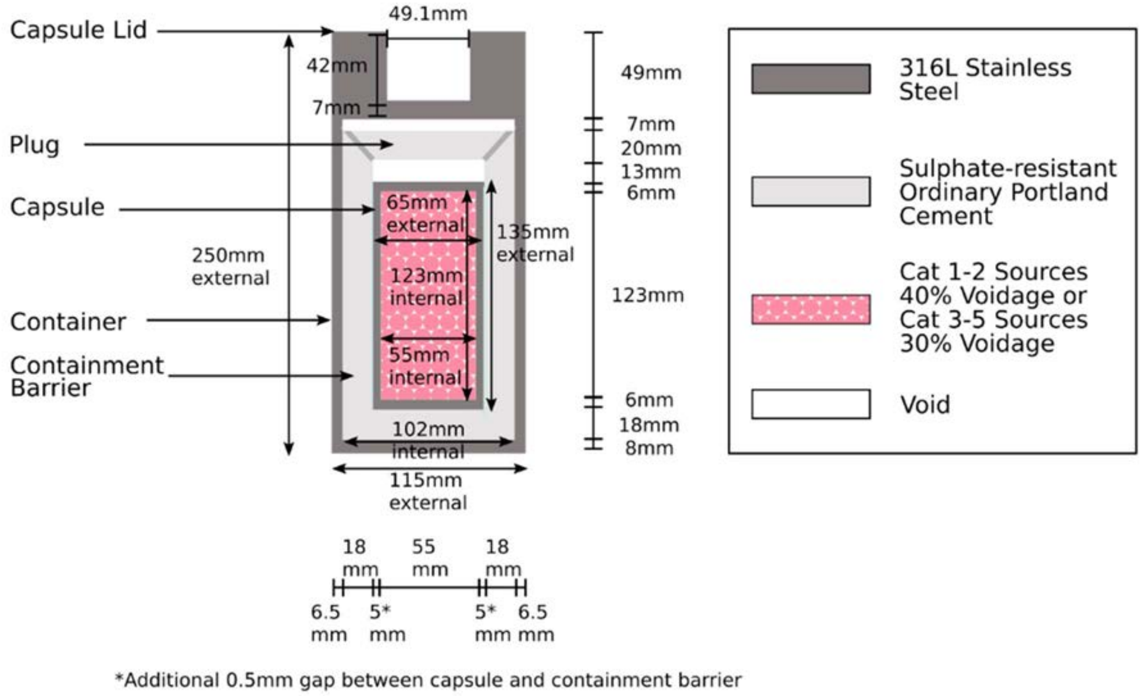

Figure 25. SDC-250 container and capsule.

TABLE 12. CALCULATED DISPOSAL FACILITY DIMENSIONS AND CONFIGURATIONS FOR DISPOSAL IN THE SATURATED AND UNSATURATED ZONES

Parameters		Units	Saturated Zone	Unsaturated Zone
Disposal Borehole Diameter		mm	260	
Casing	Outer Diameter		160	
	Inner Diameter		140	
	Thickness		10	
Gap Between Casing and Borehole Wall		m	0.5	
Pitch Height			1	
Closure Zone Thickness			75	30
Disposal Zone Thickness			24	
Borehole Plug Thickness			0.5	
Distance to the Disposal Zone			10.5	-
Distance to the Water Table			-	10.5
Depth to the Water Table			65	
Total Disposal Borehole Depth			100	55

POST-CLOSURE SAFETY ASSESSMENT

General

With the emphasis on post-closure safety assessment, this section illustrates the use of the hierarchy of models to progressively move from the simple models (Tier 1) to the more complex models (Tier 5) as part of the graded approach to safety assessment.

Assumptions and conditions

For the purpose of the example calculations, the following assumptions were made, and conditions applied consistent with the description provided at the beginning of this appendix (note that these assumptions and conditions are for illustrative purposes and are applicable only to the calculations presented in this section; they are not necessarily applicable to other post-closure safety assessments):

- The dose constraint of 0.3 mSv/y was used as the compliance criteria in all calculations. However, other dose or risk criteria might be applicable for country-specific assessments or for specific calculations associated with Defect Scenarios, for example.

- The Tier 3, Tier 4 and Tier 5 assessment calculations considered only the Design Scenario and the Defect Scenario D4 as defined in the GSA [13]. The Design Scenario represents the most realistic set of conditions, while the Defect Scenario D4 represents the most conservative set of conditions.

- Both the water release and the gas pathways were considered in the applicable tiers (Tier 3 to Tier 5).

- The Tier 1 to Tier 5 calculations were performed using the inventory of radionuclides listed in Table 11. In the definition of the generic inventory, the assumptions summarized in Table 13 were used as justification. The inventory did not include all 31 radionuclides included in Table 10 for the following reasons:

  - Most of the short-lived radionuclides were included to illustrate their relative contributions during the Tier 1 to Tier 3 calculations.

  - The long-lived radionuclides were represented by Ra-226 and Am-241 as the most common long-lived radionuclides associated with DSRS. Others such as Pu-238 and Pu-239 are less common.

- Some of the radionuclides screened out in earlier tiers were nonetheless taken forward in subsequent tier calculations for illustrative purposes.

- The evaluations were performed for disposal in saturated and in unsaturated conditions. For disposal in the saturated zone, porous high flow conditions were assumed. The latter represents conservative conditions.

The data and parameter values required to perform the Tier 1 to Tier 5 calculations are documented in Appendix II. Most of these data and parameter values were defined to be consistent with the data presented in the GSA. A safety assessment for a specific disposal facility would require site-specific data and parameter values.

TABLE 13. DSRS INVENTORY

Radionuclide	Inventory (Bq)	Comment
Co-60	1.11E+14	Consistent with the Cat 1-2 inventory assessed in Ref. [16]. Five times higher than Ghana Cat 2 Chinese source. Allows consideration of thermal and radiolysis impacts from a Cat 1-2 source.
Cs-137	5.1E+11	Consistent with Malaysian inventory and the Cat 3-5 inventory assessed in Ref. [16].
Ra-226	3.0E+11	Consistent with the Cat 3-5 inventory assessed in Ref. [16] – a conservative upper limit for Ra-226 inventory. Three times higher than total Malaysian Ra-226 inventory and thirty times higher than total Ghana Ra-226 inventory.
Am-241	1.7E+12	Consistent with the Cat 3-5 inventory assessed in Ref. [16] – assumed to be two well logging sources. An order of magnitude higher than the total Malaysian Am-241 inventory and two orders of magnitude higher than total Ghana Am-241 inventory.
Ni-63	2.0E+10	Illustrative inventory values taken to be slightly lower than the maximum inventory values cited in Table 24 of Ref. [13].
Au-195	3.00E+07	
Ba-133	4.50E+07	
Cd-109	2.10E+09	
Co-57	1.00E+10	
Eu-152	3.00E+07	
Fe-55	8.00E+08	
Gd-153	1.00E+11	
Hg-203	1.00E+06	
Ir-192	8.30E+14	
Mn-54	1.00E+05	
Na-22	2.40E+06	
Po-210	1.00E+10	
Se-75	2.00E+11	
Sm-151	2.00E+08	
Tl-204	4.00E+07	
Y-88	1.00E+05	
Yb-169	1.20E+11	
Zn-65	2.70E+05	

Note*: This inventory differs from the example unit inventory (50 capsules each containing 1 TBq of each radionuclide) considered in the GSA [13]. Some values are based on realistic inventories that could require disposal in the Member States, while others are illustrative inventory values taken to be slightly lower than the maximum inventory values cited in Table 24 of Ref. [13].*

Tier 1

As noted in Section 4.3.1, two options are available for the Tier 1 assessment. Here the Tier 1 assessment assumes that a human is directly exposed to the entire inventory of each radionuclide following the end of the institutional control period, which is assumed to be 30 years. There is no differentiation between the Design Scenario and the Defect Scenario D4.

A spreadsheet was used to calculate the potential doses (see Table 14). These doses have no relationship with doses that could occur in reality. Their only use is to identify those radionuclides in the inventory that potentially could be relevant, and to screen out those radionuclides that will have decayed to radiologically irrelevant activity concentrations after 30 years.

Doses for eight radionuclides are greater than the assumed dose constraint of 0.3 mSv/y (highlighted in bold in Table 14) and, therefore, Co-60, Ni-63, Ba-133, Cs-137, Sm-151, Eu-152, Ra-226 and Am-241 need to be considered in Tier 2 calculations. The other radionuclides in the inventory will not lead to any relevant radiological exposures after the assumed institutional control period of 30 years and do not need to be considered further.

TABLE 14. TIER 1 CALCULATED POTENTIAL DOSES

Radionuclide	Inventory at 30 y (Bq)	Ingestion Dose (Sv)	Inhalation Dose (Sv) [(a)]	External Dose (Sv) [(b)]
Na-22	8.1E+02	2.6E-06	1.0E-06	2.8E-13
Mn-54	2.8E-06	2.0E-15	4.2E-15	3.8E-22
Fe-55	3.6E+05	1.2E-04	2.8E-04	1.2E-12
Co-57	7.4E-03	1.6E-12	4.1E-12	1.4E-19
Co-60	2.1E+12	**7.3E+03**	**2.1E+04**	**7.5E+00**
Ni-63	1.6E+10	**2.4E+00**	**7.7E+00**	0.0E+00
Zn-65	8.2E-09	3.2E-17	1.3E-17	7.6E-25
Se-75	5.9E-17	1.5E-25	5.9E-26	3.6E-33
Y-88	1.2E-26	1.5E-35	5.2E-35	5.1E-42
Cd-109	1.6E+02	3.3E-07	1.3E-06	8.3E-17
Ba-133	6.4E+06	**9.7E-03**	**2.0E-02**	3.7E-10
Cs-137	2.6E+11	**3.3E+03**	**1.2E+03**	**2.0E-01**
Sm-151	1.6E+08	**1.6E-02**	**6.3E-01**	7.1E-09
Eu-152	6.3E+06	**8.8E-03**	**2.6E-01**	1.0E-05
Gd-153	2.4E-03	6.5E-13	5.0E-12	2.0E-20
Yb-169	9.7E-93	6.9E-102	2.9E-101	4.3E-109
Ir-192	2.7E-30	3.8E-39	1.8E-38	3.5E-46
Au-195	2.8E-11	7.1E-21	4.8E-20	3.6E-28
Hg-203	2.8E-65	5.3E-74	6.7E-74	1.1E-81
Tl-204	1.6E+05	2.0E-04	6.4E-05	2.7E-14
Po-210	1.5E-14	1.8E-20	4.9E-20	1.8E-31
Ra-226 [(c)]	3.0E+11	**8.3E+04**	**1.0E+06**	**7.0E-01**
Am-241 [(c)]	1.6E+12	**3.2E+05**	**6.8E+07**	**4.8E-02**

Notes:

[(a)] Assumes inventory to be in 1 m^3 of air and exposure duration of 1 hour.
[(b)] Assumes an exposure duration of 10 hours.
[(c)] No consideration of ingrowth of daughters with a half-life of 25 days or more.

Tier 2

As noted in Section 4.4.2, the results from the Tier 2 evaluation can be compared to WAC calculated in the GSA or the clearance levels published in GSR Part 3 [31]. To do this, the radiological hazard of the disposed waste was quantified by calculating the hazard quotient (HQ) defined as:

$$HQ_j = \frac{Conc_j}{CL_j}$$

where:

HQ_j is the hazard quotient for the j^{-th} radionuclide [unitless],

$Conc_j$ is the concentration of the j^{-th} radionuclide in the disposed waste [Bq/kg],

CL_j is the clearance level for the j^{-th} radionuclide [Bq/kg].

The initial radionuclide concentrations in the waste were calculated by dividing the initial radionuclide inventory by the total mass of the disposed waste, which was estimated by multiplying the volume of the disposed waste by the bulk density of the disposed waste. The radionuclide concentrations in the disposed waste as a function of time were obtained by accounting for radionuclide decay and for the ingrowth of daughter radionuclides. The time-dependent concentrations were used for calculating time-dependent HQs.

As used here, the HQs are a measure of the hazard posed by each radionuclide present in the waste – the higher the HQ, the higher the hazard. The sum of the HQs for all radionuclides present in the waste gives the total HQ and is a measure of the total radiological hazard of the waste. If the total HQ is below 1, then according to the calculations, the waste could be cleared and no longer considered as radioactive waste.

Table 15 presents the initial concentrations for all the radionuclides listed in Table 11 and the calculated HQs. The eight radionuclides remaining from the Tier 1 calculations (Co-60, Ni-63, Ba-133, Cs-137, Sm-151, Eu-152, Ra-226 and Am-241) are highlighted in bold. Figure 26 is a graphical representation of the HQ values. Table 15 and Figure 26 show that, based on the HQs for the initial concentrations, some of the screened Tier 1 radionuclides are still included, while some of the remaining Tier 1 radionuclides are eliminated based on the HQs for the initial concentrations. The total HQ is dominated by the short-lived Ir-192.

Figure 27 presents the time-dependent HQs higher than a value of 1 and with a starting time of 10 years, considering the decay and ingrowth of radionuclides. The results show that the total HQ decreases rapidly due to decay of short-lived radionuclides; after the 30-year institutional control period, only four radionuclides have a HQ of more than 1: Co-60, Am-241, Ra-226, and Cs-137.

TABLE 15. INITIAL CONCENTRATION FOR EACH RADIONUCLIDE AND CALCULATED HAZARD QUOTIENTS BASED ON THE CLEARANCE LEVEL IN GSR PART 3 [31]

Radionuclide	**Activity (Bq)**	**Initial Concentration (Bq/kg)**	**Clearance Level (Bq/kg)**	**HQ**
Ir-192	8.30E+14	9.77E+11	1.00E+04	9.77E+07
Co-60	**1.11E+14**	**1.31E+11**	**1.00E+04**	**1.31E+07**
Am-241	**1.70E+12**	**2.00E+09**	**1.00E+03**	**2.00E+06**
Ra-226	**3.00E+11**	**3.53E+08**	**1.00E+04**	**3.53E+04**
Se-75	2.00E+11	2.35E+08	1.00E+05	2.35E+03
Yb-169	1.20E+11	1.41E+08	1.00E+05	1.41E+03
Gd-153	1.00E+11	1.18E+08	1.00E+05	1.18E+03
Po-210	1.00E+10	1.18E+07	1.00E+04	1.18E+03
Cs-137	**5.10E+11**	**6.00E+08**	**1.00E+06**	**6.00E+02**
Co-57	1.00E+10	1.18E+07	1.00E+05	1.18E+02
Ba-133	**4.50E+07**	**5.30E+04**	**1.00E+05**	**5.30E-01**
Au-195	3.00E+07	3.53E+04	1.00E+05	3.53E-01
Eu-152	**3.00E+07**	**3.53E+04**	**1.00E+05**	**3.53E-01**
Na-22	2.40E+06	2.82E+03	1.00E+04	2.82E-01
Cd-109	2.10E+09	2.47E+06	1.00E+07	2.47E-01
Ni-63	**2.00E+10**	**2.35E+07**	**1.00E+08**	**2.35E-01**
Fe-55	8.00E+08	9.41E+05	1.00E+07	9.41E-02
Zn-65	2.70E+05	3.18E+02	1.00E+04	3.18E-02
Sm-151	**2.00E+08**	**2.35E+05**	**1.00E+07**	**2.35E-02**
Hg-203	1.00E+06	1.18E+03	1.00E+05	1.18E-02
Mn-54	1.00E+05	1.18E+02	1.00E+04	1.18E-02
Y-88	1.00E+05	1.18E+02	1.00E+04	1.18E-02
Tl-204	4.00E+07	4.71E+04	1.00E+07	4.71E-03
Total				

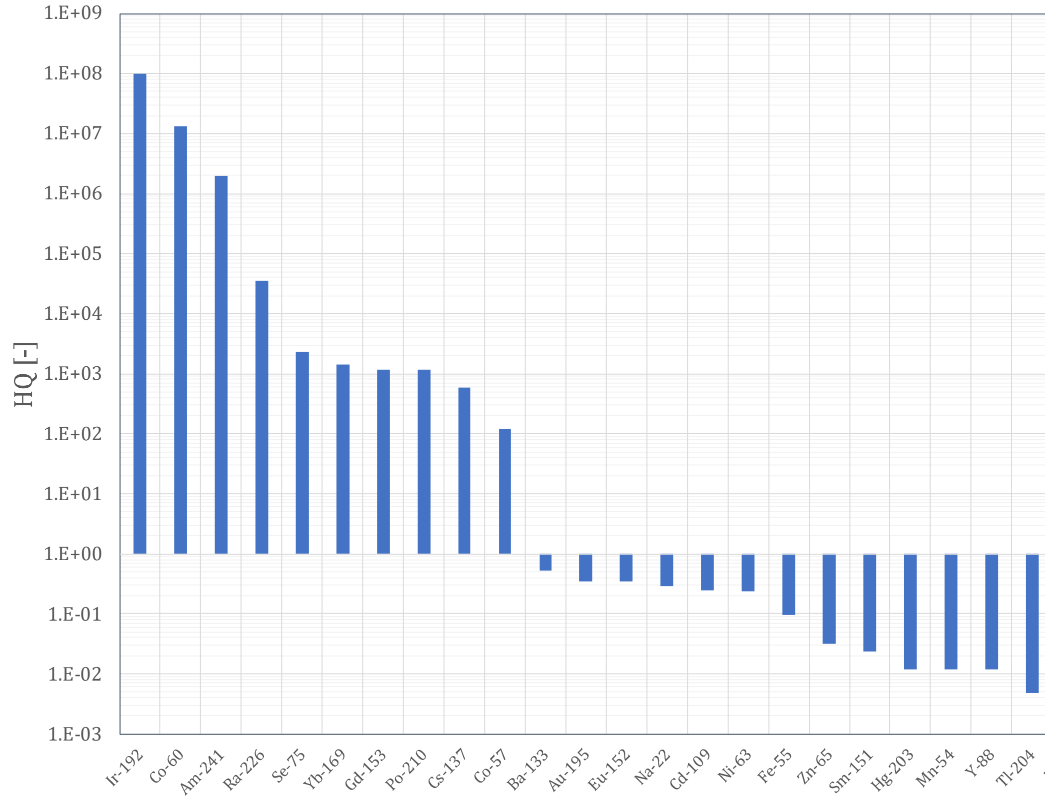

Figure 26. Initial Hazard Quotients for the inventory in Table 15.

