

Geological Disposal

Generic Operational Environmental Safety Assessment

December 2016



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Preface

Radioactive Waste Management Limited (RWM) has been established as the delivery organisation responsible for the implementation of a safe, sustainable and publicly acceptable programme for the geological disposal of the higher activity radioactive wastes in the UK. As a pioneer of nuclear technology, the UK has accumulated a legacy of higher activity wastes and material from electricity generation, defence activities and other industrial, medical and research activities. Most of this radioactive waste has already arisen and is being stored on an interim basis at nuclear sites across the UK. More will arise in the future from the continued operation and decommissioning of existing facilities and the operation and subsequent decommissioning of future nuclear power stations.

Geological disposal is the UK Government's policy for higher activity radioactive wastes. The principle of geological disposal is to isolate these wastes deep underground inside a suitable rock formation, to ensure that no harmful quantities of radioactivity will reach the surface environment. To achieve this, the wastes will be placed in an engineered underground facility – a geological disposal facility (GDF). The facility design will be based on a multi-barrier concept where natural and man-made barriers work together to isolate and contain the radioactive wastes.

To identify potentially suitable sites where a GDF could be located, the Government has developed a consent-based approach based on working with interested communities that are willing to participate in the siting process. The siting process is on-going and no site has yet been identified for a GDF.

Prior to site identification, RWM is undertaking preparatory studies which consider a number of generic geological host environments and a range of illustrative disposal concepts. As part of this work, RWM maintains a generic Disposal System Safety Case (DSSC). The generic DSSC is an integrated suite of documents which together give confidence that geological disposal can be implemented safely in the UK.

Executive Summary

RWM has been established as the delivery organisation responsible for the implementation of a safe, sustainable and publicly acceptable programme for the geological disposal of the UK's higher activity waste.

As part of this implementation process, RWM has developed a generic Disposal System Safety Case (DSSC), a suite of documents that considers the safety and environmental implications of the geological disposal of radioactive waste. The generic DSSC consists of an Overview, providing context and a summary of the conclusions and key safety arguments; three main documents, one for each of the three components of the overall safety case (the generic Transport Safety Case (TSC), the generic Operational Safety Case (OSC), and the generic Environmental Safety Case (ESC)); and a number of underpinning documents.

This report sets out RWM's current understanding of what an operational environmental safety assessment (OESA) will involve, and supports the generic ESC and the generic OSC. Assessment of operational environmental safety includes the following scope:

- Period: the operational period of the GDF from the start of construction to the surrender of environmental permits at the end of the period of institutional control
- Facilities: environmental safety impacts associated with both surface facilities and underground facilities
- Source: operational practices associated with acceptance of wastes at the GDF site, transfer to the underground, waste emplacement operations, backfilling and sealing operations, and decommissioning, closure and institutional control of facilities
- Pathways: radioactive and non-radioactive discharges in the form of gases, particulates, liquids, solids and radiation (shine), under both routine and accident conditions
- Receptors: effects on members of the public and on non-human biota

This generic OESA focuses on the potential radioactive gaseous emissions from the GDF during the operational period until the completion of all backfilling and sealing activities, and presents an illustrative quantitative assessment, using conservative assumptions, of their potential effects on members of the public and non-human biota. At this early stage of development, non-radioactive discharges, and radioactive solid and liquid discharges, from the GDF are considered qualitatively, noting that such discharges (and any doses arising from them) will be highly site-specific and will be managed on a site-specific basis.

Additional qualitative supporting arguments for the operational environmental safety of the GDF are also presented.

Emissions associated with gases generated in packaged wastes have been estimated. Modelling of discharges from legacy Intermediate Level Waste (ILW), legacy Low Level Waste and nuclear new build unshielded ILW has been undertaken for all significant radioactive gases (C-14-bearing methane, C-14-bearing carbon monoxide, tritium, and Rn-222). Additionally, modelling of discharges from waste in robust shielded containers, nuclear new build shielded ILW, and depleted, natural and low-enriched uranium has been undertaken for Rn-222. Discharges are not expected from packaged High Level Waste, spent fuel, plutonium or highly enriched uranium, as these wastes and materials are assumed to be packaged in high integrity, unvented disposal containers. Potential discharges of naturally occurring Rn-222 from the host rock are also considered.

A gas release base scenario (expected to provide a reasonable generic representation of most aspects of the likely gas generation behaviour) and a bounding case (taking the highest peak release rate for each radionuclide across three variant scenarios) are

considered in a quantitative assessment of dose. In the base scenario, for a higher strength rock, the planning basis is that backfilling of the vaults takes place after all emplacement operations have ceased. In the bounding case, the highest peak release rate for tritium arises from a staged backfilling variant, and the highest peak release of C-14 arises from a higher temperature variant. The peak release rate for Rn-222 is not increased in any of the variant scenarios, so it is the same in both the base scenario and the bounding case.

Illustrative calculations of dose to members of the public from gaseous emissions have been undertaken. The calculations have been done on conservative bases (for example by taking the peak emissions year for each gas under consideration), and show that the dose is expected to be below the legal dose limit for members of the public of 1 mSv per year in both base and bounding scenarios.

Significant uncertainties and conservatisms in the calculations, particularly relating to the modelling of radioactive gas release from the GDF, are identified, and a semi-quantitative discussion of their potential implications is presented. Scoping calculations have been undertaken to investigate the effect on calculated radiological dose of increasing the effective release height and indicate that a significant reduction in radiological doses can be achieved if effective release height is increased. Other potential mitigation measures are also discussed.

A generic illustrative assessment of the potential dose rates to non-human biota, assuming the bounding gas release case, has also been undertaken. The calculated dose rates from gaseous emissions for all the organisms considered are insignificant compared to a predicted no-effect dose rate value for all organisms of 10 μ Gy per hour derived in the European Commission ERICA (Environmental Risk for Ionising Contaminants: Assessment and Management) project. The assessment of dose to non-human biota from radioactive gaseous emissions to the atmosphere from the GDF is therefore concluded to require no further consideration at this stage.

Any actual radiological dose from gaseous emissions to the atmosphere from the GDF will be determined by site-specific factors, and will be a function of actual gaseous discharge rates during each year of GDF operation in combination with local environmental factors and the location and habits of exposed groups. RWM is aware of the need to actively manage gas release in the operational phase of the GDF, through the application of best available techniques and the principle of optimisation, to ensure that regulatory dose constraints are met and doses to the public are minimised.

This report will be updated, in line with updates to the DSSC, as part of each major stage of the GDF development programme. Over time, the design options under consideration and the choices that will have to be made will change from an emphasis on strategy to one on implementation. This approach is consistent with a staged development and approval process.

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1 Introduction

1.1 The generic Disposal System Safety Case

RWM has been established as the delivery organisation responsible for the implementation of a safe, sustainable and publicly acceptable programme for geological disposal of the UK's higher activity radioactive waste. Information on the approach of the UK Government and devolved administrations of Wales and Northern Ireland¹ to implementing geological disposal, and RWM's role in the process, is included in an overview of the generic Disposal System Safety Case (the Overview) [1].

A geological disposal facility (GDF) will be a highly-engineered facility, located deep underground, where the waste will be isolated within a multi-barrier system of engineered and natural barriers designed to prevent the release of harmful quantities of radioactivity and non-radioactive contaminants to the surface environment. To identify potentially suitable sites where a GDF could be located, the Government is developing a consent-based approach based on working with interested communities that are willing to participate in the siting process [2]. Development of the siting process is ongoing and no site has yet been identified for a GDF.

In order to progress the programme for geological disposal while potential disposal sites are being sought, RWM has developed illustrative disposal concepts for three types of host rock. These host rocks are typical of those being considered in other countries, and have been chosen because they represent the range that may need to be addressed when developing a GDF in the UK. The host rocks considered are:

- higher strength rock, for example, granite
- lower strength sedimentary rock, for example, clay
- evaporite rock, for example, halite

The inventory for disposal in the GDF is defined in the Government White Paper on implementing geological disposal [2]. The inventory includes the higher activity radioactive wastes and nuclear materials that could, potentially, be declared as wastes in the future. For the purposes of developing disposal concepts, these wastes have been grouped as follows:

- high heat generating wastes (HHGW): that is, spent fuel from existing and future power stations and High Level Waste (HLW) from spent fuel reprocessing. High fissile activity wastes, that is, plutonium (Pu) and highly enriched uranium (HEU), are also included in this group. These have similar disposal requirements, even though they don't generate significant amounts of heat.
- low heat generating wastes (LHGW): that is, Intermediate Level Waste (ILW) arising from the operation and decommissioning of reactors and other nuclear facilities, together with a small amount of Low Level Waste (LLW) unsuitable for near surface disposal, and stocks of depleted, natural and low-enriched uranium (DNLEU).

RWM has developed six illustrative disposal concepts, comprising separate concepts for HHGW and LHGW for each of the three host rock types. Designs and safety assessments for the GDF are based on these illustrative disposal concepts.

¹ Hereafter, references to Government mean the UK Government including the devolved administrations of Wales and Northern Ireland. Scottish Government policy is that the long term management of higher activity radioactive waste should be in near-surface facilities and that these should be located as near as possible to the site where the waste is produced.

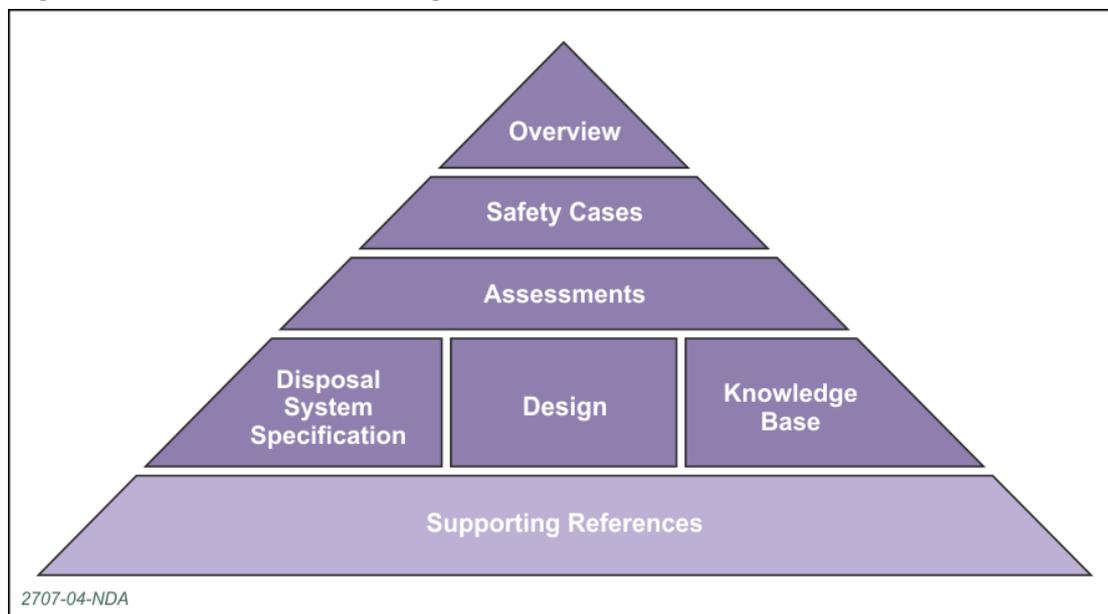
High level information on the inventory for disposal, the illustrative disposal concepts and other aspects of the disposal system is collated in a technical background document (the Technical Background) [3] that supports this generic Disposal System Safety Case.

The generic Disposal System Safety Case (DSSC) plays a key role in the iterative development of a geological disposal system. This iterative development process starts with the identification of the requirements for the disposal system, from which a disposal system specification is developed. Designs, based on the illustrative disposal concepts, are developed to meet these requirements, which are then assessed for safety and environmental impacts. An ongoing programme of research and development informs these activities. Conclusions from the safety and environmental assessments identify where further research is needed, and these advances in understanding feed back into the disposal system specification and facility designs.

The generic DSSC provides a demonstration that geological disposal can be implemented safely. The generic DSSC also forms a benchmark against which RWM provides advice to waste producers on the packaging of wastes for disposal.

Document types that make up the generic DSSC are shown in Figure 1. The Overview provides a point of entry to the suite of DSSC documents and presents an overview of the safety arguments that support geological disposal. The safety cases present the safety arguments for the transportation of radioactive wastes to the GDF, for the operation of the facility, and for long-term safety following facility closure. The assessments support the safety cases and also address non-radiological, health and socio-economic considerations. The disposal system specification, design and knowledge base provide the basis for these assessments. Underpinning these documents is an extensive set of supporting references. A full list of the documents that make up the generic DSSC, together with details of the flow of information between them, is given in the Overview.