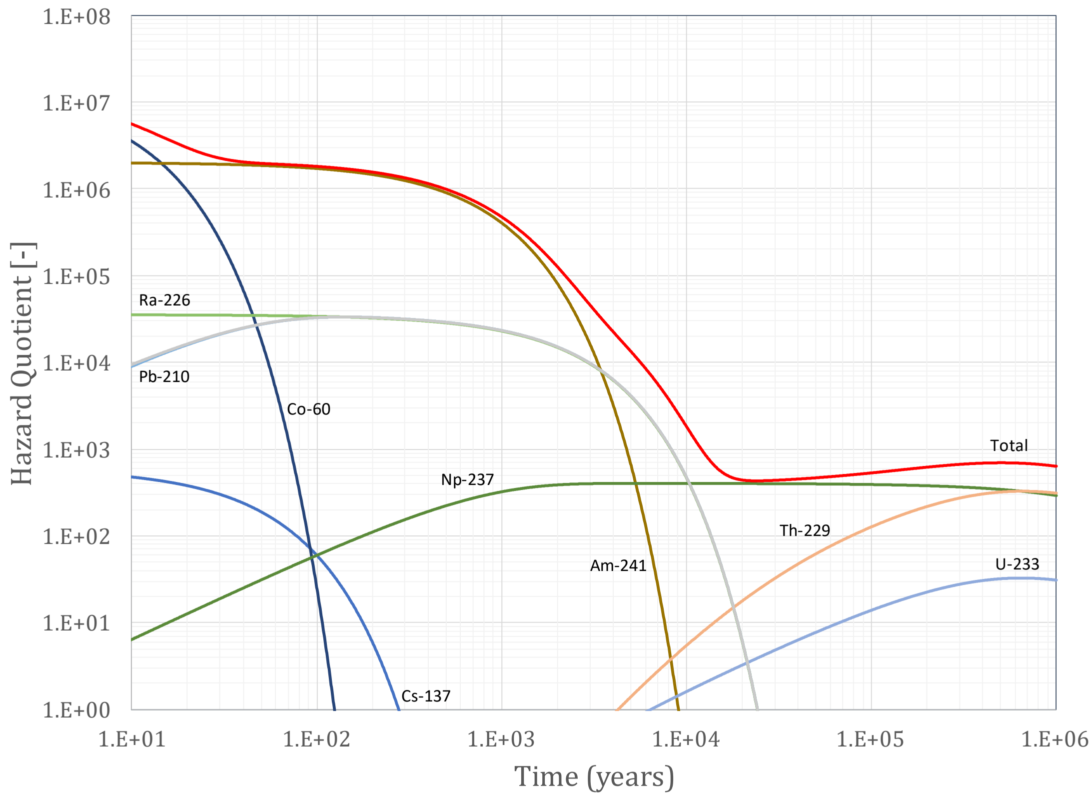

Figure 27. Hazard Quotients for the inventory in Table 15 starting from year 10.

Tier 3

The BDC Scoping Tool described in Ref. [14] was used to undertake the Tier 3 calculations. The calculations performed with the BDC Scoping Tool are not intended to reflect a real situation, but just establish a very conservative estimate of potential exposures for the given inventory disposed of in a borehole disposal facility.

For the Design Scenario, the calculated peak values of the radionuclide concentrations in groundwater and the resulting peak doses for ingestion of drinking water and their associated timings are summarized in Table 16. The calculated peak dose via the gas pathway (due to Rn-222 ingrown from Ra-226) is 910 Sv/y for disposal in the unsaturated zone. It is calculated to occur at 9200 years. These results suggest that the Ra-226 chain (for disposals in the unsaturated zone) and Am-241 chain (for disposals in the unsaturated and saturated zones) require further analysis using Tier 4 calculations for the Design Scenario, since the calculated doses for these chains (highlighted in bold in Table 16) exceed the assumed dose constraint of 0.3 mSv/y.

The Defect Scenario D4 can be approximated using the BDC Scoping Tool by reducing the thickness of the disposal container and capsule to ensure failure within a year and by reducing the inventory for the groundwater pathway by an order of magnitude to reflect the assumption that only 10% of the disposed inventory is available for release into groundwater. The calculated peak values of the radionuclide concentrations in groundwater and the resulting peak doses via ingestion of drinking water are summarized in Table 17.

The calculated peak dose via the gas pathway (due to Rn-222 ingrown from Ra-226) is 1800 Sv/y for disposal in the unsaturated zone. It occurs in the first year of disposal and assumes 100% of the disposed inventory is available for release via the gas pathway. Since for the D4 Defect Scenario, the calculated doses for five radionuclides (Co-60, Ni-63, Cs-137, Ra-226 and Am-241) exceed the assumed dose constraint of 0.3 mSv/y (highlighted in bold in Table 17), this scenario requires further analysis using Tier 4 calculations.

TABLE 16. TIER 3 CALCULATED PEAK RADIONUCLIDE CONCENTRATIONS IN A WATER ABSTRACTION BOREHOLE AND POTENTIAL DOSES FOR THE DESIGN SCENARIO

Radionuclide	Time (y)		Concentration in Abstraction Borehole Water (Bq/m^3)		Dose from Drinking Abstraction Borehole Water (Sv/y)	
	Disposal in Unsaturated Zone	**Disposal in Saturated Zone**	**Disposal in Unsaturated Zone**	**Disposal in Saturated Zone**	**Disposal in Unsaturated Zone**	**Disposal in Saturated Zone**
Co-60 [(a)]	2.1E+02	2.0E+02	1.0E+01	1.6E+00	2.5E-08	4.0E-09
Ni-63	9.2E+03	9.2E+05	<1.0E-10	<1.0E-10	<1.0E-15	<1.0E-15
Ba-133			<1.0E-10	<1.0E-10	<1.0E-15	<1.0E-15
Cs-137			<1.0E-10	<1.0E-10	<1.0E-15	<1.0E-15
Sm-151			<1.0E-10	<1.0E-10	<1.0E-15	<1.0E-15
Eu-152			<1.0E-10	<1.0E-10	<1.0E-15	<1.0E-15
Ra-226 chain			-	-	**9.1E+02**	<1.0E-15
Ra-226			*5.6E+08*	*<1.0E-10*	*1.2E+02*	*<1.0E-15*
Pb-210			*5.7E+08*	*<1.0E-10*	*2.9E+02*	*<1.0E-15*
Po-210			*5.7E+08*	*<1.0E-10*	*5.0E+02*	*<1.0E-15*
Am-241 chain			-	-	**2.9E+00**	**6.5E-01**
Am-241			*6.9E+04*	*<1.0E-10*	*1.2E+02*	*<1.0E-15*
Np-237			*3.4E+07*	*1.1E+06*	*2.7E+00*	*8.9E-02*
Pa-233			*3.4E+07*	*1.1E+06*	*2.2E-02*	*7.0E-04*
U-233			*1.3E+06*	*1.2E+06*	*4.7E-02*	*4.3E-02*
Th-229			*4.0E+05*	*1.2E+06*	*1.8E-01*	*5.2E-01*

Note:

[(a)] Co-60 concentrations and doses are calculated assuming that the disposal container and capsule fail after 100 and 200 years, respectively.

TABLE 17. TIER 3 CALCULATED PEAK RADIONUCLIDE CONCENTRATIONS IN A WATER ABSTRACTION BOREHOLE AND POTENTIAL DOSES FOR THE DEFECT SCENARIO D4

Radionuclide	Time (y)		Concentration in Abstraction Borehole Water (Bq/m^3)		Dose from Drinking Abstraction Borehole Water (Sv/y)	
	Disposal in Unsaturated Zone	Disposal in Saturated Zone	Disposal in Unsaturated Zone	Disposal in Saturated Zone	Disposal in Unsaturated Zone	Disposal in Saturated Zone
Co-60	1.1E+01	1.0E+00	2.6E+11	4.2E+10	**6.6E+02**	**1.0E+02**
Ni-63			1.8E+08	8.6E+06	**2.0E-02**	**9.4E-04**
Ba-133			2.2E+05	1.8E+04	2.4E-04	2.0E-05
Cs-137			4.0E+09	2.2E+08	**3.8E+01**	**2.1E+00**
Sm-151			1.8E+06	8.6E+04	1.3E-04	6.2E-06
Eu-152			1.7E+05	1.2E+04	1.7E-04	1.3E-05
Ra-226 chain			-	-	**1.8E+03**	**3.1E+00**
Ra-226			*3.0E+09*	*1.3E+08*	*6.1E+02*	*2.7E+01*
Pb-210			*8.6E+08*	*4.4E+06*	*4.3E+02*	*2.2E+00*
Po-210			*8.2E+08*	*2.5E+06*	*7.2E+02*	*2.2E+00*
Am-241 chain			-	-	**2.4E+03**	**1.1E+02**
Am-241			*1.7E+10*	*7.4E+08*	*2.4E+03*	*1.1E+02*
Np-237			*6.0E+04*	*2.7E+02*	*4.8E-03*	*2.1E-05*
Pa-233			*5.9E+04*	*2.4E+02*	*3.7E-05*	*1.5E-07*
U-233			*1.4E+00*	*5.3E-04*	*5.2E-08*	*2.0E-11*
Th-229			*4.7E-04*	*1.7E-08*	*2.1E-10*	*7.6E-15*

Tier 4

For both disposal options considered, i.e. disposal in the saturated and unsaturated zones, dose calculations were undertaken for the design and defect scenarios and considering release to a water abstraction borehole. In this example, the Tier 4 and Tier 5 models used the same model for the gas pathway. The results for the Tier 4 calculations are presented below.

Disposal in the saturated zone

The peak values of the potential doses obtained via the groundwater pathway for the case of disposal in the saturated zone are presented in Table 18. The time series of the calculated doses are shown in Figures 28 and 29. In Table 19, calculated peak values of the radionuclide concentrations in groundwater and water abstracted from a borehole are presented.

TABLE 18. TIER 4 CALCULATED PEAK VALUES OF POTENTIAL DOSE AND THE TIME OF THE PEAK OBTAINED FOR THE DESIGN AND DEFECT SCENARIOS FOR THE CASE OF DISPOSAL IN THE SATURATED ZONE

Radionuclide	Design scenario		Defect scenario	
	Time of peak (y)	**Annual dose (Sv/y)**	**Time of peak (y)**	**Annual dose (Sv/y)**
Am-241	9.2E+05	<10E-15	2.5E+03	5.5E-08
Co-60	2.4E+02	<10E-15	3.8E+01	4.5E-08
Cs-137	9.2E+05	<10E-15	1.9E+02	1.1E-04
Ni-63	9.2E+05	<10E-15	2.2E+02	3.0E-04
Np-237	9.4E+05	1.3E-04	2.2E+04	1.7E-04
Pb-210	9.2E+05	<10E-15	4.9E+03	**4.5E-01**
Po-210	9.2E+05	<10E-15	4.9E+03	**7.7E-01**
Ra-226	9.2E+05	<10E-15	5.1E+03	**1.4E-01**
Th-229	9.4E+05	**6.3E-04**	2.9E+04	5.0E-05
U-233	9.4E+05	6.4E-05	2.7E+04	7.9E-06

Note: Values above the dose constraint of 0.3 mSv/y are highlighted in bold.

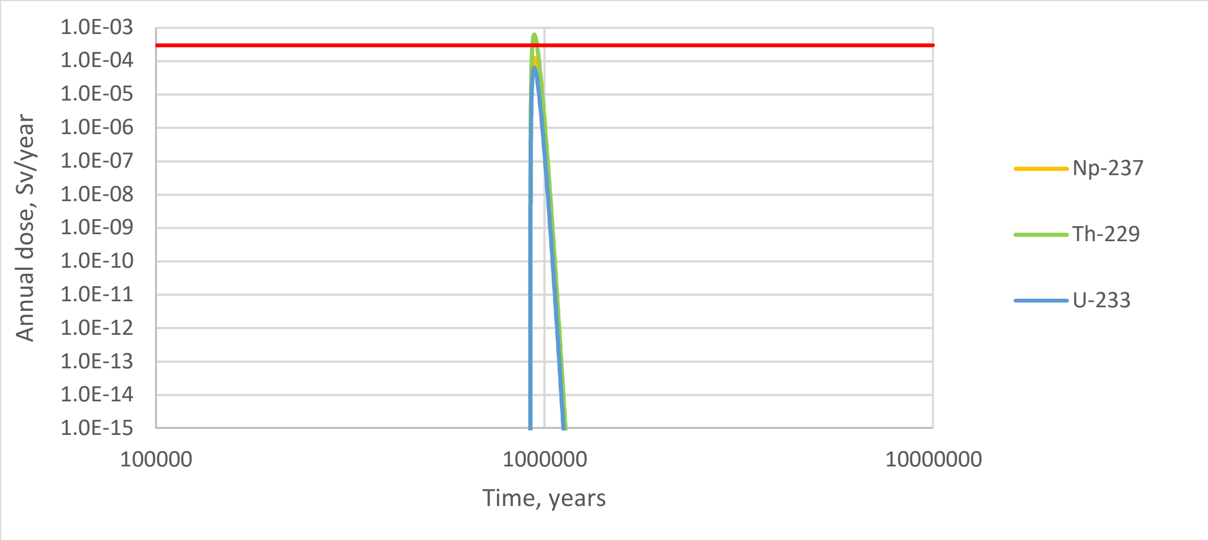

Figure 28. Time series of the radionuclide specific total annual doses obtained for the design scenario for the case of disposal in the saturated zone.

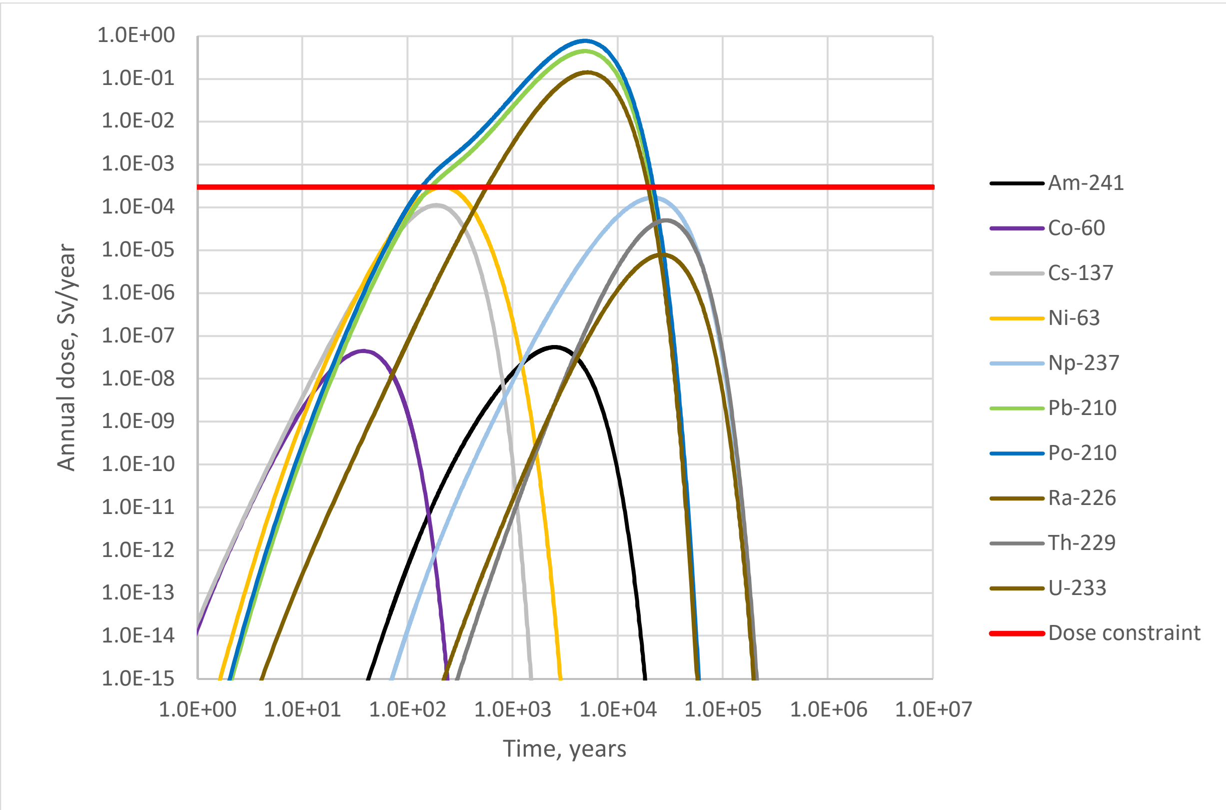

Figure 29. Time series of the radionuclide specific total annual doses obtained for the defect scenario for the case of disposal in the saturated zone.