Figure 1 Structure of the generic DSSC



1.2 Introduction to the generic Operational Environmental Safety Assessment

This document is the Generic Operational Environmental Safety Assessment (OESA), which supports both the Generic Operational Safety Case (OSC) [4] (together with the

Generic Operational Safety Assessment [5; 6; 7; 8]) and the Generic Environmental Safety Case (ESC) [9] (together with the Generic Post-closure Safety Assessment (PCSA) [10]).

The generic DSSC was previously published in 2010. There are now a number of drivers for updating the safety case as an entire suite of documents, most notably the availability of an updated inventory for disposal [11].

This document updates and replaces the 2010 generic Operational Environmental Safety Assessment published as part of the 2010 generic DSSC suite [12]. This issue includes the following improvements:

- Adoption of the 2013 Derived Inventory
- Improved understanding of the production and consequences of C-14-bearing gases from the carbon-14 integrated project
- Improved understanding of the emanation of Rn-222 from Ra-226 bearing waste packages, including DNLEU
- Consideration of the consequences of the aerial discharges of radioactive gases released by the GDF host rock
- New assessment of doses to non-human biota from C-14 and tritium, using an improved version of the ERICA (Environmental Risk from Ionising Contaminants: Assessment and Management) assessment tool
- Application of updated screening values for dose rates to non-human biota

1.3 Objective

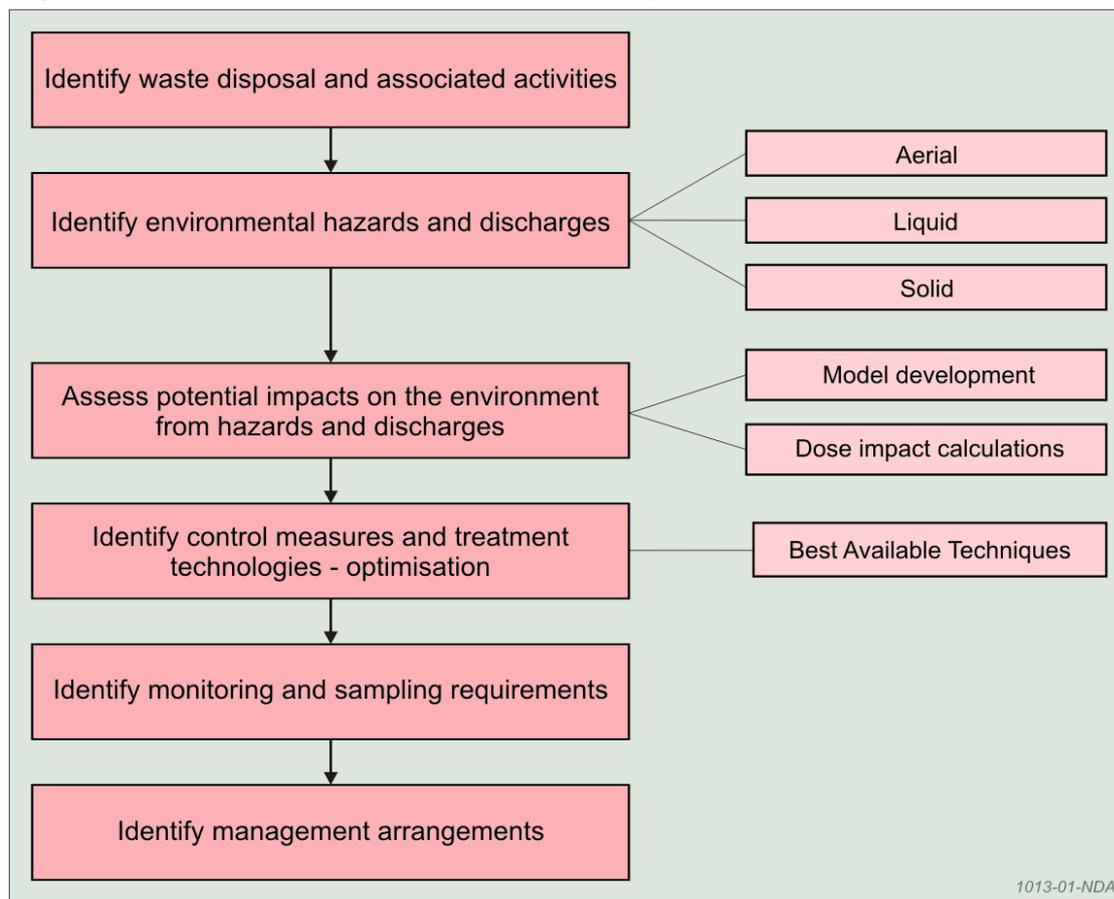
As part of the generic DSSC, this generic OESA addresses the environmental safety of potential radioactive discharges and non-radioactive (for example, chemically toxic to humans and/or the environment) discharges arising from the operational phase of the GDF until the completion of all backfilling and sealing activities. It supports the generic ESC and the generic OSC. It considers the radiological impact to members of the public and to non-human biota². The report is intended to cover aspects required to be considered under the Environmental Permitting Regulations 2010 [13] and the Ionising Radiations Regulations 1999 [14].

The approach to production of an OESA is shown in Figure 2. The process shown applies to potential radioactive and non-radioactive discharges to the aerial, aquatic and terrestrial environments. Although the control measures associated with the GDF will largely be based on passive features, operational functions will require active safety features. This combination of passive and active features is addressed within an OESA to demonstrate an appropriate level of overall environmental safety for the operational period of the GDF.

The primary purpose of this generic OESA is to provide an illustrative quantitative indication of off-site doses associated with the operational phase of the GDF. These doses are compared with relevant criteria set out in the RWM Radiological Protection Criteria Manual (RPCM), and will facilitate the development of an understanding of the importance of various factors that lead to the derived results (including data and assumptions).

This generic OESA builds upon and develops the off-site dose assessments presented in the earlier safety case (published in 2003) referred to as the Generic Operational Safety Assessment (GOSA) [15], and the 2010 generic OESA [12].

² The term 'non-human biota' refers to wildlife populations of flora and fauna.

Figure 2 Operational environmental safety assessment process

This process has only been applied in outline in the generic DSSC, in a manner appropriate to the generic stage of the GDF implementation programme.

1.4 Scope

An OESA brings together all currently available information that is relevant to the assessment of environmental safety during the operational period of the GDF. Assessment of operational environmental safety includes the following scope:

- **Period:** the operational period of the GDF from the start of construction to the surrender of environmental permits at the end of the period of institutional control
- **Facilities:** environmental safety impacts associated with both surface facilities and underground facilities
- **Source:** operational practices associated with acceptance of wastes at the GDF site, transfer to the underground, waste emplacement operations, backfilling and sealing operations, and decommissioning, closure and institutional control of facilities
- **Pathways:** radioactive and non-radioactive discharges in the form of gases, particulates, liquids, solids and radiation (shine), under both routine and accident conditions
- **Receptors:** Effects on members of the public and on non-human biota

However, this scope overlaps in part with that of other reports within the generic DSSC suite. Rather than duplicate information, this report addresses only relevant considerations

that are not included in any other report; that is, the off-site impact of routine discharges from the site, excluding those from direct radiation shine and those during the GDF construction phase prior to the arrival of waste. The reports in which other relevant aspects are discussed are identified below:

- The hazards during the construction period, including non-radiological safety, are discussed in Volume 1 of the Generic Operational Safety Assessment [5]
- The hazard of off-site doses received as a result of direct radiation shine from sources on the site is addressed in Volume 2 of the Generic Operational Safety Assessment [6]
- Fault conditions resulting in consequences off site are considered in Volume 3 of the Generic Operational Safety Assessment [7]
- Potential gas releases from emplaced wastes between final sealing and closure of the GDF and the end of institutional control are considered in the Generic Post-closure Safety Assessment [10]

1.5 Document structure

The remainder of this report is summarised, by section, below.

- Section 2 (Basis for Assessment) briefly sets out the basis for the assessment presented in this report, in terms of strategy, inventory for disposal, discharge, and operations
- Section 3 (Safety) sets out RWM's policy and the principles against which the results of this assessment are judged
- Section 4 (Derivation of Off-Site Discharges) describes how the off-site discharges have been derived based on the wastes and activities assessed
- Section 5 (Methodology for Assessing Doses from Gaseous Emissions to the Atmosphere) describes the methodology for calculating off-site doses based on the discharges from the site derived in section 4, including a description of the illustrative exposure scenarios assumed for representative members of the public
- Section 6 (Quantitative Assessment Results) presents calculated doses to representative members of the public and non-human biota and compares these doses with applicable design criteria and safety objectives
- Section 7 (Safety Analysis and Discussion) discusses the results in context with the design criteria and safety objectives; uncertainties and conservatism in the assessment and potential mitigation measures that could be employed are discussed, and qualitative safety arguments are presented
- Section 8 (Future Work) describes the future work in this area that is either ongoing or planned to address issues raised by this report
- Section 9 (Summary) summarises the assessments undertaken, the results and conclusions with respect to the operational environmental safety of the GDF

Common terms and acronyms used throughout the generic DSSC are defined in the glossary in the Technical Background. A document-specific glossary at the end of this report provides definitions of key technical terms and acronyms used in this report but not used widely elsewhere in the generic DSSC.

2 Basis for Assessment

The Environmental Permitting Regulations [13] refer to the disposal of radioactive waste where 'disposal' is defined as referring to discharges whether into water, air, sewer, drain or otherwise³. To avoid confusion with the disposal of waste within the facility, in this report the terms 'discharges', 'emissions' and 'releases' are used to refer to disposals off site. This generic OESA therefore considers the radiological exposure of individual members of the public resulting from off-site discharges during the operational period of the GDF. Radiological exposure of non-human biota populations is also considered, as are non-radioactive discharges.

The operational period of the GDF is considered to begin when construction of the facility starts, and to end with the surrender of environmental permits at the end of the period of institutional control. During the operational period it is anticipated that construction activities might take place at the same time as waste emplacement, and, possibly, closure of parts of the facility that have already been filled with waste.

This generic OESA does not present an assessment of discharges that could occur during the GDF construction phase prior to waste emplacement (for example drilling liquids or leachates associated with spoil heaps), although it is noted that this will be needed as the GDF implementation process proceeds and the site-specific phase begins. Work supporting the generic OSC, for example the Generic Operational Safety Case – Volume 1 (Construction and non-radiological safety assessment) [5], specifically looks at the hazards associated with activities during the construction period and demonstrates awareness of the issues for future consideration.

Illustrative geological disposal concepts for a GDF constructed in a higher strength host rock, in a lower strength sedimentary host rock, and in an evaporite host rock are described in the Technical Background [3]. Although the inventory for disposal will be the same for each of these disposal concepts, GDF operations will need to be adapted to host rock properties. For example, for planning purposes, a notional period of 10 years has been assumed for the backfilling of access tunnels, sealing and closure of the GDF, irrespective of the host rock type. However, for a GDF in a higher strength rock, backfilling of the vaults is assumed to occur once all emplacement operations have ceased, whereas for a GDF in a lower strength sedimentary rock, vault backfilling is assumed to be undertaken as each vault is filled. For an evaporite host rock, vaults are assumed to be 'closed' by natural creep processes and a specific backfilling step is assumed not to be required. Such differing approaches to the way the GDF is operated dependent on host rock could affect the rate of discharges to be assessed in an OESA. GDF operations may potentially also need to be adapted in consideration of stakeholder input.

2.1 OESA strategy

The strategy underlying the development of the three strands of this generic OESA – qualitative assessment, management of uncertainty, and quantitative assessment – is described below.

2.1.1 OESA: qualitative assessment

RWM's understanding of the geological disposal system during the operational period comes from a variety of sources:

³ 'Disposal' in relation to waste includes its removal, deposit, destruction, discharge (whether into water or into the air or into a sewer or drain or otherwise) or burial (whether underground or otherwise) and 'dispose of' is to be construed accordingly.