TABLE 19. TIER 4 CALCULATED PEAK RADIONUCLIDE CONCENTRATIONS IN A WATER ABSTRACTION BOREHOLE AND IN GROUNDWATER FOR THE DESIGN AND DEFECT SCENARIOS IN THE CASE OF DISPOSAL IN THE SATURATED ZONE

Radionuclide	Design Scenario		Defect Scenario	
	Borehole Water (Bq/m^3)	Groundwater (Bq/m^3)	Borehole Water (Bq/m^3)	Groundwater (Bq/m^3)
Am-241	<10E-10	<10E-10	1.8E-02	6.6E-02
Co-60	<10E-10	<10E-10	8.9E-01	3.2E+00
Cs-137	<10E-10	<10E-10	4.1E+02	1.5E+03
Ni-63	<10E-10	<10E-10	9.1E+04	3.2E+05
Np-237	5.3E+01	1.9E+02	7.1E+01	2.5E+02
Pb-210	<10E-10	<10E-10	3.2E+04	1.1E+05
Po-210	<10E-10	<10E-10	3.2E+04	1.1E+05
Ra-226	<10E-10	<10E-10	5.9E+03	2.1E+04
Th-229	5.6E+01	2.0E+02	4.5E+00	1.6E+01
U-233	5.6E+01	2.0E+02	6.9E+00	2.5E+01

For the Design Scenario, the calculated potential doses from most radionuclides are below the screening criterion of 0.3 mSv/y and below 10 µSv/y, which is considered as the “de minimis” level in GSR Part 3 [31]. The highest potential dose values were obtained for radionuclides from the Am-241 decay chain. The highest annual doses were calculated for Th-229 which are a factor of two above the dose constraint. Hence, additional calculations for the Am-241 sources using the Tier 5 models should be carried out. It is also possible to calculate doses using other biosphere models that are less conservative than the models from Ref. [32] that are used in Tier 4. For this, the activity concentrations in water from an abstraction borehole obtained with the Tier 4 model could be used. Release rates from the geosphere calculated with the Tier 4 models can also be used as input to external biosphere models.

For the Defect Scenarios, the calculated potential doses from all radionuclides, except those from the Ra-226 decay chain, are below the 0.3 mSv/y dose constraint and below the 10 µSv/y “de minimis” level in GSR Part 3 [31]. The peak annual doses from Ra-226, Pb-210 and Po-210 together are about 4500 times above the dose constraint.

Disposal in the unsaturated zone

The peak values of the potential doses obtained for the case of disposal in the unsaturated zone are presented in Table 20. The time series of the doses are shown in Figures 30 and 31. Calculated peak values of the radionuclide concentrations in groundwater and water abstracted from a borehole are presented in Table 21.

The calculated annual doses from all radionuclides are below the 0.3 mSv/y dose constraint and are in most cases below the 10 µSv/y “de minimis” level in GSR Part 3 [31] for both the Design

Scenario and Defect Scenarios. Given the results obtained, it could be concluded that for this case, calculations with the Tier 4 model are sufficient for demonstrating compliance with the regulatory criteria and that it is, therefore, unnecessary to perform further calculations using the Tier 5 model.

TABLE 20. TIER 4 CALCULATED PEAK VALUES OF POTENTIAL DOSE AND THE TIME OF THE PEAK OBTAINED FOR THE DESIGN AND DEFECT SCENARIOS FOR THE CASE OF DISPOSAL IN THE UNSATURATED ZONE

Radionuclide	Design scenario		Defect scenario	
	Time of peak (y)	Annual dose Sv/y	Time of peak (y)	Annual dose Sv/y
Am-241	1.6E+04	<10E-15	6.8E+03	<10E-15
Co-60	2.9E+02	<10E-15	9.0E+01	<10E-15
Cs-137	9.6E+03	<10E-15	4.4E+02	<10E-15
Ni-63	1.4E+04	<10E-15	9.7E+02	8.7E-11
Np-237	3.4E+05	9.1E-06	3.3E+05	9.1E-06
Pb-210	2.5E+04	1.3E-09	1.6E+04	8.6E-08
Po-210	2.5E+04	2.3E-09	1.6E+04	1.5E-07
Ra-226	2.5E+04	4.3E-10	1.6E+04	2.8E-08
Th-229	3.7E+05	3.5E-05	3.6E+05	3.5E-05
U-233	3.7E+05	3.6E-06	3.6E+05	3.6E-06

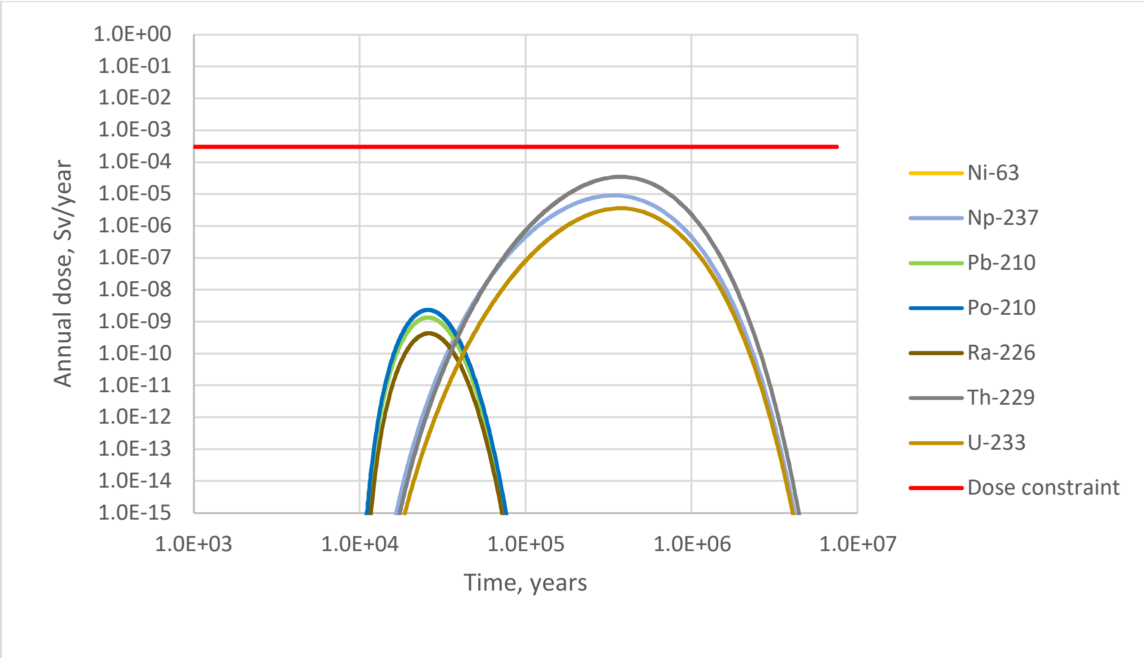

Figure 30. Time series of the radionuclide specific total annual doses obtained for the design scenario for the case of disposal in the unsaturated zone.

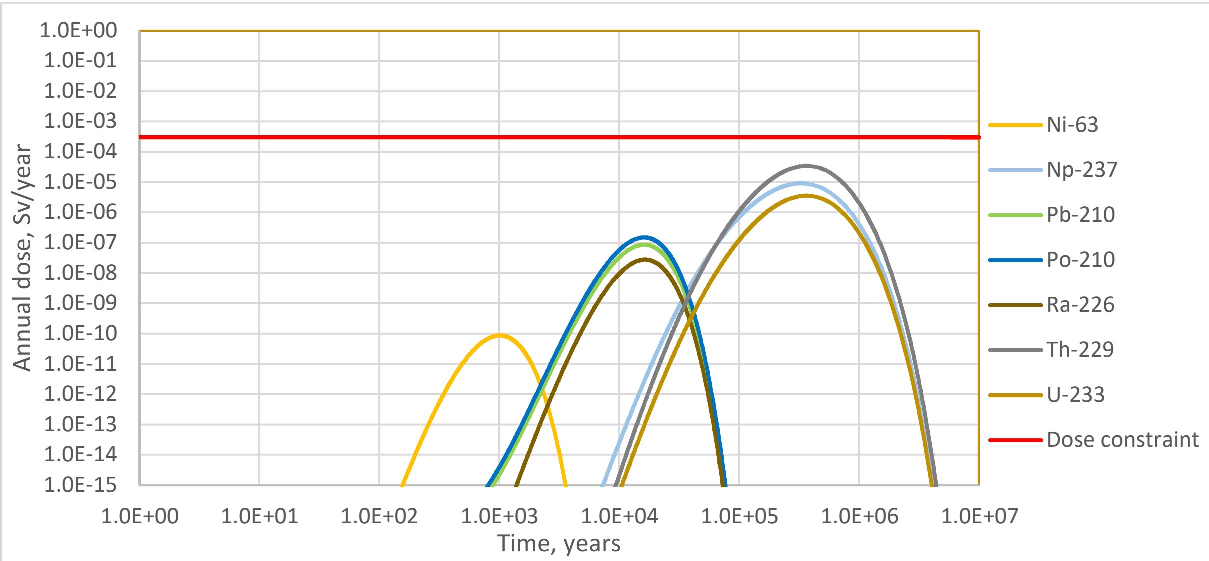

Figure 31. Time series of the radionuclide specific total annual doses obtained for the defect scenario for the case of disposal in the unsaturated zone.

TABLE 21. TIER 4 CALCULATED PEAK RADIONUCLIDE CONCENTRATIONS IN A WATER ABSTRACTION BOREHOLE AND IN GROUNDWATER FOR THE DESIGN AND DEFECT SCENARIOS IN THE CASE OF DISPOSAL IN THE UNSATURATED ZONE

Radionuclide	Design Scenario		Defect Scenario	
	Abstraction Borehole Water (Bq/m^3)	Groundwater (Bq/m^3)	Abstraction Borehole Water (Bq/m^3)	Abstraction Borehole Water (Bq/m^3)
Am-241	1.2E-25	2.2E-24	2.9E-19	5.2E-18
Co-60	6.8E-30	1.2E-28	1.8E-18	3.3E-17
Cs-137	4.7E-46	8.4E-45	7.7E-09	1.4E-07
Ni-63	2.5E-30	4.4E-29	2.2E-02	3.8E-01
Np-237	3.8E+00	6.8E+01	3.9E+00	6.9E+01
Pb-210	1.2E-04	2.1E-03	6.2E-03	1.1E-01
Po-210	1.2E-04	2.1E-03	6.2E-03	1.1E-01
Ra-226	2.2E-05	3.9E-04	1.1E-03	2.0E-02
Th-229	3.1E+00	5.6E+01	3.1E+00	5.5E+01
U-233	3.2E+00	5.6E+01	3.1E+00	5.6E+01

Tier 5

For the Design Scenario, the calculated peak values of the radionuclide concentrations in groundwater and the associated peak values of the potential doses and their associated timings are summarized for disposal in the unsaturated and saturated zones in Tables 22 and 23. The calculated potential doses are also shown in Figures 32 to 35. The calculated peak value of potential dose via the gas pathway (due to Rn-222 ingrown from Ra-226) was 1.7×10^{-12} Sv/y for disposal in the unsaturated zone. It was calculated to occur at 9200 years. All calculated peak values of the potential doses are below the assumed 0.3 mSv/y dose constraint.

For the Defect Scenario D4, the calculated peak values of the radionuclide concentrations in groundwater and the associated peak values of the potential doses and their associated timings are summarized for disposal in the unsaturated and saturated zones in Tables 24 and 25. The calculated potential doses are also shown in Figures 36 to 39. The calculated peak value of the potential dose via the gas pathway (due to Rn-222 ingrown from Ra-226) was 1.9×10^{-10} Sv/y for disposal in the unsaturated zone and was calculated to occur in the first year of disposal. Only the dose from the Ra-226 chain is calculated to exceed the assumed 0.3 mSv/y dose constraint.

TABLE 22. TIER 5 CALCULATED PEAK RADIONUCLIDE CONCENTRATIONS IN GROUNDWATER AND DRINKING WATER FOR THE DESIGN SCENARIO

Radionuclide	Time of Peak Concentration (y)		Peak Concentration in Groundwater (Bq/m^3)		Peak Concentration in Drinking Water (Bq/m^3)	
	Disposal in Unsaturated Zone	**Disposal in Saturated Zone**	**Disposal in Unsaturated Zone**	**Disposal in Saturated Zone**	**Disposal in Unsaturated Zone**	**Disposal in Saturated Zone**
Co-60 [(a)]	-	-	-	-	-	-
Cs-137 [(a)]	-	-	-	-	-	-
Ra-226 chain	-	-	-	-	-	-
Ra-226	2.6E+04	9.2E+05	*5.5E-05*	*< 1.0E-10*	*2.1E-06*	*< 1.0E-10*
Pb-210	2.6E+04	9.2E+05	*2.9E-04*	*< 1.0E-10*	*1.1E-05*	*< 1.0E-10*
Po-210	2.6E+04	9.2E+05	*2.9E-04*	*< 1.0E-10*	*1.1E-05*	*< 1.0E-10*
Am-241 chain	-	-		-		-
Am-241	1.7E+04	9.2E+05	*< 1.0E-10*	*< 1.0E-10*	*< 1.0E-10*	*< 1.0E-10*
Np-237	4.6E+05	9.5E+05	*5.1E+01*	*7.4E+01*	*1.9E+00*	*1.1E+01*
Pa-233	4.6E+05	9.5E+05	*5.1E+01*	*7.4E+01*	*1.9E+00*	*1.1E+01*
U-233	4.9E+05	9.5E+05	*4.7E+01*	*7.8E+01*	*1.8E+00*	*1.2E+01*
Th-229	4.9E+05	9.5E+05	*4.7E+01*	*7.8E+01*	*1.8E+00*	*1.2E+01*

Note:

[(a)] Decays to trivial levels before capsule fails.

TABLE 23. TIER 5 CALCULATED PEAK VALUES OF POTENTIAL DOSE FROM DRINKING OF WATER FROM AN ABSTRACTION BOREHOLE, AND FROM ALL EXPOSURE ROUTES FOR THE DESIGN SCENARIO

Radionuclide	**Time of Peak Dose (y)**		**Peak Drinking Water Dose (Sv/y)**		**Peak Total Dose (Sv/y)**	
	Disposal in Unsaturated Zone	**Disposal in Saturated Zone**	**Disposal in Unsaturated Zone**	**Disposal in Saturated Zone**	**Disposal in Unsaturated Zone**	**Disposal in Saturated Zone**
Co-60 [(a)]	-	-	-	-	-	-
Cs-137 [(a)]	-	-	-	-	-	-
Ra-226 chain	2.6E+04	9.2E+05	1.6E-11	< 1.0E-15	1.8E-10	< 1.0E-15
Ra-226	2.7E+04	9.2E+05	*4.2E-13*	*< 1.0E-15*	*4.6E-11*	*< 1.0E-15*
Pb-210	2.6E+04	9.2E+05	*5.6E-12*	*< 1.0E-15*	*6.4E-11*	*< 1.0E-15*
Po-210	2.6E+04	9.2E+05	*9.7E-12*	*< 1.0E-15*	*7.2E-11*	*< 1.0E-15*
Am-241 chain	4.9E+05	9.5E+05	1.0E-06	6.6E-06	1.3E-05	4.9E-05
Am-241	1.7E+04	9.2E+05	*< 1.0E-15*	*< 1.0E-15*	*< 1.0E-15*	*< 1.0E-15*
Np-237	4.6E+05	9.5E+05	*1.6E-07*	*8.9E-07*	*1.6E-06*	*7.4E-06*
Pa-233	4.6E+05	9.5E+05	*1.2E-09*	*7.1E-09*	*3.1E-07*	*1.9E-07*
U-233	4.9E+05	9.5E+05	*6.6E-08*	*4.4E-07*	*7.6E-07*	*3.0E-06*
Th-229	4.9E+05	9.5E+05	*7.9E-07*	*5.2E-06*	*1.0E-05*	*3.8E-05*

Note:

[(a)] Decays to trivial levels before capsule fails.

TABLE 24. TIER 5 CALCULATED PEAK RADIONUCLIDE CONCENTRATIONS IN GROUNDWATER AND DRINKING WATER FOR THE DEFECT SCENARIO D4

Radionuclide	Time of Peak Concentration (y)		Peak Concentration in Groundwater (Bq/m^3)		Peak Concentration in Drinking Water (Bq/m^3)	
	Disposal in Unsaturated Zone	**Disposal in Saturated Zone**	**Disposal in Unsaturated Zone**	**Disposal in Saturated Zone**	**Disposal in Unsaturated Zone**	**Disposal in Saturated Zone**
Co-60 [(a)]	-	-	-	-	-	-
Cs-137	4.0E+02	2.0E+02	5.0E-07	2.6E+02	1.9E-08	3.8E+01
Ra-226 chain	-	-	-	-	-	-
Ra-226	1.5E+04	6.1E+03	*2.4E-02*	*3.1E+03*	*9.1E-04*	*4.7E+02*
Pb-210	1.5E+04	6.0E+03	*1.3E-01*	*1.7E+04*	*4.9E-03*	*2.5E+03*
Po-210	1.5E+04	6.0E+03	*1.3E-01*	*1.7E+04*	*4.9E-03*	*2.5E+03*
Am-241 chain	-	-	-	-	-	-
Am-241	6.3E+03	4.3E+03	*< 1.0E-10*	*3.0E-04*	*< 1.0E-10*	*4.4E-05*
Np-237	3.3E+05	3.6E+04	*6.6E+01*	*9.9E+01*	*2.5E+00*	*1.5E+01*
Pa-233	3.3E+05	3.6E+04	*6.6E+01*	*9.9E+01*	*2.5E+00*	*1.5E+01*
U-233	3.7E+05	5.6E+04	*5.4E+01*	*1.7E+01*	*2.0E+00*	*2.6E+00*
Th-229	3.7E+05	6.5E+04	*5.4E+01*	*1.4E+01*	*2.0E+00*	*2.2E+00*

Note:

[(a)] Decays to trivial levels before reaching the abstraction borehole.

TABLE 25. TIER 5 CALCULATED PEAK VALUES OF POTENTIAL DOSES FROM DRINKING OF WATER FROM AN ABSTRACTION BOREHOLE, AND FROM ALL EXPOSURE ROUTES FOR THE DEFECT SCENARIO D4

Radionuclide	**Time of Peak Dose (y)**		**Peak Drinking Water Dose (Sv/y)**		**Peak Total Potential Dose (Sv/y)**	
	Disposal in Unsaturated Zone	**Disposal in Saturated Zone**	**Disposal in Unsaturated Zone**	**Disposal in Saturated Zone**	**Disposal in Unsaturated Zone**	**Disposal in Saturated Zone**
Co-60 [a]	-	-	-	-	-	-
Cs-137	4.0E+02	2.0E+02	< 1.0E-15	3.6E-07	2.6E-15	3.0E-06
Ra-226 chain	1.6E+04	6.0E+03	6.9E-09	3.6E-03	8.0E-08	**2.7E-02**
Ra-226	1.6E+04	6.4E+03	*1.9E-10*	*9.6E-05*	*2.0E-08*	*1.1E-03*
Pb-210	1.5E+04	6.0E+03	*2.5E-09*	*1.3E-03*	*2.8E-08*	*9.5E-03*
Po-210	1.6E+04	6.0E+03	*4.3E-09*	*2.2E-03*	*3.2E-08*	*1.6E-02*
Am-241 chain	3.7E+05	4.5E+04	1.2E-06	2.1E-06	1.5E-05	1.6E-05
Am-241	6.4E+03	4.3E+03	*<1.0E-15*	*6.5E-12*	*<1.0E-15*	*5.0E-11*
Np-237	3.3E+05	3.6E+04	*2.0E-07*	*1.6E-06*	*2.1E-06*	*9.9E-06*
Pa-233	3.3E+05	3.6E+04	*1.6E-09*	*9.5E-09*	*4.0E-07*	*2.5E-07*
U-233	3.7E+05	5.6E+04	*7.6E-08*	*9.6E-08*	*8.7E-07*	*6.7E-07*
Th-229	3.7E+05	6.5E+04	*9.0E-07*	*9.7E-07*	*1.2E-05*	*7.1E-06*

Note:

[a] Decays to trivial levels before reaching the abstraction borehole.

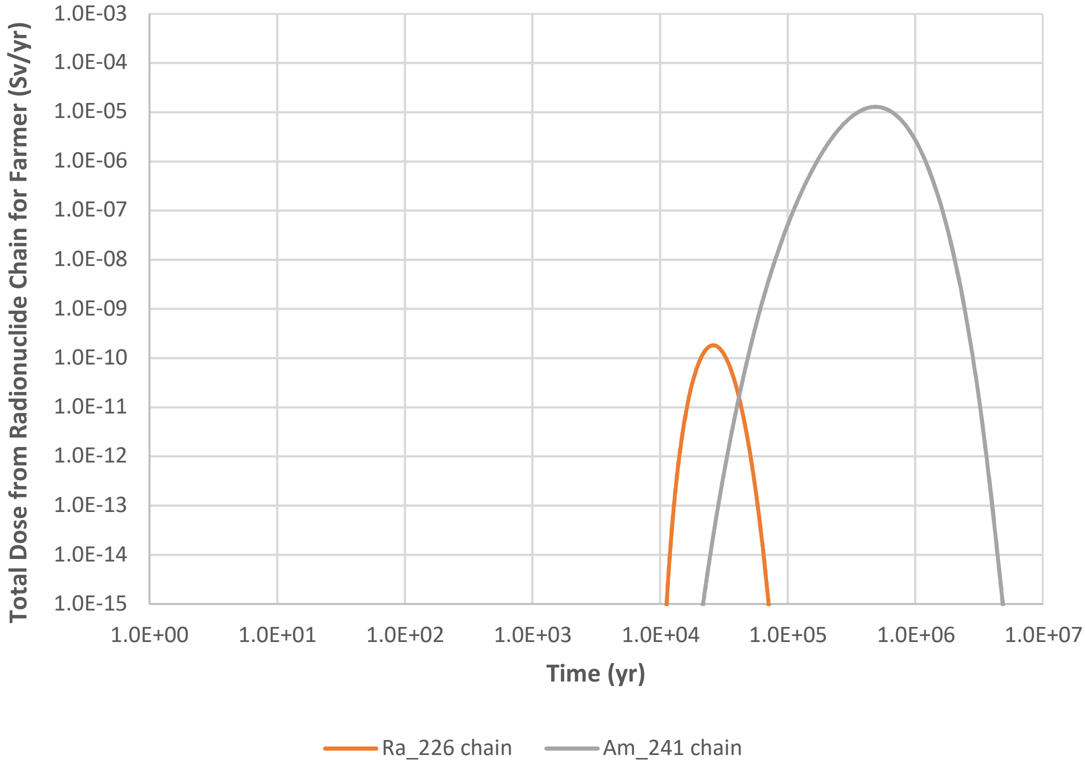

Figure 32. Tier 5 Calculated Potential Doses via the Groundwater Pathway from the Ra-226 and Am-241 Chains for the Design Scenario for Disposal in the Unsaturated Zone.

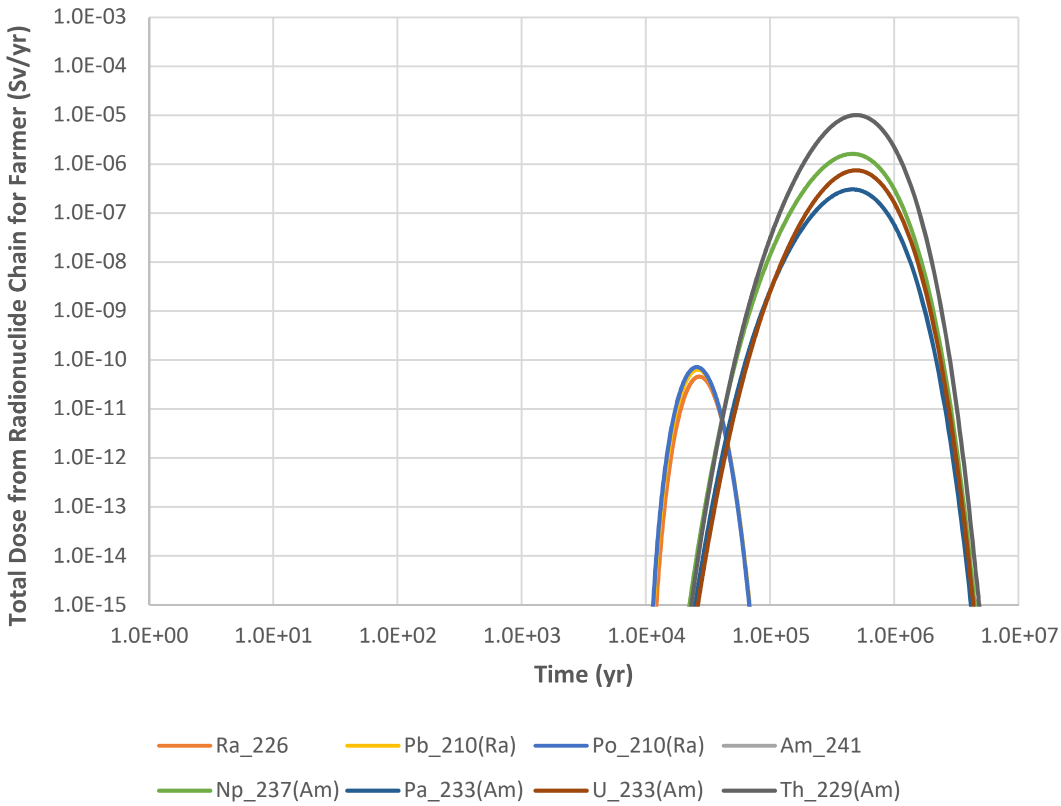

Figure 33. Tier 5 Calculated Potential Doses via the Groundwater Pathway from Indiviual Radionuclides for the Design Scenario for Disposal in the Unsaturated Zone.

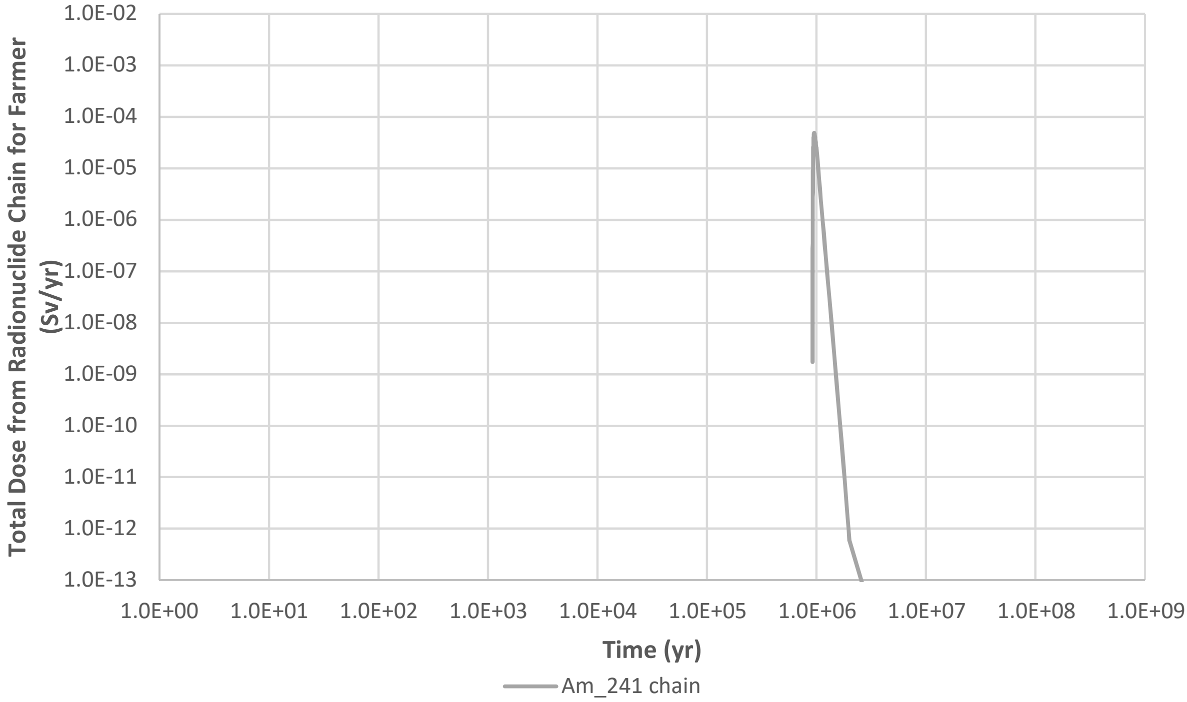

Figure 34. Tier 5 Calculated Potential Doses via the Groundwater Pathway from the Am-241 Chain for the Design Scenario for Disposal in the Saturated Zone.

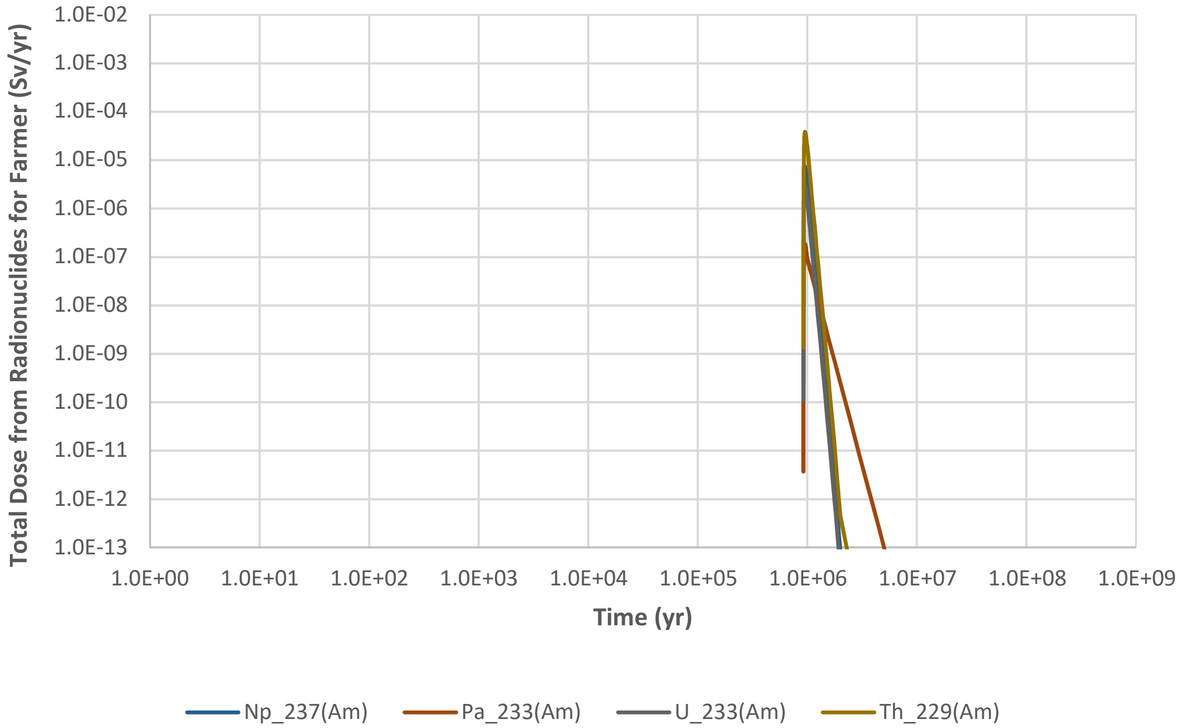

Figure 35. Tier 5 Calculated Potential Doses via the Groundwater Pathway from Individual Radionuclides for the Design Scenario for Disposal in the Saturated Zone.

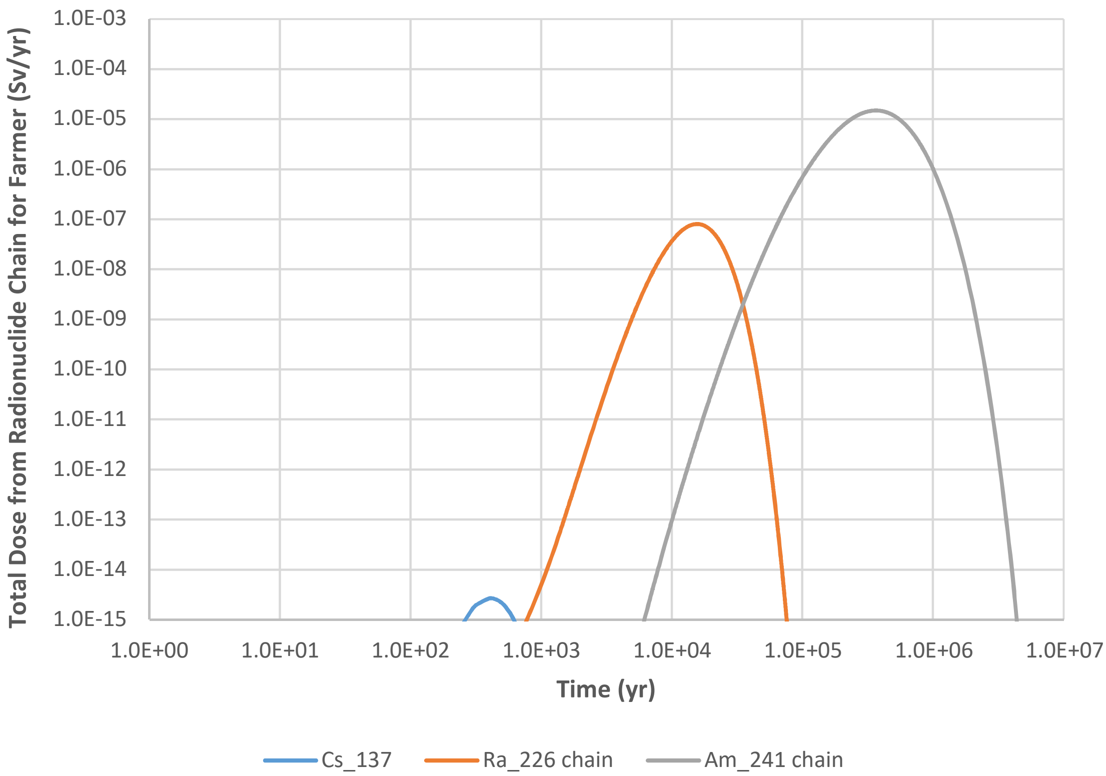

Figure 36. Tier 5 Calculated Potential Doses via the Groundwater Pathway from the Ra-226 and Am-241 Chains for the Defect Scenario D4 for Disposal in the Unsaturated Zone.

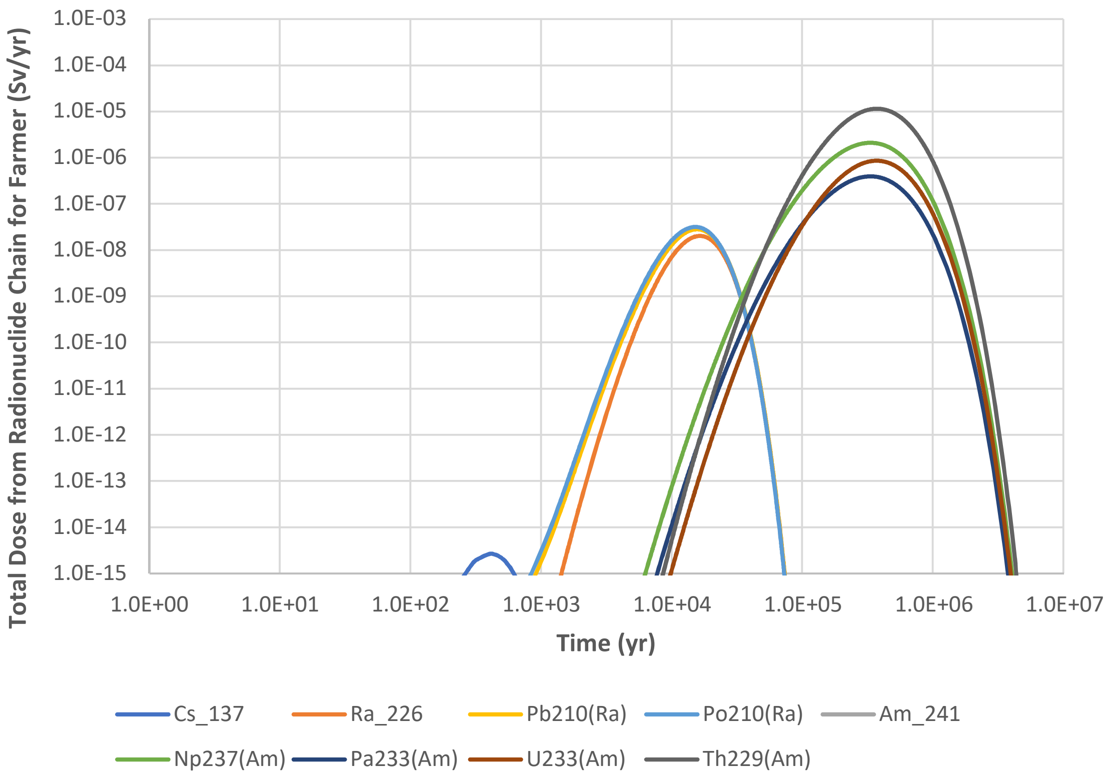

Figure 37. Tier 5 Calculated Potential Doses via the Groundwater Pathway from Individual Radionuclides for the Defect Scenario D4 for Disposal in the Unsaturated Zone.