- Key inputs come from RWM's own research programme and from previous research work by RWM's predecessors, NDA RWMD and Nirex. As the programme moves forward, RWM expects to develop further understanding relevant to operational environmental safety, for example with respect to waste package behaviour during the operational period, the host rock, and the transport of potential radioactive and non-radioactive contaminants at the particular site(s) under consideration.
- RWM contributed to the UK National Dose Assessment Working Group, a consortium of people and organisations with responsibility for, or an interest in, the assessment of radiation doses to the public from the operation of nuclear facilities, which produced a number of guidance reports on the topic of dose assessment.
- RWM participates in several national and international initiatives concerned with reducing uncertainties in environmental modelling [16]. For example, it is/was a member of the following initiatives:
 - The BIOPROTA forum, which was established in 2002 to address uncertainties in assessments of contaminant releases into the environment arising from radioactive waste disposal. The project focuses on biosphere migration and accumulation mechanisms relevant to key radionuclides (for example C-14), and is relevant to both the OESA and PCSA.
 - The International Atomic Energy Agency (IAEA)'s MODARIA programme, which started in 2012 and aimed to improve capabilities in the field of environmental radiation dose assessment by means of acquisition of improved data for model testing, model testing and comparison, reaching consensus on modelling philosophies, approaches and parameter values, development of improved methods and exchange of information. MODARIA builds on the work of the previous IAEA Environmental Modelling for Radiation Safety (EMRAS) projects I and II.
 - The NERC-TREE project (part of the UK Radioactivity and the Environment programme), which is running from 2013 to 2018 and aims to reduce uncertainty in estimating the risk to humans and wildlife (non-human biota) associated with exposure to radioactivity and to reduce unnecessary conservatism in risk calculations.

In section 7.2, this understanding is used to summarise qualitative supporting arguments for operational environmental safety of the GDF, to discuss the potential for off-site release of contaminants from the GDF and public exposure to such discharges, and to provide examples of experience in the UK and overseas of providing operational environmental safety for relevant radioactive waste management facilities. The discussion in section 7.2 is at a level of detail appropriate to this generic stage of the GDF implementation programme, and is supported by RWM's research status reports.

The qualitative assessment cannot be used to identify absolute magnitudes of public exposure, but may be used to justify the inclusion of components in the quantitative assessment, or the exclusion of components considered to be of minor significance.

2.1.2 OESA: management of uncertainty

Aspects of uncertainty that are important to the OESA and that need to be managed include:

- Uncertainties in developing a representative conceptual model of the system for undertaking assessment calculations
- Uncertainties in estimates of contaminant concentration and their distribution in the GDF, including in the inventory for disposal

- Uncertainties in detailed design and operational management of the GDF and the resultant conditions in the GDF
- Uncertainties in contaminant generation mechanisms and migration routes within and beyond the GDF
- Uncertainties in calculating potential consequences to exposed individuals and other receptors

Some of these uncertainties are site-specific while others are site-independent. Uncertainties of both types, and their management, are identified in more detail in the relevant sections of this report, and are brought together and discussed in the overall context of the OESA in section 7.1.1.

2.1.3 OESA: quantitative assessment

The conceptual models that underpin an OESA need to consider the most important features of the waste, processes that could potentially lead to releases of radioactivity and other contaminants during operation of the GDF, routes by which any released contaminants could impact the environment, and the points at which impacts may be assessed. Such models can be used to evaluate the relative magnitude of source terms, the most likely routes for entry to the environment, and receptors of particular importance for any specified activity or site.

Relevant characteristics for contaminant releases need to be established. For example, for discharges to the atmosphere, this includes the height of the release, local meteorological conditions and dispersion characteristics, and the distance to the site boundary. The relevant characteristics of the receptors (members of the public and non-human biota) also need to be established.

It is not possible to state the particular computational tools that will be used in later stages of the GDF implementation process to undertake impact assessments as such tools are being continually developed and updated owing to their widespread use within and outside the nuclear industry. RWM intends to use industry-standard tools that are available at the time each update of the OESA is conducted.

RWM has produced a Data Report [17], which documents the data underpinning the generic DSSC, including the OESA. Data used in this OESA are presented in this report for convenience and transparency, and references are also provided to the relevant section of the Data Report where appropriate.

2.1.4 Using the OESA

An OESA needs to consider potential discharges and associated measures of exposure and environmental impact associated with the operational phase of the GDF.

At this generic stage of GDF implementation, the OESA presents RWM's current understanding of operational environmental safety, identifies requirements for further information once specific sites and GDF designs become available, and presents illustrative calculations that allow comparison of potential impacts against relevant standards. These calculations focus on the potential off-site impacts of gaseous releases from emplaced wastes.

2.2 Inventory

This generic OESA presents an assessment of operational discharges associated with all wastes and materials covered by the 2013 Derived Inventory [11]⁴, as described in the Technical Background, which is used throughout the DSSC documentation suite as the underpinning dataset.

Uncertainties in the inventory for disposal are managed in the generic DSSC through consideration of a number of inventory scenarios, which are summarised in the Technical Background. The generic DSSC adopts a modular approach, also summarised in the Technical Background, which enables analysis of the impact (for example on assessment results) of different waste types not being included in the inventory for disposal, of the inventory increasing or decreasing, or of the nature of the inventory changing. In this generic OESA, contributions of different waste types to the assessments (the modular approach) and the effects of inventory scenarios are discussed in largely qualitative fashion in sections 4.5 and 7.1.1 respectively.

As the OESA is developed further, RWM will consider the impact of any update to the Derived Inventory on the calculations. If appropriate, work will be commissioned to qualitatively discuss this impact (for example as an addendum to the OESA) or to update the calculations to confirm that the results are not significantly different.

In addition to the emplaced radioactive inventory, the host rock may also be a source of naturally occurring radon; this is discussed in section 4.

2.3 Operations, including closure

As described in the Technical Background, different illustrative disposal concepts have been defined for three potential host rock types. These concepts assume co-location of LHGW and HHGW (including high fissile activity wastes) in a single facility, separated by a suitable respect distance. The illustrative geological disposal concepts comprise a number of basic elements including:

- **Surface facilities:**
 - Construction area facilities
 - Package receipt facilities
 - Decontamination, inspection and maintenance facilities for empty transport containers
 - Remote handling facilities to allow minor repairs to be undertaken on any incoming transport containers failing acceptance tests
 - Active effluent treatment plant and active ventilation plant
 - Active laundry and laboratories
- **Underground facilities:**
 - Underground access is assumed to be provided by a drift tunnel and/or shafts, dependent on the host rock

⁴ Previously, RWM published an Upper Inventory as part of the 2010 Derived Inventory, which included wastes from a nuclear new build programme and allowed it to consider inventory uncertainty. Following confirmation of the nuclear new build programme in the 2014 White Paper 'Implementing Geological Disposal' [2], the 2013 Derived Inventory includes these wastes and there is no separate 2013 Upper Inventory. Uncertainties in the inventory are now managed through the use of inventory scenarios.