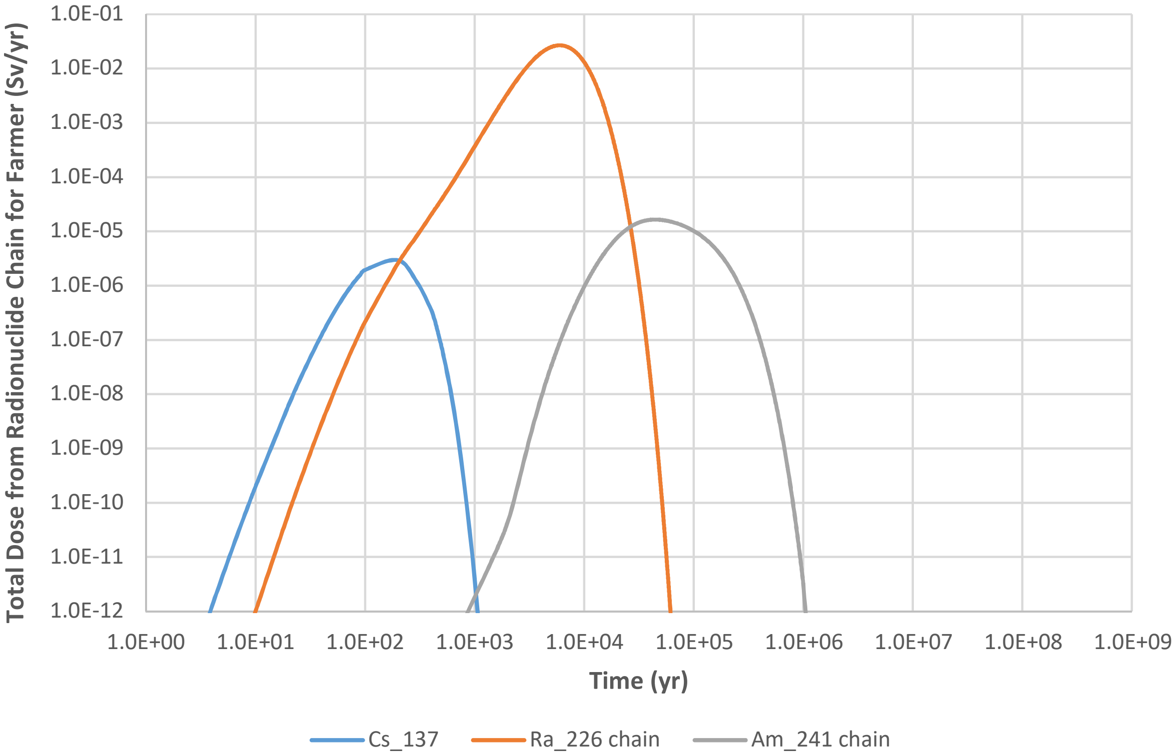

Figure 38. Tier 5 Calculated Potential Doses via the Groundwater Pathway from the Ra-226 and Am-241 Chains for the Defect Scenario D4 for Disposal in the Saturated Zone.

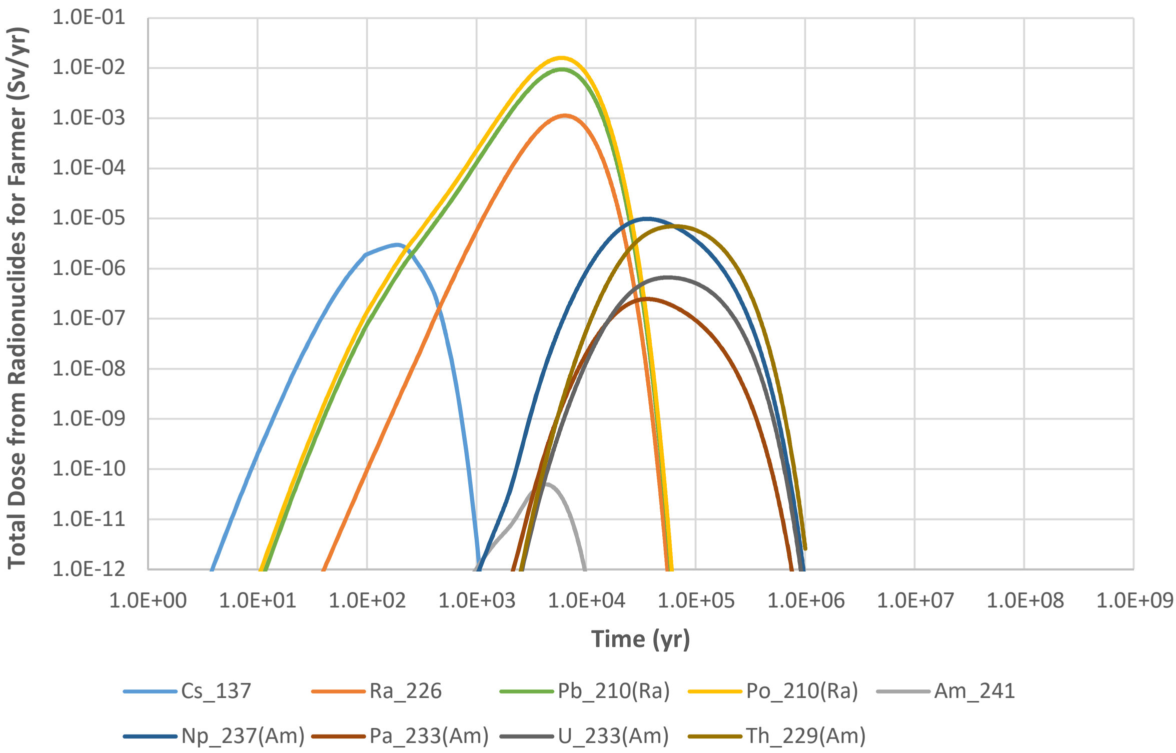

Figure 39. Tier 5 Calculated Potential Doses via the Groundwater Pathway from Individual Radionuclides for the Defect Scenario D4 for Disposal in the Saturated Zone.

Discussion of results

The tiered approach to safety assessment described in the main body of this publication was successfully applied in this appendix to an example inventory of waste DSRS. The application of the tiered approach was successful in terms of:

- Verifying that the different models had been correctly implemented and provided sensible results. The fact that the different models discussed in this publication were implemented by different experts using different software (e.g., spreadsheets, the BDC Scoping Tool, Ecolego, AMBER), but give sensible and comparable results whose differences can be understood to relate to the different degrees of conservatism in the different model tiers, is highly confidence building.

- Identifying radionuclides that might be screened out from more detailed assessment and, thereby, focussing the further safety assessment of the most important radionuclides.

- Demonstrating that the calculated potential doses decrease with increasing realism and decreasing conservatism in the models and calculations (see Figures 40 and 41).

- Building confidence in the potential for safe disposal of the inventory considered, including a range of radionuclides.

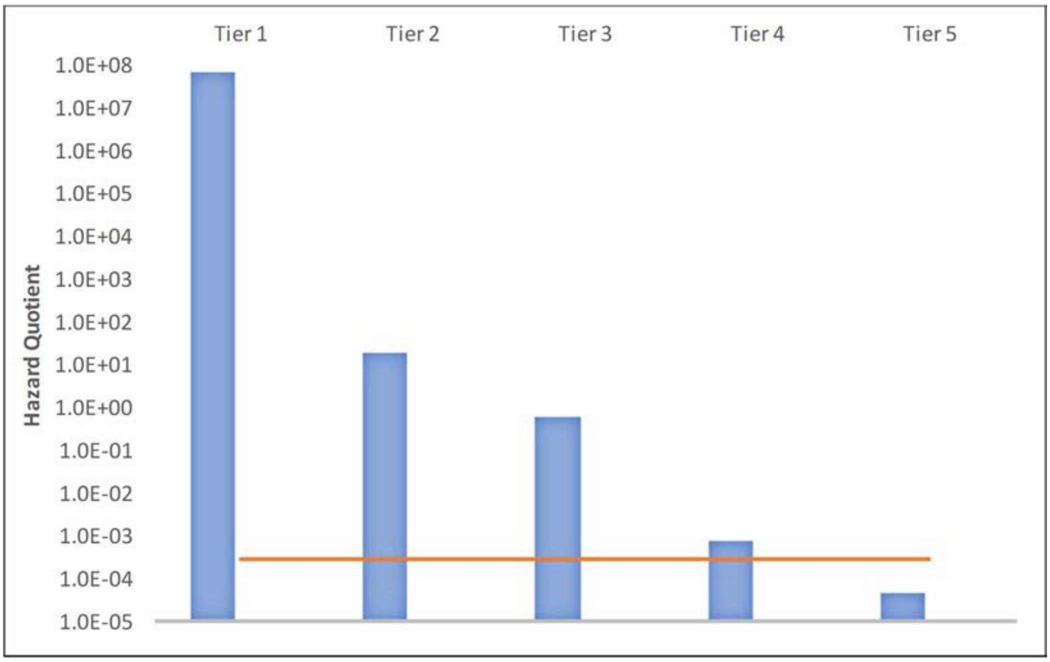

Figure 40. Comparison of the assessment results for the five models for saturated conditions, showing the decrease in risk from Tier 1 to Tier 5.

The Tier 1 and Tier 2 calculations were extremely conservative and do not provide estimates of potential doses that could actually occur, but they do allow identification of the key radionuclides for more detailed assessment in subsequent tiers.

The plot of HQ against time at Tier 2 (see Figure 27) provides an excellent way of illustrating and communicating the effect of radionuclide decay.

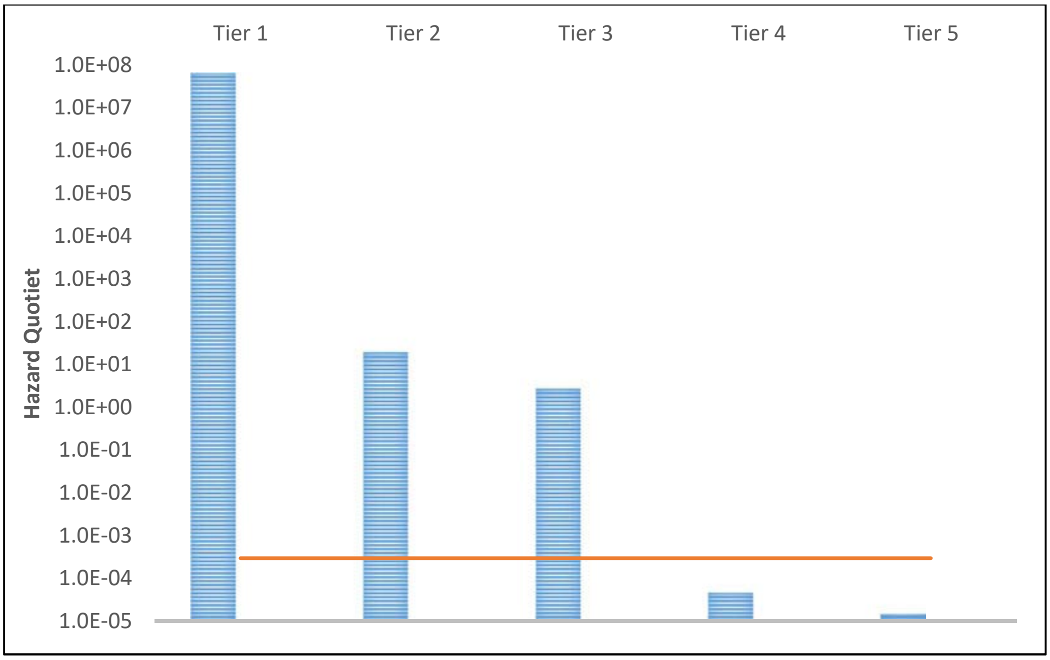

Figure 41. Comparison of the assessment results for the five models for unsaturated conditions, showing the decrease in risk from Tier 1 to Tier 5.

The Tier 3 model is implemented as the BDC Scoping Tool, which like the other models is available from the IAEA. The BDC Scoping Tool is still highly conservative. It is particularly useful for providing information on barrier performance for input to the Tier 4 and Tier 5 models.

The Tier 3 to Tier 5 models allow the differentiation of the cases involving disposal in the unsaturated and saturated zones. These models suggest that disposal in the unsaturated zone results in lower calculated potential doses as compared to disposal in the saturated zone with an assumed high rate of groundwater flow.

In the case of the example considered in this document, the Tier 3 calculations suggested that, for the inventory considered, the dose criterion might be exceeded by two radionuclides for the Design Scenario and five radionuclides for the Defect Scenario D4. At a greater level of realism, the Tier 4 calculations for disposals in the saturated zone suggested that the dose criterion might be exceeded by the Am-241 chain for the Design Scenario, and that the Ra-226 chain might exceed the dose constraint for the Defect Scenario D4. The Tier 5 calculations show the dose criterion would not be exceeded by any of the radionuclides for the Design Scenario and only by the Ra-226 chain for the Defect Scenario D4.

It is important to note that there will always be a need to consider carefully which regulatory criterion to use when considering the significance of model results from the various tiers and scenarios. The worked example used an assumed dose constraint of 0.3 mSv/y for all tiers and scenarios, even though the Defect Scenario D4 is an unrealistic, ‘what if’ scenario. One approach to evaluating the significance of scenarios that are uncertain to occur is to estimate the probability of the occurrence of the scenario and use this probability value to calculate risk (see e.g. SSR-5 [10] and SSG-23 [22]).

The comparison of the Tier 4 and Tier 5 results shows that the two sets of results are broadly consistent in terms of the timing of the peak doses and the relative importance of the radionuclides contributing to doses (see Figure 42). The more detailed Tier 5 calculations, with their more detailed discretisation of the near field, give lower doses at slightly later times than the Tier 4 calculations.

The Tier 5 calculations also show that the drinking water dose is typically within an order of magnitude of the total dose evaluated across the range of exposure pathways considered. They also confirm the Tier 4 finding that it is only doses for Ra-226 disposals calculated for the Defect Scenario D4 that result in the dose constraint of 0.3 mSv/y being exceeded for the high flow geosphere system.

The comparison of the Tier 4 and Tier 5 results builds confidence in the implementation of the GSA model in that a simplified implementation in a different calculational tool produced broadly consistent results for which the differences could be explained. For this example, the Tier 4 model was implemented in Ecolego, whereas the Tier 5 model was consistent with the GSA that was implemented in AMBER.

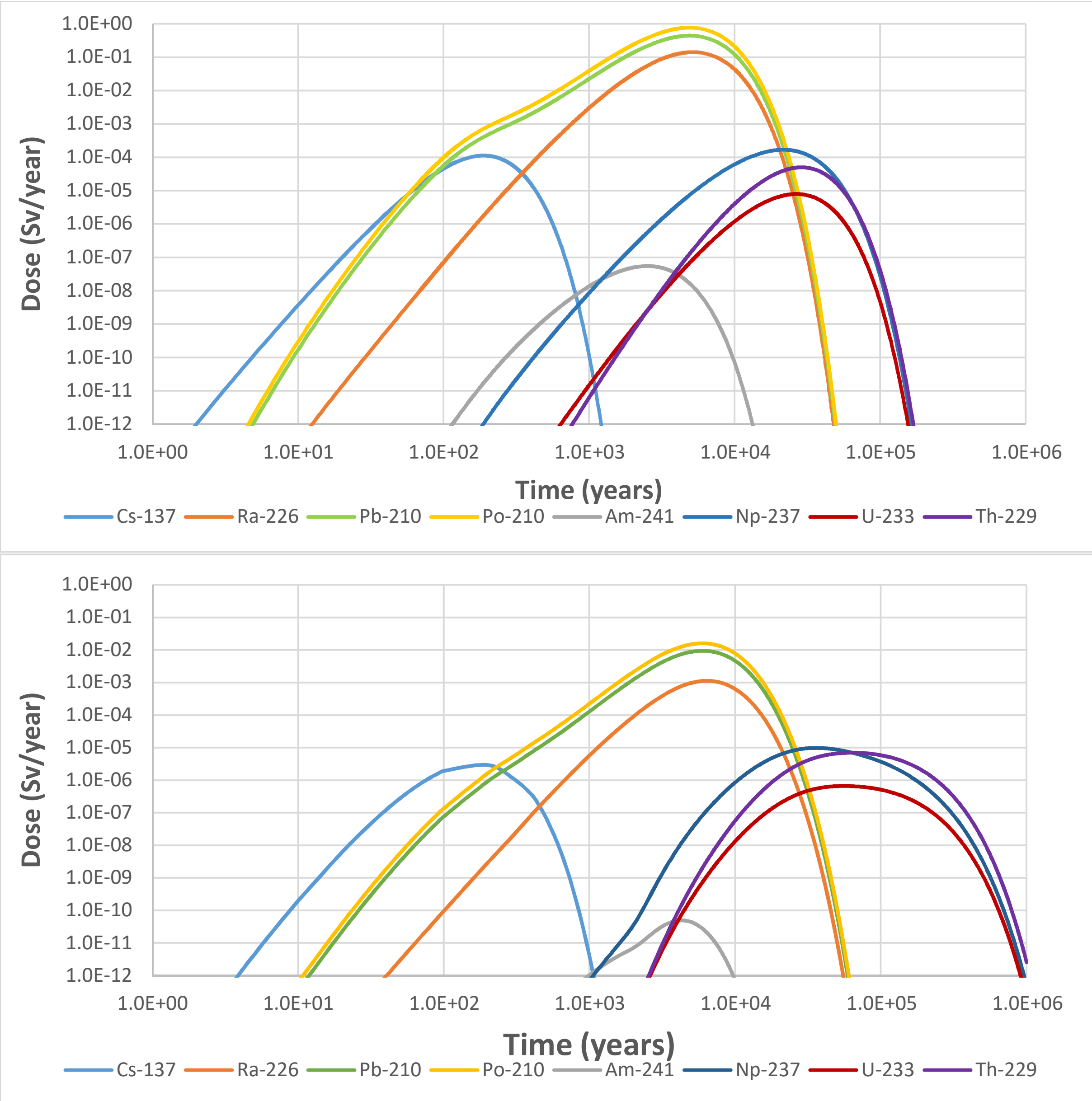

Figure 42. Comparison of calculated doses from the Tier 4 model (above) and the Tier 5 model (below) for selected radionuclides in the Defect Scenario.