- LHGW disposal area: facilities will be needed to manage a variety of LHGW packages. Some will be removed from transport containers while others will be disposed of directly. Inspection and monitoring activities will also be undertaken.
- HHGW disposal area: each HLW/SF waste container is assumed to be taken underground in its transport container. The waste container will then be removed from the transport container in a reception area and then transported to the disposal tunnel for disposal.
- Waste package maintenance facility: if the GDF is constructed in higher strength rock, the possibility of keeping the disposal areas open (for example, to allow an extended period of retrievability) is considered. In such a case, a facility may be required to remediate waste package degradation.

Normal operations associated with the facilities listed above and considered in this assessment include:

- The receipt of waste packages arriving by road and rail
- Transfer of waste packages from delivery vehicle to drift wagon or via the shaft where there is no drift (including monitoring and inspection activities)
- Transfer of waste packages underground
- Transfer to and emplacement of waste packages in the appropriate disposal vault or tunnel
- Decontamination (where necessary) of transport containers for re-use
- Inspection and maintenance activities
- Backfilling, sealing and closure

Furthermore:

- The GDF will be ventilated during normal operations. The ventilation system will be designed to protect both workers and the public by filtering out particulate contaminants before they reach the surface, such that any releases to the atmosphere are within permitted levels.
- Depending on the hydrogeological setting, there may be groundwater intrusion into the active areas during the operational period; this will be managed and kept to a minimum level. Any liquid ingress will be diverted away from waste packages and collected to minimise the potential for corrosion or contamination.

The elements of the disposal facility are described further in section 4, including how activities associated with each part of the site may give rise to discharges off site.

2.4 Discharges

Solid, liquid and gaseous radioactive discharges potentially arising from the routine operations described in section 2.3 are considered within the assessment. The source of these discharges and their management are considered further in section 4. The radiological impacts to members of the public and non-human biota, as a result of these discharges, are presented in section 6 of this report.

The wastes assigned to the GDF may also present hazards associated with non-radioactive discharges. These discharges are discussed in section 4. There are many other non-radioactive discharges that could be associated with the GDF, for example groundwater discharges during construction. These are discussed briefly in section 4 and will be considered fully as part of the assessment supporting an application for an

environmental permit and as part of an environmental impact assessment (EIA), such as will be conducted at later stages in the implementation process.

Strategic environmental assessment (SEA) and EIA form part of the environmental assessments that are undertaken at the appropriate stages of the GDF programme in compliance with the relevant directives [18]. These assessments help to identify the potential environmental, social and economic impacts of implementing the GDF.

The waste storage and packaging requirements, described in the Technical Background, ensure that both radioactive and non-radioactive discharges associated directly with the wastes are appropriately minimised. This is discussed further in section 4.

Note that this generic OESA does not assess any discharges that may arise as part of either package refurbishment or disposal area refurbishment. The need for such refurbishment operations will be determined later in the implementation programme and the implications of any discharges that could arise as a result of refurbishment will be assessed.

There will also be solid non-radioactive wastes generated from decommissioning operations (for example building rubble, non-radioactive scrap). These will be considered in a future version of the OESA.

3 Safety Objectives

RWM has developed internal policy documents that describe how UK regulations and regulatory guidance and international recommendations and guidance are interpreted. Particularly relevant to the OESA are the Design Principles [19] and the RPCM. The generic Disposal System Specification (DSS) [20; 21] specifies that these internal policies shall be addressed in the design of the disposal system.

3.1 Geological disposal facility design principles

The GDF Design Principles define a consistent set of principles that apply to all elements of the GDF and associated waste transport systems. Their purpose is to provide the foundation upon which the designs, and design assessments, are to be developed. In addressing the high level requirements for radiological safety, the GDF design principles also embody the initial high level safety functional requirements (SFRs) for the facility. SFRs are defined in RWM's Engineering Design Manual, and are developed in more detail in the Nuclear Operational Safety Manual. Together these manuals define the process for management of SFRs. The SFRs will be subject to review as part of design development.

It should be noted that the application of the design principles for the purpose of this assessment is limited because the assessment is based on illustrative geological disposal concepts and related designs. This assessment will provide input to the development of site-specific designs and of the SFRs as part of the iterative design process described in the DSS.

3.2 Radiological Protection Criteria Manual

The RPCM sets out the radiological protection approach and criteria within and against which all work undertaken by RWM directed towards the implementation of geological disposal is to be undertaken. The policy and criteria presented in the RPCM are consistent with statutory radiological protection requirements and applicable guidance, and have been adhered to in undertaking this assessment. Conformance with the provisions of the RPCM will assist the process of ensuring that the packaging and transport of waste, the design of geological disposal facilities, the conduct of operations and the eventual closure of those facilities will meet all current statutory radiological protection requirements.

3.2.1 Statutory basis

The criteria relevant to an OESA (presented in Table 1 in section 3.2.3) arise directly or indirectly from the following legislation:

- Health and Safety at Work etc Act 1974 [22]
- The Management of Health and Safety at Work Regulations 1999 (SI 1999/3242) [23]
- Nuclear Installations Act 1965 [24]
- The Ionising Radiations Regulations 1999 (SI 1999/3232) [14]
- The Environmental Permitting (England and Wales) Regulations 2010 (SI 2010/675) [13]
- Article 37 of The Treaty Establishing the European Atomic Energy Community (Euratom) [25]

With the exception of the Nuclear Installations Act 1965 (which applies in Northern Ireland, as well as Great Britain), equivalent statutory requirements are set down in legislation specific to Northern Ireland.

3.2.2 Regulatory guidance and other material

In addition to the legislation listed above, policy, guidance and other material has been issued that is of assistance to designers and operators of nuclear facilities. Key documents noted in the RPCM and relevant to this assessment are listed below. RWM is also aware of other supporting relevant guidance documents not listed, and these will be considered at an appropriate stage in the implementation of geological disposal.

Council Directive 96/29/Euratom of 13 May 1996 laying down basic safety standards for protection of the health of workers and the general public against the dangers arising from ionising radiation, 1996 [26].

The Treaty Establishing the European Atomic Energy Community ('the Euratom Treaty') requires that basic standards be laid down within the Euratom Community for the protection of the health of workers and the general public against the dangers arising from ionising radiations. Council Directive 96/29/Euratom established the required basic safety standards, taking into account the latest developments in scientific knowledge concerning radiological protection at that time. The Directive has been given effect in England and Wales through (among other things) the Ionising Radiations Regulations 1999 and the Environmental Permitting (England and Wales) Regulations 2010; and in Northern Ireland through (among other things) the Ionising Radiations Regulations (Northern Ireland) 2000 and the Radioactive Substances (Basic Safety Standards) Regulations (Northern Ireland) 2003. This national legislation must be interpreted, as far as possible, in light of the wording and purpose of Council Directive 96/29/Euratom in order to achieve an outcome consistent with the objective(s) pursued by it.

Council Directive 96/29/Euratom will be repealed by the new basic safety standards Directive (Council Directive 2013/59/Euratom – see below) in February 2018.

Council Directive 2013/59/Euratom of 5 December 2013 laying down basic safety standards for protection against the dangers arising from exposure to ionising radiation, 2013 [27].

Council Directive 2013/59/Euratom establishes the most recent revision of the basic standards required by the Euratom Treaty, taking into account the latest recommendations of the International Commission on Radiological Protection (ICRP), in particular those in ICRP103 (see below), and new scientific evidence and operational experience. It will replace the previous basic safety standards Directive (96/29/Euratom – see above) as well as the Directives that supplement it in more specific areas (Council Directives 89/618/Euratom, 90/641/Euratom and 2003/122/Euratom, among others). Among other things, it introduces a new (lower) limit for equivalent dose for the lens of the eye in occupational exposure and specific requirements for the control of naturally occurring radon in workplaces.

Member States have until 6 February 2018 to bring into force the laws, regulations and/or administrative provisions necessary to give effect to Council Directive 2013/59/Euratom. These must be interpreted, as far as possible, in light of the wording and purpose of the Directive in order to achieve an outcome consistent with the objective(s) pursued by it.

ICRP103, Recommendations of the ICRP, 2007 [28]

The ICRP provides recommendations and guidance on protection against ionising radiation. As ICRP's most recent recommendations, this publication explains how the ICRP's fundamental principles of radiological protection (justification, optimisation of protection and application of dose limits) apply to three controllable exposure situations: planned, emergency and existing exposure situations. The recommendations have been used to inform the development of the most recent basic safety standards Directive under the Euratom Treaty (Council Directive 2013/59/Euratom – see above).

Application of the 2007 Recommendations of the ICRP to the UK, Advice from the Health Protection Agency, 2009 [29]

The Health Protection Agency (HPA) is responsible for advising government departments on radiological protection. The HPA, now part of Public Health England, has welcomed the ICRP recommendations, which represent an update, consolidation and development of the previous recommendations. Overall HPA concluded that the revised recommendations do not imply the need for any major changes to the system of protection applied in the UK.

Safety Assessment Principles (SAPs) for Nuclear Facilities, 2014 Edition, Rev 0 [30]

The Office for Nuclear Regulation (ONR) applies its Safety Assessment Principles (SAPs) to assessments of safety at existing or proposed nuclear facilities, usually through its assessment of safety cases in support of regulatory decisions. The principles presented in the SAPs relate only to nuclear safety, radiological protection and radioactive waste management. They do not deal with conventional hazards associated with a nuclear facility, except where they have a direct effect on nuclear safety or radioactive waste management.

The primary purpose of the SAPs is to provide inspectors with a framework for making consistent regulatory judgements on the safety of activities. The principles are supported by Technical Assessment Guides (TAGs), and other guidance, to further assist decision making within the nuclear safety regulatory process. Although it is not their prime purpose, the SAPs may also provide guidance to designers and duty-holders on the appropriate content of safety cases, clarifying ONR's expectations in this regard. However, they are not sufficient on their own to be used as design or operational standards. Although in most cases the SAPs provide guidance, in those places where they refer to legal requirements they may be mandatory, depending on the circumstances.

Statutory Guidance to the Environment Agency concerning the regulation of radioactive discharges into the environment, 2009 [31]

This guidance is provided by the Secretary of State for Energy and Climate Change and the Secretary of State for Health in relation to England, and the Welsh Ministers in relation to Wales, on how the Environment Agency should implement the UK Strategy for Radioactive Discharges. The guidance considers the Environment Agency's Radioactive Substances Regulation Environmental Principles (see below) a suitable underpinning to the Statutory Guidance.

Radioactive Substances Regulation (RSR) – Environmental Principles, 2010 [32]

This document has been produced by the Environment Agency to provide a standardised framework and technical guidance for the assessments and judgements made by the Environment Agency Radioactive Substances Regulators. The document provides an overall hierarchy and topic framework for the principles, an objective for radioactive substances regulation, fundamental principles, and generic developed principles. The document explains that specific principles for facilities for the disposal of solid radioactive wastes are given in the Guidance for Requirements on Authorisation (GRA; see below) and take account of relevant RSR Environmental Principles.

Geological Disposal Facilities on Land for Solid Radioactive Wastes: Guidance on Requirements for Authorisation, 2009 (GRA) [33]

The geological disposal GRA was issued by the Environment Agency and the Northern Ireland Environment Agency. The Radioactive Substances Act 1993 (RSA 93) gave the environment agencies legal powers and duties to authorise the disposal of radioactive waste in the UK. The GRA is aimed principally at the developers of proposed geological disposal facilities for radioactive waste. It explains the requirements that a developer or operator is expected to fulfil when applying for an authorisation to develop or operate such a facility. The guidance sets out radiological protection requirements and explains the

regulatory process that leads to a decision on whether to authorise radioactive waste disposal.

Relevant aspects of RSA 93 have been replaced in England and Wales by the Environmental Permitting Regime [13]. The 2010 Regulations include provisions broadly similar to those of RSA 93, but within a common framework for environmental permitting across different regulatory regimes.