APPENDIX II

DATA AND PARAMETER VALUES FOR TIER 1 TO TIER 5 MODELS

INVENTORY

The DSRS inventory given in Table 13 is used in the calculations. Associated decay information for those radionuclides considered in more detailed Tier 3 and Tier 5 calculations are given in Tables 26 to 28. Note that the Tier 4 model does not explicitly include Pa-233, rather, secular equilibrium is assumed.

TABLE 26. RADIONUCLIDES DISPOSED AND ASSOCIATED DAUGHTERS CONSIDERED (REQUIRED FOR TIER 3 TO TIER 5)

Disposed Radionuclide	**Short-lived Daughter(s)**	**Daughter(s)**
Co-60		
Ni-63		
Cs-137	*	
Ra-226	*	→ Pb-210* → Po-210
Am-241		→ Np-237 → Pa-233 → U-233 → Th-229*

TABLE 27. SHORT-LIVED DAUGHTERS WITH HALF-LIVES OF LESS THAN 25 DAYS ASSUMED TO BE IN SECULAR EQUILIBRIUM WITH THEIR PARENTS (REQUIRED FOR TIER 3 TO TIER 5)

Parent	**Short-Lived Daughters**
Cs-137	→ (branching ratio 0.94) Ba-137m
Pb-210	→ Bi-210
Ra-226	→ Rn-222→ Po-218 → (branching ratio 0.9998) Pb-214 → Bi-214 → (branching ratio 0.9998) Po-214
	→ (branching ratio 0.0002) At-218 → Bi-214 → (branching ratio 0.9998) Po-214
Th-229	→ Ra-225 → Ac-225 → Fr-221 → At-217 → Bi-213 → (branching ratio 0.9784) Po-213 → Pb-209
	→ (branching ratio 0.0216) Tl-209 → Pb-209

TABLE 28. RADIONUCLIDE HALF-LIVES AND DECAY RATES (REQUIRED FOR TIER 1 TO TIER 5)

Radionuclide	Half-life (y)	Decay rate (y^{-1})
Co-60	5.27E+00	1.32E-01
Ni-63	9.60E+01	7.22E-03
Cs-137	3.00E+01	2.31E-02
Pb-210	2.23E+01	3.11E-02
Po-210	3.79E-01	1.83E+00
Rn-222	1.05E-02	6.60E+01
Ra-226	1.60E+03	4.33E-04
Th-229	7.34E+03	9.44E-05
Pa-233	7.39E-02	9.38E+00
U-233	1.59E+05	4.36E-06
Np-237	2.14E+06	3.24E-07
Am-241	4.32E+02	1.60E-03

NEAR FIELD PARAMETER VALUES

Tables 29 to 39 document near field parameter values used in the safety assessment caluslations; the values are from Ref. [13].

TABLE 29. EFFECTIVE DIFFUSION COEFFICIENTS (M^2/Y) FOR THE NEAR FIELD (REQUIRED FOR TIER 4 AND TIER 5)

Element	Capsule	Cement	
		Undegraded	Degraded
Co	4E-2	8E-5	4E-3
Ni	8E-2	8E-5	4E-3
Cs	8E-2	8E-5	4E-3
Pb	8E-2	8E-5	4E-3
Po	8E-2	8E-5	4E-3
Ra	8E-2	8E-5	4E-3
Th	1E-1	8E-5	4E-3
Pa	1E-1	8E-5	4E-3
U	1E-1	8E-5	4E-3
Np	1E-1	8E-5	4E-3
Am	1E-1	8E-5	4E-3

Note: Tier 4 calculations only use degraded values.

TABLE 30. NEAR FIELD SORPTION COEFFICIENTS (M^3/KG) (REQUIRED FOR TIER 4 AND TIER 5)

Element	Capsule	Cement	
		Undegraded	**Degraded**
Co	0	0.1	0.01
Ni	0	0.1	0.01
Cs	0	0.001	0.005
Pb	0	0.5	0.1
Po	0	0.5	0.1
Ra	0	0.05	0.05
Th	0	5	1
Pa	0	0.5	0.1
U	0	1 (unsaturated)	0.1 (unsaturated)
		5 (saturated)	1 (saturated)
Np	0	2 (unsaturated)	0.2 (unsaturated)
		5 (saturated)	1 (saturated)
Am	0	5	1

Note: Tier 4 calculations only use degraded values.

TABLE 31. NEAR FIELD FLOW DATA (REQUIRED FOR TIER 3 TO TIER 5)

Parameter	Units	Near field		
		Capsule	Cement	
			Undegraded	**Degraded**
Hydraulic conductivity	m/y	1E+6	3.2E-1	3.2E+2
Total porosity	-	1.0E+0	1.0E-1	2.5E-1
Grain density	kg/m^3	1.0E+3	2.4E+3	2.4E+3

Note: Tier 4 calculations only use degraded values.

TABLE 32. NEAR FIELD TRANSPORT DATA (REQUIRED FOR TIER 5)

Compartment Type	Disposal Zone	Number of Compartments	Length in Direction of Flow (m)	Area Perpendicular to Flow (m^2)	Diffusion Length (m)	Area for Diffusion (m^2)
Capsules (containing source containers)	Unsaturated	1	4.92E-01	3.32E-03	2.55E-02	1.00E-01
	Saturated	1	3.25E-02	3.20E-02	2.55E-02	1.00E-01
Containment Barrier	Unsaturated	1	6.92E-01	8.17E-03	1.55E-02	2.50E-01
	Saturated	1	1.85E-02	7.06E-02	1.55E-02	2.50E-01
Disposal Zone (horizontally adjacent to disposal containers)	Unsaturated	1	4.00E+00	5.01E-03	3.13E-02	2.01E+00
	Saturated	1	1.25E-02	5.60E-01	3.13E-02	2.01E+00
Disposal Zone (vertically above disposal containers)	Unsaturated	1	2.25E+00	1.04E-02	-	-
	Saturated	-	-	-	-	-
Disturbed Zone (Backfill)	Unsaturated	1	4.00E+00	3.30E-02	2.49E-01	3.27E+00
	Saturated	1	5.00E-02	1.04E+00	2.49E-01	3.27E+00
Disposal Zone (Plug)	Unsaturated	1	5.00E-01	5.31E-02	-	-
	Saturated	-	-	-	-	-

Note: Parameter values derived using the approach described in the footnotes for Table 75 of Ref. [13] for the waste package dimensions given in Figure 25.

TABLE 33. NEAR FIELD TRANSPORT DATA (REQUIRED FOR TIER 4)

Compartment Type	Disposal Zone	Number of Compartments	Length in Direction of Flow (m)	Area Perpendicular to Flow (m^2)
Near field barrier	Unsaturated	5	1	0.0155
	Saturated	2	0.0635	0.14

TABLE 34. NEAR FIELD GAS PARAMETERS (REQUIRED FOR TIER 3 TO TIER 5)

Parameter	**Units**	**Value**
Timescale for gas production after the failure of capsule	y	1E+2
Cross-sectional area of borehole	m^2	5.3E-2
Fraction of radionuclide released as gas	-	1E+0
Emanating fraction for Rn-222	-	2E-1
Depth of closure zone	m	3E+1
Diffusion length for Rn-222 in the borehole	m	1E+0

TABLE 35. FAILURE TIMES FOR THE NEAR FIELD COMPONENTS FOR THE DESIGN SCENARIO FOR THE CO-60 WASTE PACKAGE (REQUIRED FOR TIER 5)

Component	**Failure Times (y, from time of disposal)**			
	Unsaturated Disposal Zone		**Saturated Disposal Zone**	
	Start of Failure	**Totally Failed**	**Start of Failure**	**Totally Failed**
Backfill Cement	1.85E+3	2.27E+4	5.13E+3	5.17E+3
Stainless steel disposal container [(a)]	1.00E+2		1.00E+2	
Containment Barrier	3.82E+2	3.57E+3	1.56E+3	1.57E+3
Stainless steel capsule [(a)]	2.00E+2		2.00E+2	

Note: [(a)] Conservatively assumes that superduplex stainless steel disposal container/capsule used for the Cat 1 and Cat 2 Co-60 has a lifetime of 100 years once contacted by water.

TABLE 36. FAILURE TIMES FOR THE NEAR FIELD COMPONENTS FOR THE DESIGN SCENARIO FOR THE CS-137, RA 226 AND AM-241 WASTE PACKAGES (REQUIRED FOR TIER 5)

Component	**Failure Times (y, from time of disposal)**			
	Unsaturated Disposal Zone		**Saturated Disposal Zone**	
	Start of Failure	**Totally Failed**	**Start of Failure**	**Totally Failed**
Backfill Cement	1.85E+3	2.27E+4	5.13E+3	5.17E+3
Stainless steel disposal container	8.21E+3		5.20E+5	
Containment Barrier	8.41E+3	1.06E+4	5.21E+5	5.21E+5
Stainless steel capsule	9.16E+3		9.20E+5	

TABLE 37. FAILURE TIMES FOR THE NEAR FIELD COMPONENTS FOR DEFECT SCENARIO D4 FOR THE CO-60 WASTE PACKAGE (REQUIRED FOR TIER 5)

Component	Failure Times (y, from time of disposal)			
	Unsaturated Disposal Zone		Saturated Disposal Zone	
	Start of Failure	Totally Failed	Start of Failure	Totally Failed
Backfill Cement	1.85E+3	2.27E+4	5.13E+3	5.17E+3
Defective disposal container [(a)]	0	5.00E+1	0	5.00E+1
Containment Barrier	1.93E+2	2.37E+3	1.46E+3	1.47E+3
Defective capsule [(a)]	0	5.00E+1	0	5.00E+1

Note: [(a)] Assumes pinhole defect in superduplex stainless steel disposal container/capsule used for the Cat 1 and Cat 2 Co-60 at time of disposal and disposal container/capsule fails twice as rapidly as for Design Scenario (due to internal and external corrosion). Initially, 10% of the inventory is available for release from the capsule. Once the capsule has totally failed, 100% of inventory is available for release.

TABLE 38. FAILURE TIMES FOR THE NEAR FIELD COMPONENTS FOR DEFECT SCENARIO D4 FOR THE CS-137, RA-226 AND AM-241 WASTE PACKAGES (REQUIRED FOR TIER 5)

Component	Failure Times (y, from time of disposal)			
	Unsaturated Disposal Zone		Saturated Disposal Zone	
	Start of Failure	Totally Failed	Start of Failure	Totally Failed
Backfill Cement	1.85E+3	2.27E+4	5.13E+3	5.17E+3
Defective disposal container [(a)]	0	4.11E+3	0	2.60E+5
Containment Barrier	1.93E+2	2.37E+3	1.46E+3	1.47E+3
Defective capsule [(a)]	0	8.79E+2	0	4.60E+5

Note: [(a)] Assumes pinhole defect in stainless steel disposal container/capsule at time of disposal and disposal container/capsule fails twice as rapidly as for Design Scenario (due to internal and external corrosion). Initially, 10% of the inventory is available for release from the capsule. Once the capsule has totally failed, 100% of inventory is available for release.

TABLE 39. FAILURE TIMES FOR THE NEAR FIELD COMPONENTS UNDER SATURATED AND UNSATURATED CONDITIONS (REQUIRED FOR TIER 4)

Component	Failure time (y)			
	Saturated Zone		Unsaturated Zone	
	Design Scenario	Defect Scenario	Design Scenario	Defect Scenario
316L stainless steel Container	9.2E+5	0	9.6E+3	0
Duplex stainless steel	200	-	200	0

GEOSPHERE PARAMETER VALUES

Tables 40 to 45 document geosphere parameter values used in the safety assessment calculations; the values are from Ref. [13].

TABLE 40. GEOCHEMICAL PARAMETER VALUES FOR THE UNSATURATED AND SATURATED ZONES (REQUIRED FOR TIER 3)

Determinant	Units	Unsaturated Zone	Saturated Zone
pH	pH	4.1	8.46
Eh	mV	996	-281
Cl	mg/l	0.53	0.52
SO_4	mg/l	2.88	10.66
TIC [(a)]	mg/l	0.23	42.52

Note: [(a)] TIC – total inorganic carbon.

TABLE 41. HYDROGEOLOGICAL PARAMETER VALUES FOR THE UNSATURATED AND SATURATED ZONES (REQUIRED FOR TIER 3 TO TIER 5)

Parameter	Units	Unsaturated Zone	Saturated Zone
Hydraulic conductivity	m/y	-	1E+3
Hydraulic gradient	-	-	1E-2
Percolation rate	m/y	5E-2	-
Total porosity	-	1.5E-1	1E-1
Degree of saturation	-	3.3E-1	1E+0
Grain density	kg/m^3	2.65E+3	2.65E+3

TABLE 42. GEOSPHERE TRANSPORT DATA FOR DISPOSAL IN THE UNSATURATED ZONE (REQUIRED FOR TIER 5)

Parameter	Units	Unsaturated Zone		Saturated Zone
Number of compartments	-	Adjacent to borehole	1	5
		Below borehole	5	
Length of each compartment in direction of water flow	m	Adjacent to borehole	4.00E+00	2.00E+01
		Below borehole	2.00E+00	
Area of each compartment perpendicular to water flow	m^2	Adjacent to borehole	7.32E-01	4.00E+00
		Below borehole	7.85E-01	
Diffusion length between adjacent compartments	m	Adjacent to borehole	7.32E-01	2.00E+01
		Below borehole	7.85E-01	
Area over which diffusion occurs	m^2	Adjacent to borehole	7.85E-01	4.00E+00

Note: Parameter values derived using the approach described in the footnotes for Table 79 of Ref. [13] for the waste package dimensions given in Figure 25.

TABLE 43. GEOSPHERE TRANSPORT DATA FOR DISPOSAL IN THE SATURATED AND UNSATURATED ZONE (REQUIRED FOR TIER 4)

Parameter	Units	Unsaturated Zone	Saturated Zone
Number of compartments	-	5	5
Length of each compartment in direction of water flow	m	2.00E+00	2.00E+01
Area of each compartment perpendicular to water flow	m^2	1.55E-02	5.00E+00

TABLE 44. EFFECTIVE DIFFUSION COEFFICIENTS (M^2/Y) FOR THE GEOSPHERE (REQUIRED FOR TIER 4 AND TIER 5)

Element	**D_{eff}[(a)]**
Co	2E-2
Cs	2E-2
Ni	2E-2
Pb	2E-2
Po	2E-2
Ra	2E-2
Th	2E-2
Pa	2E-2
U	2E-2
Np	2E-2
Am	2E-2

Note: (a) Effective diffusion coefficients for the case of a High and Medium Flow Rate Porous System from Table 66 in Appendix XII Ref. [13].

TABLE 45. GEOSPHERE SORPTION COEFFICIENTS (M^3/KG) (REQUIRED FOR TIERS 4 AND 5)

	Sorption Coefficient (M^3/KG)[(a)]	
Element	**Unsaturated Zone**	**Saturated Zone**
Co	0.1	0.1
Ni	0.1	0.1
Cs	0.05	0.05
Pb	0.1	0.1
Po	0.1	0.1
Ra	0.5	0.5
Th	1	1
Pa	1	1
U	1	1
Np	1	1
Am	5	5

Note: (a) Sorption coefficients from Table 67 in Appendix XII Ref. [13].

BIOSPHERE PARAMETER VALUES

The biosphere is the same stylized biosphere as considered in the GSA [13]. The associated data (taken from the GSA) are provided in Tables 46 to 54. Dose coefficients (taken from the GSA) are provided in Table 55. The Tier 3 to Tier 5 calculations also require a dose coefficient for inhalation of Rn-222 gas. 79 µSv/y per Bq/m^3 is given in the GSA.

TABLE 46. BIOSPHERE SORPTION COEFFICIENTS (M^3/KG) (REQUIRED FOR TIER 5)

Element	Unsaturated Zone
Co	0.54
Cs	1.8
Pb	0.54
Po	2.7
Ra	9.0
Th	5.4
Pa	2.7
U	1.5
Np	0.055
Am	8.1

TABLE 47. SOIL TO PLANT CONCENTRATION FACTORS (BQ/KG FRESH WT/BQ/KG DRY SOIL) FOR CROPS (REQUIRED FOR TIER 5)

Element	Root Vegetables	Green Vegetables
Co	3E-2	3E-2
Cs	3E-2	3E-2
Pb	1E-2	1E-2
Po	2E-4	2E-4
Ra	4E-2	4E-2
Th	5E-4	5E-4
Pa	4E-2	4E-2
U	1E-3	1E-3
Np	1E-3	1E-2
Am	1E-3	1E-3

TABLE 48. WEATHERING RATES (Y^{-1}) (REQUIRED FOR TIER 5)

Element	Root Vegetables	Green Vegetables
Co	1.8E+1	1.8E+1
Cs	1.8E+1	1.8E+1
Pb	1.8E+1	1.8E+1
Po	1.8E+1	1.8E+1
Ra	1.8E+1	1.8E+1
Th	1.8E+1	1.8E+1
Pa	1.8E+1	1.8E+1
U	1.8E+1	1.8E+1
Np	1.8E+1	5.1E+1
Am	1.8E+1	5.1E+1

TABLE 49. FRACTION OF ACTIVITY TRANSFERRED FROM EXTERNAL TO INTERNAL PLANT SURFACES (-) (REQUIRED FOR TIER 5)

Element	Root Vegetables	Green Vegetables
Co	1.7E-1	1.8E-1
Cs	3.0E-1	1.9E-1
Pb	2.2E-1	2.2E-1
Po	2.2E-1	2.2E-1
Ra	9.9E-2	1.8E-1
Th	2.9E-1	3.8E-2
Pa	2.9E-1	4.5E-1
U	4.3E-2	3.6E-1
Np	2.9E-1	4.5E-1
Am	2.9E-1	2.8E-1

TABLE 50. FOOD PREPARATION LOSSES (-) (REQUIRED FOR TIER 5)

Element	Root Vegetables	Green Vegetables
Co	0.0E+0	9.0E-1
Cs	0.0E+0	9.0E-1
Pb	0.0E+0	9.0E-1
Po	0.0E+0	9.0E-1
Ra	0.0E+0	9.0E-1
Th	0.0E+0	9.0E-1
Pa	0.0E+0	9.0E-1
U	0.0E+0	9.0E-1
Np	0.0E+0	9.0E-1
Am	0.0E+0	9.0E-1

TABLE 51. TRANSFER COEFFICIENTS TO ANIMAL PRODUCE (REQUIRED FOR TIER 5)

Element	Beef (d/kg fresh weight)	Cow's Milk (d/l)	Fish (m^3/kg fresh weight)
Co	1.0E-2	3.0E-4	3E-1
Cs	5.0E-2	7.9E-3	2E+0
Pb	4.0E-4	3.0E-4	3E-1
Po	5.0E-3	3.4E-4	5E-2
Ra	9.0E-4	1.3E-3	5E-2
Th	2.7E-3	5.0E-6	1E-1
Pa	5.0E-5	5.0E-6	1E-2
U	3.0E-4	4.0E-4	1E-2
Np	1.0E-3	5.0E-6	3E-2
Am	4.0E-5	1.5E-6	3E-2

TABLE 52. BIOSPHERE COMPARTMENT PARAMETERS AND PROCESSES (REQUIRED FOR TIER 5)

Parameter	**Units**	**Farmer**	**House Dweller**
		Surface Soil	**House**
Depth	m	2.5E-01	2.4E+00
Length	m	3.51E+01	3.0E+00
Width	m	1.0E+01	4.0E+00
Total porosity	-	3.0E-01	-
Degree of saturation	-	3.3E-01	-
Grain density	kg/m^3	2.65E+03	-
Percolation rate	m/y	5.0E-02	-
Inhalable dust concentration	kg/m^3	2.0E-08	-
Erosion rate	m/y	3.0E-04	-
Volume of irrigation water that reaches the soil	m^3/y	7.1E+01	-
Volume of non-irrigation water plus irrigation water intercepted by crops	m^3/y	1.95E+02	-
Ventilation rate	y^{-1}	-	2.2E+03
Effective diffusion of Rn	m^2/y	-	1.6E+01
Total porosity of the base of the house	-	-	2.5E-01

TABLE 53. INDIVIDUAL HUMAN BEHAVIOUR PARAMETERS (REQUIRED FOR TIER 5)

Exposure Mechanism		**Units**	**Exposure Group**	
			Farmer	**House Dweller**
Ingestion	Contaminated drinking water	m^3/y	0.73	-
	Contaminated root vegetables	kg fw/y [(a)]	235	-
	Contaminated green vegetables	kg fw/y [(a)]	62	-
	Contaminated beef	kg fw/y [(a)]	95	
	Contaminated cow's milk	kg fw/y [(a)]	300	-
	Contaminated soil	kg fw/h [(a)]	1.5E-5	-
	Contaminated fish	kg fw/y [(a)]	6.9	-
Inhalation	Contaminated outdoor air	m^3/h	1	-
	Contaminated indoor air	m^3/h	-	0.75
Occupancy	Time spent on contaminated soil	h/y	2192	-
	Time spent in contaminated building	h/y	-	6575
	Time spent in contaminated water	h/y	365	-

Note: [(a)] fw – fresh weight (of product) – for further details see Table 83 in Appendix XII of Ref. [13].

TABLE 54. NON-ELEMENT DEPENDENT PLANT AND ANIMAL PARAMETERS (REQUIRED FOR TIER 5)

Parameter	**Units**	**Root Vegetables**	**Green Vegetables**
Soil contamination of crop	kg dw soil/kg fw crop [(a), (b)]	1.5E-4	1.0E-4
Yield of crop	kg fw m^{-2} y^{-1} [(b)]	3.5E+0	3.0E+0
Depth of irrigation water applied to crop	m/y	3.0E-1	3.0E-1
Interception fraction for irrigation water	-	3.3E-1	3.3E-1
Time interval between irrigation and harvesting	y	4.0E-2	2.0E-2
		Cows	
Consumption of water per animal	m^3/d	6E-2	

Notes: [(a)] dw – dry weight (of soil).
[(b)] fw – fresh weight (of crop) – for further details see Table 84 in Appendix XII of Ref. [13].

TABLE 55. RADIONUCLIDE DOSE COEFFICIENTS FOR INGESTION (REQUIRED FOR TIER 1 TO TIER 4), AND INHALATION AND EXTERNAL IRRADIATION (REQUIRED FOR TIER 1 AND TIER 5)

Radionuclide	**Dose Coefficients for Adults**			
	Ingestion (Sv/Bq)	**Inhalation (Sv/Bq)**	**External Irradiation from Point Source (Sv/h/Bq)**	**External Irradiation from Soil (Sv/h/Bq/m^3)**
Co-60	3.4E-09	1.0E-08	3.5E-13	3.0E-13
Cs-137	1.3E-08	4.6E-09	7.8E-14	6.2E-14
Pb-210	6.9E-07	1.2E-06	5.4E-17	1.5E-16
Po-210	1.2E-06	3.3E-06	1.2E-18	9.5E-19
Ra-226	2.8E-07	3.5E-06	2.4E-13	2.1E-13
Th-229	6.1E-07	8.6E-05	Not required	2.9E-14
Pa-233	8.7E-10	3.9E-09	Not required	1.8E-14
U-233	5.1E-08	3.5E-06	Not required	2.4E-17
Np-237	1.1E-07	2.3E-05	Not required	1.4E-15
Am-241	2.0E-07	4.2E-05	3.0E-15	7.2E-16

APPENDIX III

MATHEMATICAL MODELS FOR TIER 4

This appendix presents the mathematical equations that are used in the Tier 4 screening model of the release and migration of radionuclides from the disposal borehole.

RELEASE FROM THE DISPOSAL CONTAINER

The total inventory is assumed to be contained in one disposal container, which is represented in the model with a compartment (Waste), i.e. with an ordinary differential equation. In calculating the release rate, no distinction is made between the physical and chemical form of radionuclides. Also, no distinction is made between different breaching mechanisms for the disposal container. The release rate (Rel_{rate}, in Bq/y) of radionuclides from this container is given by:

If, $t \geq T_{start}$, then:

$$Rel_{rate} = \left[\frac{1}{\max\left(\frac{T_{duration}}{100}, T_{start} + T_{duration} + t\right)}\right] \cdot Waste \qquad (1)$$

If, $t < T_{start}$, then:

$$Rel_{rate} = 0$$

where

T_{start}	is the start time of radionuclides released from the disposal container (y)
$T_{durtion}$	is the duration of radionuclides released from the disposal container (y)
100	is a scaling factor to ensure the correct numerical solution of the pulse release

Waste (Bq) is defined by:

$$Waste = Initial_{inventory} \cdot Frac_{waste}$$

where:

$Initial_{inventory}$	is the initial inventory of radionuclides in the disposal container (Bq)
$Frac_{waste}$	is the fraction of radionuclides available for release under the different scenarios (unitless)

MIGRATION PROCESSES IN GROUNDWATER

Advective transport through the unsaturated barrier

The advective transfer rate of radionuclides released from the disposal container through the unsaturated barrier compartment due to percolation of water through the unsaturated near field ($\lambda_{Leach,UB}$, in y^{-1}) is given by:

$$\lambda_{Leac\ ,UB} = \frac{q_{Perc,UB}}{\frac{L_{UB}}{N_{UB}} \cdot \varphi_{UB} \cdot R_{UB}} \tag{2}$$

where:

$q_{Perc,UB}$	is the annual percolation rate through the unsaturated barrier compartment (m/y)
L_{UB}	is the length of the unsaturated barrier compartment in the direction of water flow (m)
φ_{UB}	is the water-filled porosity of the unsaturated barrier compartment (unitless)
R_{UB}	is the element-dependent retardation of the unsaturated barrier compartment (unitless)
N_{UB}	is a transport parameter that defines the number of sub-compartments in the unsaturated barrier compartment

φ_{UB} is calculated using the following general formula:

$$\varphi_{UB} = \varepsilon_{UB} \cdot \varphi_{Total,UB} \tag{3}$$

where:

ε_{UB}	is the degree of saturation in the unsaturated barrier compartment (unitless)
$\varphi_{Total,UB}$	is the total porosity of the unsaturated barrier compartment (unitless)

For the purposes of this assessment, it is assumed that all the water-filled pores in the unsaturated barrier compartment contribute to flow and so total porosity and effective porosity have the same values.

R_{UB} is calculated using the following general formula:

$$R_{UB} = 1 + \frac{\rho_{bUB} \cdot Kd_{UB}}{\varphi_{UB}} \tag{4}$$

where:

ρ_{bUB}	is the dry bulk density of the unsaturated barrier compartment (kg/m^3)
Kd_{UB}	is the sorption coefficient of the element in the unsaturated barrier compartment (m^3/kg)
φ_{UB}	is the water-filled porosity of the unsaturated barrier compartment (unitless)

ρ_{bUB} is calculated using the following general formula:

$$\rho_{bUB} = \rho_{gUB}(1 - \varphi_{Total,UB}) \tag{5}$$

where:

ρ_{gUB}	is the grain density of the unsaturated barrier compartment (kg/m^3)

Advective transport through the saturated barrier

The advective transfer rate of radionuclides released from the disposal container through the saturated barrier compartment due to the flow of water through the saturated near field ($\lambda_{LeachSB}$, in y^{-1}) is given by:

$$\lambda_{Leach,SB} = \frac{q_{SB}}{\frac{L_{SB}}{N_{SB}} \cdot \varphi_{SB} \cdot R_{SB}} \quad (6)$$

where:

q_{SB}	is the Darcy velocity through the saturated barrier compartment (m/y)
L_{SB}	is the length of the saturated barrier compartment in the direction of water flow (m)
φ_{SB}	is the water-filled porosity of the saturated barrier compartment (unitless)
R_{SB}	is the element-dependent retardation of the saturated barrier compartment (unitless)
N_{UB}	is a transport parameter that defines the number of sub-compartments in the saturated barrier compartment

φ_{SB} is calculated using the following general formula:

$$\varphi_{SB} = \varepsilon_{SB} \cdot \varphi_{Total,SB} \quad (7)$$

where:

ε_{SB}	is the degree of saturation in the saturated barrier compartment (unitless)
$\varphi_{Total,\ SB}$	is the total porosity of the saturated barrier compartment (unitless)

For the purposes of this assessment, it is assumed that all the water-filled pores in the saturated barrier compartment contribute to flow and so total porosity and effective porosity have the same values.

q_{SB} is calculated by:

$$q_{SB} = -K_{SB} \cdot \frac{\partial H}{\partial x} \quad (8)$$

where:

K_{SB}	is the hydraulic conductivity of the saturated barrier compartment (m/y)
$\partial H/\partial x$	is the hydraulic gradient in the saturated barrier compartment (unitless)

R_{SB} is calculated using the following general formula:

$$R_{SB} = 1 + \frac{\rho_{bSB} \cdot Kd_{SB}}{\varphi_{SB}} \quad (9)$$

where:

ρ_{bSB}	is the dry bulk density of the saturated barrier compartment (kg/m^3)
Kd_{SB}	is the sorption coefficient of the element in the saturated barrier compartment (m^3/kg)
φ_{SB}	is the water-filled porosity of the saturated barrier compartment (unitless)

ρ_{bSB} is calculated using the following general formula:

$$\rho_{bSB} = \rho_{gSB}(1 - \varphi_{Total,SB}) \quad (10)$$

where:

ρ_{gSB}	is the grain density of the saturated barrier compartment (kg/m^3)

Advective transport through the unsaturated zone

The advective transfer rate of radionuclides through the unsaturated zone compartment due to percolation of water through the unsaturated geosphere ($\lambda_{Leach,UZ}$, in y^{-1}) is given by:

$$\lambda_{Leach,UZ} = \frac{q_{Perc,UZ}}{\frac{L_{UZ}}{N_{UZ}} \cdot \varphi_{UZ} \cdot R_{UZ}} \tag{11}$$

where:

$q_{PERC,UZ}$	is the annual percolation rate through the unsaturated zone compartment (m/y)
L_{UZ}	is the length of the unsaturated zone compartment in the direction of water flow (m)
φ_{UZ}	is the water-filled porosity of the unsaturated zone compartment (unitless)
R_{UZ}	is the element-dependent retardation of the unsaturated zone compartment (unitless)
N_{UZ}	is a transport parameter that defines the number of sub-compartments in the unsaturated zone compartment

φ_{UZ} is calculated using the following general formula:

$$\varphi_{UZ} = \varepsilon_{UZ} \cdot \varphi_{Total,UZ} \tag{12}$$

where:

ε_{UZ}	is the degree of saturation in the unsaturated zone compartment (unitless)
$\varphi_{Total,UZ}$	is the total porosity of the unsaturated zone compartment (unitless)

For the purposes of this assessment, it is assumed that all the water-filled pores in the unsaturated zone compartment contribute to flow and so total porosity and effective porosity have the same values.

R_{UZ} is calculated using the following general formula:

$$R_{UZ} = 1 + \frac{\rho_{bUZ} \cdot Kd_{Uz}}{\varphi_{UZ}} \tag{13}$$

where:

ρ_{bUZ}	is the dry bulk density of the unsaturated zone compartment (kg/m^3)
Kd_{UZ}	is the sorption coefficient of the element in the unsaturated zone compartment (m^3/kg)
φ_{UZ}	is the water-filled porosity of the unsaturated zone compartment (unitless)

ρ_{bUZ} is calculated using the following general formula:

$$\rho_{bUZ} = \rho_{gUZ}(1 - \varphi_{Total,UZ}) \tag{14}$$

where:

ρ_{gUZ}	is the grain density of the unsaturated zone compartment (kg/m^3)

Advective transport through the saturated zone

The advective transfer rate of radionuclides through the saturated zone compartment due to the flow of groundwater through the saturated geosphere ($\lambda_{Leach,SZ}$, in y^{-1}) is given by:

$$\lambda_{Leach,SZ} = \frac{q_{SZ}}{\frac{L_{SZ}}{N_{SZ}} \cdot \varphi_{SZ} \cdot R_{SZ}} \tag{15}$$

where:

q_{SZ}	is the Darcy velocity of the groundwater through the saturated zone compartment (m/y)
L_{SZ}	is the length of the saturated zone compartment in the direction of water flow (m)
φ_{SZ}	is the water-filled porosity of the saturated zone compartment (unitless)
R_{SZ}	is the element-dependent retardation of the saturated zone compartment (unitless)
N_{SZ}	is a transport parameter that defines the number of sub-compartments in the saturated zone compartment

φ_{SZ} is calculated using the following general formula:

$$\varphi_{SZ} = \varepsilon_{Sz} \cdot \varphi_{Total,SZ} \tag{16}$$

where:

ε_{SZ}	is the degree of saturation in the saturated zone compartment (unitless)
$\varphi_{Total,\,SZ}$	is the total porosity of the saturated zone compartment (unitless)

For the purposes of this assessment, it is assumed that all the water-filled pores in the saturated zone compartment contribute to flow and so total porosity and effective porosity have the same values. q_{SZ} is calculated by:

$$q_{SZ} = -K_{SZ} \cdot \frac{\partial H}{\partial x} \tag{17}$$

where:

K_{SZ}	is the hydraulic conductivity of the saturated zone compartment (m/y)
$\partial H/\partial x$	is the hydraulic gradient in the saturated zone compartment (unitless)

R_{SZ} is calculated using the following general formula:

$$R_{SZ} = 1 + \frac{\rho_{bSZ} \cdot Kd_{SZ}}{\varphi_{SZ}} \tag{18}$$

where:

ρ_{Bsz}	is the dry bulk density of the saturated zone compartment (kg/m^3)
Kd_{SZ}	is the sorption coefficient of the element in the saturated zone compartment (m^3/kg)
φ_{SZ}	is the water-filled porosity of the saturated zone compartment (unitless)

ρ_{bSZ} is calculated using the following general formula:

$$\rho_{bSZ} = \rho_{gSZ}(1 - \varphi_{Total,SZ}) \tag{19}$$

where:

ρ_{gSZ}	is the grain density of the saturated zone compartment (kg/m^3)

Forward and backward diffusive transport for the unsaturated barrier

The forward and backward diffusive transfer rate of radionuclides released from the disposal container in the unsaturated barrier compartment of the near field ($\lambda_{Diff,UB}$, in y^{-1}) is given by:

$$\lambda_{Diff,UB} = \frac{D_{eff,UB}}{R_{UB}\left(\frac{L_{UB}}{N_{UB}}\right)\cdot\left(\frac{L_{UB}}{N_{UB}}\right)\cdot\varphi_{UB}} \tag{20}$$

where:

$D_{eff,UB}$	is the effective diffusion coefficient for the unsaturated barrier compartment (m^2/y)
L_{UB}	is the length of the unsaturated barrier compartment in the direction of water flow (m)
φ_{UB}	is the water-filled porosity of the unsaturated barrier compartment (unitless)
R_{UB}	is the element-dependent retardation of the unsaturated barrier compartment (unitless)
N_{UB}	is a transport parameter that defines the number of sub-compartments in the unsaturated barrier compartment

The forward diffusive transfer rate of radionuclides released from the disposal container out of the unsaturated barrier compartment of the near field ($\lambda_{Diff,UB,Out}$, in y^{-1}) is very similar and is given by:

$$\lambda_{Diff,UB,Out} = \frac{D_{effUB}}{R_{UB}\frac{\left(\frac{L_{UB}}{N_{UB}}\right)}{2}\cdot\left(\frac{L_{UB}}{N_{UB}}\right)\cdot\varphi_{UB}} \tag{21}$$

Forward and backward diffusive transport for the saturated barrier

The forward and backward diffusive transfer rate of radionuclides released from the disposal container in the saturated barrier compartment of the near field ($\lambda_{Diff,SB}$, in y^{-1}) is given by:

$$\lambda_{Diff,SB} = \frac{D_{eff,SB}}{R_{SB}\left(\frac{L_{SB}}{N_{SB}}\right)\cdot\left(\frac{L_{SB}}{N_{SB}}\right)\cdot\varphi_{SB}} \tag{22}$$

where:

$D_{eff,SB}$	is the effective diffusion coefficient for the saturated barrier compartment (m^2/y)
L_{SB}	is the length of the saturated barrier compartment in the direction of water flow (m)
φ_{SB}	is the water-filled porosity of the saturated barrier compartment (unitless)
R_{SB}	is the element-dependent retardation of the saturated barrier compartment (unitless)
N_{SB}	is a transport parameter that defines the number of sub-compartments in the saturated barrier compartment

The forward diffusive transfer rate of radionuclides released from the disposal container out of the saturated barrier compartment of the near field ($\lambda_{Diff,SB,Out}$, in y^{-1}) is very similar and is given by:

$$\lambda_{Diff,SB,Out} = \frac{D_{eff,SB}}{R_{SB}\frac{\left(\frac{L_{SB}}{N_{SB}}\right)}{2}\cdot\left(\frac{L_{SB}}{N_{SB}}\right)\cdot\varphi_{SB}} \tag{23}$$

Forward and backward diffusive transport for the unsaturated zone

The forward and backward diffusive transfer rate of radionuclides released from the unsaturated barrier compartment in the unsaturated zone compartment of the geosphere ($\lambda_{Diff,UZ}$, in y^{-1}) is given by:

$$\lambda_{Diff,UZ} = \frac{D_{eff,UZ}}{R_{UZ}\left(\frac{L_{UZ}}{N_{UZ}}\right)\cdot\left(\frac{L_{UZ}}{N_{UZ}}\right)\cdot\varphi_{UZ}} \tag{24}$$

where:

$D_{eff,UZ}$	is the effective diffusion coefficient for the unsaturated zone compartment (m^2/y)
L_{UZ}	is the length of the unsaturated zone compartment in the direction of water flow (m)
N_{UZ}	is a transport parameter that defines the number of sub-compartments in the unsaturated zone compartment
φ_{UZ}	is the water-filled porosity of the unsaturated zone compartment (unitless)
R_{UZ}	is the element-dependent retardation of the unsaturated zone compartment (unitless)

The forward diffusive transfer rate of radionuclides released from the unsaturated barrier compartment out of the unsaturated zone compartment of the near field ($\lambda_{Diff,UZ,Out}$, in y^{-1}) is very similar and is given by:

$$\lambda_{Diff,UZ,Out} = \frac{D_{eff,UZ}}{R_{UZ}\frac{\left(\frac{L_{UZ}}{N_{UZ}}\right)}{2}\cdot\left(\frac{L_{UZ}}{N_{UZ}}\right)\cdot\varphi_{UZ}}. \tag{25}$$

Forward and backward diffusive transport for the saturated zone

The forward and backward diffusive transfer rate of radionuclides in the saturated zone compartment of the geosphere ($\lambda_{Diff,SZ}$, in y^{-1}) is given by:

$$\lambda_{Diff,SZ} = \frac{D_{eff,SZ}}{R_{SZ}\left(\frac{L_{SZ}}{N_{SZ}}\right)\cdot\left(\frac{L_{SZ}}{N_{SZ}}\right)\cdot\varphi_{SZ}} \tag{26}$$

where:

$D_{eff,SZ}$	is the effective diffusion coefficient for the saturated zone compartment (m^2/y)
L_{SZ}	is the length of the saturated zone compartment in the direction of water flow (m)
φ_{SZ}	is the water-filled porosity of the saturated zone compartment (unitless)
R_{SZ}	is the element-dependent retardation of the saturated zone compartment (unitless)
N_{sz}	is a transport parameter that defines the number of sub-compartments in the saturated zone compartment

The forward diffusive transfer rate of radionuclides released from the saturated zone compartment out of the saturated zone compartment of the geosphere ($\lambda_{Diff,SZ,Out}$, in y^{-1}) is very similar and is given by:

$$\lambda_{Diff,SZ,Out} = \frac{D_{eff,SZ}}{R_{SZ} \cdot \frac{\left(\frac{L_{SZ}}{N_{SZ}}\right)}{2} \cdot \left(\frac{L_{SZ}}{N_{SZ}}\right) \cdot \varphi_{SZ}} \tag{27}$$

Forward and backward dispersive transport for the unsaturated barrier

The forward and backward dispersive transfer rate of radionuclides released from the disposal container through the unsaturated barrier compartment due to percolation of water through the unsaturated near field ($\lambda_{Disp,UB}$, in y^{-1}) is given by:

$$\lambda_{Disp,UB} = \max\left[\left(\frac{\frac{q_{Perc,UB} \cdot L_{UB}}{Pe - q_{Perc,UB} \cdot \frac{\left(\frac{L_{UB}}{N_{UB}}\right)}{2}}}{R_{UB}\left(\frac{L_{UB}}{N_{UB}}\right) \cdot \left(\frac{L_{UB}}{N_{UB}}\right) \cdot \varphi_{UB}}\right), 0\right] \tag{28}$$

where:

$q_{PERC,UB}$	is the annual percolation rate through the unsaturated barrier compartment (m/y)
L_{UB}	is the length of the unsaturated barrier compartment in the direction of water flow (m)
φ_{UB}	is the water-filled porosity of the unsaturated barrier compartment (unitless)
R_{UB}	is the element-dependent retardation of the unsaturated barrier compartment (unitless)
N_{UB}	is a transport parameter that defines the number of sub-compartments in the unsaturated barrier compartment
Pe	is the Peclet number (unitless)

Forward and backward dispersive transport for the saturated barrier

The forward and backward dispersive transfer rate of radionuclides released from the disposal container through the saturated barrier compartment due to the flow of water through the saturated near field ($\lambda_{Disp,SB}$, in y^{-1}) is given by:

$$\lambda_{Disp,SB} = \max\left[\left(\frac{\frac{q_{SB} \cdot L_{SB}}{Pe - q_{SB} \cdot \frac{\left(\frac{L_{SB}}{N_{SB}}\right)}{2}}}{R_{SB}\left(\frac{L_{SB}}{N_{SB}}\right) \cdot \left(\frac{L_{SB}}{N_{SB}}\right) \cdot \varphi_{SB}}\right), 0\right] \tag{29}$$

where:

q_{SB}	is the Darcy velocity of the groundwater through the saturated barrier compartment (m/y)
L_{SB}	is the length of the saturated barrier compartment in the direction of water flow (m)
φ_{SB}	is the water-filled porosity of the saturated barrier compartment (unitless)
R_{SB}	is the element-dependent retardation of the saturated barrier compartment (unitless)
N_{SB}	is a transport parameter that defines the number of sub-compartments in the saturated barrier compartment
Pe	is the Peclet number (unitless)

Forward and backward dispersive transport for the unsaturated zone

The forward and backward dispersive transfer rate of radionuclides released from the unsaturated barrier compartment through the unsaturated zone compartment due to percolation of water through the unsaturated geosphere ($\lambda_{Disp,UZ}$, in y^{-1}) is given by:

$$\lambda_{Disp,UZ} = \max\left[\left(\frac{\frac{q_{Perc,UZ} \cdot L_{UZ}}{Pe - q_{Perc} \cdot \frac{\left(\frac{L_{UZ}}{N_{UZ}}\right)}{2}}}{R_{UZ}\left(\frac{L_{UZ}}{N_{UZ}}\right) \cdot \left(\frac{L_{UZ}}{N_{UZ}}\right) \cdot \varphi_{UZ}}\right), 0\right] \tag{30}$$

where:

$q_{PERC,UZ}$	is the annual percolation rate through the unsaturated zone compartment (m/y)
L_{UZ}	is the length of the unsaturated zone compartment in the direction of water flow (m)
φ_{UZ}	is the water-filled porosity of the unsaturated zone compartment (unitless)
R_{UZ}	is the element-dependent retardation of the unsaturated zone compartment (unitless)
N_{UZ}	is a transport parameter that defines the number of sub-compartments in the unsaturated zone compartment
Pe	is the Peclet number (unitless)

Forward and backward dispersive transport for the saturated zone

The forward and backward dispersive transfer rate of radionuclides through the saturated geosphere ($\lambda_{Disp,SZ}$, in y^{-1}) is given by:

$$\lambda_{Disp,SZ} = \max\left[\left(\frac{\frac{q_{SZ} \cdot L_{SZ}}{Pe - q_{SZ} \cdot \frac{\left(\frac{L_{SZ}}{N_{SZ}}\right)}{2}}}{R_{SZ}\left(\frac{L_{SZ}}{N_{SZ}}\right) \cdot \left(\frac{L_{SZ}}{N_{SZ}}\right) \cdot \varphi_{SZ}}\right), 0\right] \tag{31}$$

where:

q_{SZ}	is the Darcy velocity of the groundwater through the saturated zone compartment (m/y)
L_{SZ}	is the length of the saturated zone compartment in the direction of water flow (m)
φ_{SZ}	is the water-filled porosity of the saturated zone compartment (unitless)
R_{SZ}	is the element-dependent retardation of the saturated zone compartment (unitless)
N_{SZ}	is a transport parameter that defines the number of sub-compartments in the saturated zone compartment
Pe	is the Peclet number (unitless)

GAS MIGRATION PROCESSES

For calculation of doses to the House Dweller Exposure Group via the inhalation of radioactive gases, the concentration of radionuclides to which the house dweller is exposed via inhalation can be calculated as a linear function of the inventory of radionuclides in the disposal borehole. Therefore, there is no need to represent explicitly the migration of radionuclides in gaseous form between different parts of the disposal system, and so inter-compartmental transfer rates need not be specified.

EXPOSURE MECHANISMS

For the Design Scenario, it is assumed that exposure can only occur once the disposal container has failed, radionuclides are released, and the institutional control period has ended.

The exposure mechanisms identified for the Design Scenario are listed in Table 56. Equations are given below that are used to calculate the annual effective dose received by an average adult member of an exposure group from these exposure mechanisms.

Ingestion of water from an abstraction borehole

The annual individual effective dose to a human from the consumption of drinking water abstracted from a borehole (D_{Wat}, in Sv/y) is given by:

$$D_{Wat} = C_{Wat} \cdot Ing_{Wat} \cdot DC_{Ing} \tag{32}$$

where:

C_{Wat}	is the radionuclide concentration in the abstracted water (Bq/m^3)
Ing_{Wat}	is the individual ingestion rate of water (m^3/y)
DC_{Ing}	is the dose coefficient for ingestion (Sv/Bq)

TABLE 56. EXPOSURE MECHANISMS AND ASSOCIATED EQUATIONS FOR THE DESIGN SCENARIO

Mechanism	Equations
Ingestion of water	(32) to (34)
Inhalation of released gases	(35) to (41)
Ingestion of crops	Ref. [32] as updated by Ref. [33]
Ingestion of milk and meat	Ref. [32] as updated by Ref. [33]

The concentration in the water abstracted from the borehole (C_{Wat}) is a function of the flux of water in the aquifer and the demand for water to be abstracted. This means that C_{Wat} can be calculated using the following conditions:

If $Flux_{aquifer} \geq Demand_{water}$

$$C_{Wat} = C_{Wat,out}$$

If $Flux_{aquifer} < Demand_{water}$

$$C_{Wat} = C_{Wat,out} \cdot \frac{Flux_{aquifer}}{Demand_{water}} \tag{33}$$

where:

$Demand_{water}$ is the water requirements to be abstracted from the borehole (m^3/y)
$C_{Wat,out}$ is the radionuclide concentration in the pore water of the aquifer (Bq/m^3)
$Flux_{aquifer}$ (in m^3/y) represents the mass flux through the aquifer and is determined using the following equation:

$$Flux_{aquifer} = A_{SZ} \cdot q_{SZ} \tag{34}$$

where:

q_{SZ} is the Darcy velocity of the groundwater through the saturated zone compartment (m/y)
A_{SZ} is the cross-sectional area through the aquifer (m^2)

Inhalation of gas

The annual individual effective dose to a human due to inhalation of gases other than Rn-222 (D_{gas}, in Sv/y) is given by:

$$D_{Gas} = BR_{gas} \cdot O_{gas} \cdot C_{gas} \cdot DC_{In} \tag{35}$$

where:

DC_{inh} is the dose coefficient for inhalation of the gaseous form of radionuclide (Sv/Bq)
BR_{gas} is the breathing rate of the human in the house (m^3/h)
O_{gas} is the individual occupancy in the house (h/y)
C_{gas} is the concentration of the gas in the house (Bq/m^3)

C_{gas} is given by:

$$C_{Gas} = \frac{\Phi_g \cdot A_b}{(\lambda_v \cdot V_h)} \tag{36}$$

where:

Φ_g is the flux of the radionuclide into the house ($Bq\ m^{-2}\ y^{-1}$)
A_b is the cross-sectional area of the borehole (m^2)
V_h is the total volume of the house (m^3)
λ_v is the rate of ventilation of the house (y^{-1})

Φ_g is given by:

$$\Phi_g = \frac{I_g \cdot f_g}{(\tau_g \cdot A_b)} \tag{37}$$

where:

I_g is the disposed inventory of the radionuclide, decay-corrected to the start time of the capsule's physical failure (Bq)
τ_g is the timescale over which gas production is assumed to take place following the failure of the capsule (y)
f_g is the total fraction of the inventory of the radionuclide which is assumed to be released as a gas

For Rn-222, the dose coefficient for inhalation of Rn-222 is given in Sv/h per Bq/m^3, which means that:

$$D_{Rn} = O_{gas} \cdot C_{Rnair} \cdot DC_{InhRn} \tag{38}$$

where:

D_{Rn}	is the individual dose from the inhalation of Rn-222 (Sv/y)
DC_{InhRn}	is the dose coefficient for inhalation of Rn-222 (Sv/h per Bq/m^3)
O_{gas}	is the individual occupancy in the house (h/y)
C_{Rnair}	is the concentration of the Rn-222 in the house (Bq/m^3)

C_{Rnair} is given by:

$$C_{Rnair} = \frac{\Phi_{Rn} \cdot A_b}{(\lambda_v + \lambda_{Rn}) V_h} \tag{39}$$

Φ_g	is the flux of Rn-222 into the house (Bq m^{-2} y^{-1})
A_b	is the cross-sectional area of the borehole (m^2)
V_h	is the total volume of the house (m^3)
λ_v	is the rate of ventilation of the house (y^{-1})
λ_{Rn}	is the Rn-222 decay rate (y^{-1})

Φ_{Rn} is given by:

$$\Phi_{Rn} = \chi_{RnBh} \cdot \lambda_{Rn} \cdot \varepsilon_{Rn} \cdot \rho_{Bh} \cdot D_{Bh} \cdot e^{\frac{-d_{gh}}{D_{gh}}} \tag{40}$$

where:

χ_{RaBh}	is the concentration of Ra-226 in the disposal zone (Bq/kg)
ε_{Rn}	is the Rn emanating fraction (unitless)
ρ_{Bh}	is the dry bulk density of the borehole's disposal zone (kg/m^3)
D_{Bh}	is the diffusion length for Rn in the borehole (m)
d_{Bh}	is the thickness of the borehole's closure zone (m)

An Rn-222 emanation fraction of 0.2 is used, while the grain density in the closure zone is 2400 kg/m^3 (used to calculate the dry bulk density). χ_{RaBh} is calculated using:

$$\chi_{RnBh} = \frac{\mathrm{Amount}_{RnBh}}{(\rho_{Bh} \cdot V_{Bh})} \tag{41}$$

where:

$Amount_{RaBh}$	is the amount of Ra-226 in the borehole's disposal zone (Bq)
ρ_{Bh}	is the dry bulk density of the borehole's disposal zone (kg/m^3)
V_{Bh}	is the volume of the borehole's disposal zone (m^3)

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ABBREVIATIONS

BDC	Borehole disposal concept
BDS	Borehole disposal system
C-S-H	Calcium-silicate-hydrate
DSRS	Disused sealed radioactive sources
dw	Dry weight (of soil)
Eh	Standard oxidation–reduction potential
FEPs	Features, events and processes
fw	Fresh weight (of product)
GSA	Generic safety assessment
HQ	Hazard Quotient
K_d	Sorption coefficient
QA	Quality assurance
QC	Quality control
RTG	Radioisotope thermoelectric generator
RWMO	Radioactive Waste Management Organization
SRS	Sealed radioactive sources
SIMBOD	Source Inventory Management for Borehole Disposal
TIC	Total inorganic carbon
WAC	Waste acceptance criteria

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