Since the environment agencies' publication of the GRA in 2009, the ICRP and IAEA have both published guidance on disposal of solid radioactive waste. The advice and recommendations in this international guidance do not impose legal obligations on designers or operators of such activities in the UK, but UK regulators may take them into account when making regulatory decisions or carrying out other regulatory activities. It is anticipated that any future revision of the GRA is likely to take account of relevant aspects of this international guidance. The Annex to the RPCM provides a summary of the most relevant recommendations provided in the following international guidance:

- ICRP, 2013, Radiological Protection in Geological Disposal of Long-lived Solid Radioactive Waste. ICRP Publication 122. Ann. ICRP 42(3)
- IAEA, 2011, IAEA Safety Standards for Protecting People and the Environment – Disposal of Radioactive Waste. IAEA Specific Safety Requirements No. SSR-5
- IAEA, 2011, IAEA Safety Standards for Protecting People and the Environment – Geological Disposal Facilities for Radioactive Waste. IAEA Specific Safety Guide No. SSG-14

Geological Disposal Facilities on Land for Solid Radioactive Wastes: Guidance on Requirements for Authorisation – Supplementary guidance related to the implementation of the Groundwater Directive, 2012 [34].

In 2012, the Environment Agency issued supplementary guidance related to the implementation of the Groundwater Directive (Directive 2006/118/EC), which it intends to be read in the overall context of the GRA (it does not replace or supersede any aspect of the GRA). Relevant aspects of this supplementary guidance to the GRA (the dose guidance level for the groundwater pathway during the period of authorisation and the risk guidance level for the groundwater pathway after the period of authorisation) have been included in the RPCM.

Principles for the Assessment of Prospective Public Doses arising from Authorised Discharges of Radioactive Waste to the Environment, 2012 [35]

The UK's environment agencies, the Food Standards Agency and the Health Protection Agency (now Public Health England) produced a set of principles and associated guidance on the assessment of public doses for the purpose of authorising discharges of gaseous and aqueous radioactive waste to the environment. The document is intended to enable radiological assessments to be produced by applicants for environmental permits, and assessed by environment agency staff involved in the determination of such applications, in a more consistent and transparent manner. The principles and guidance do not apply to the assessment of the impact of disposals of solid radioactive waste (see the GRA).

Radiological Protection Objectives for the Land-based Disposal of Solid Radioactive Wastes, Advice from the Health Protection Agency, 2009 [36]

The HPA/Public Health England is responsible for advising government departments on radiological protection criteria to be applied to the disposal of all types of solid radioactive waste. The advice is intended to be applied at the planning stages for a disposal facility. The advice includes recommendations for the protection of the public, recommendations on the assessment of doses and radiological risks and on optimisation of radiological protection.

3.2.3 Relevant criteria

For the OESA, the relevant criteria against which the results of this assessment will be compared are presented in Table 1.

Table 1 Dose limits for normal operation to be applied during the period of licensing and/or environmental permitting

Limit	Dose to any person other than an employee or trainee, mSv per year
Legal dose limit ⁽¹⁾	1
Source-related dose constraint for members of the public for a new disposal facility ⁽²⁾	0.15
Dose guidance level (groundwater pathway) ⁽³⁾	0.02
Basic Safety Objective ^(4,5)	0.02

1. The legal effective dose limits are values of effective or equivalent dose that must not be exceeded for an individual (Regulation 11 and Schedule 4 of the Ionising Radiations Regulations 1999 [14], and paragraph 1 of Part 4 of Schedule 23 of the Environmental Permitting (England and Wales) Regulations 2010 [13]). They apply to the sum of the relevant doses* from external exposures in the year and the 50-year committed doses (up to age 70 for children)** from intakes in the same year. There are other dose limits and restrictions on exposure applicable to pregnant women, breast feeding mothers and women of reproductive capacity (see Ionising Radiations Regulations 1999 regulation 8(5) and Schedule 4).

* 'relevant doses' means the effective and/or equivalent doses from all planned exposure situations (including past planned exposure situations that resulting in the presence of radionuclides in the environment) giving rise to exposure of the individual(s) in question, but excluding medical exposures of patients.

** 'committed dose' means the committed effective dose or committed equivalent dose (depending on whether the legal dose limit is an effective dose limit or an equivalent dose limit). The 'committed effective dose' is the sum of the committed organ or tissue equivalent doses resulting from an intake, each multiplied by the appropriate tissue weighting factor. The 'committed equivalent dose' is the integral over time (50 years for adults and up to the age of 70 for infants and children) of the equivalent dose rate in tissue or organ that will be received by an individual as a result of an intake.

2. The Environmental Permitting (England and Wales) Regulations 2010 [13] require the regulators, when determining applications for environmental permits, to have regard to the source-related dose constraint of 0.3 mSv/y in relation to each source from which radioactive discharges will be made. A source is a facility, or group of facilities, which can be optimised as an integral whole (that is, considered as one source) in terms of radioactive waste disposals. The doses to be compared with the source-related dose constraint are only those that can be altered by changes in the operating regime of the source; that is, the sum of the effective doses from planned discharges and direct radiation from the source. Doses resulting from exposure to historical discharges should not be included. The GRA [33] notes that for the operational and active institutional control phases, the Health Protection Agency (HPA), now part of Public Health England, has recommended that a dose constraint

of 0.15 mSv (annual dose) should apply to exposure to the public from a new disposal facility for radioactive waste. HPA's recommendation should be taken into account as well as directions from the UK Government and Devolved Administrations. In the RPCM, RWM has adopted the lower constraint recommended by the HPA.

3. The radiation dose to members of the public through the groundwater pathway during the period of authorisation of the facility should be consistent with, or lower than, the dose guidance level (the dose standard against which the radiological contamination of the groundwater pathway during the Period of authorisation is assessed). The dose guidance level indicates the standard expected but does not suggest that there is an absolute requirement for this level to be met. (Supplementary Guidance to the GRA)
4. The Basic Safety Objectives cited in the Office for Nuclear Regulation's Safety Assessment Principles [30] apply in relation to the aggregate effective dose received through (as applicable) exposure to direct radiation, and inhalation and ingestion of radioactive substances (including ingestion via the food chain) from sources on the site. The Basic Safety Objectives form benchmarks that reflect modern safety standards and expectations. They also recognise that there is a level beyond which further consideration of the safety case will not be a reasonable use of resources, compared with the benefit of applying these resources to areas of higher risk.
5. In Statutory Guidance to the Environment Agency concerning the regulation of radioactive discharges into the environment [31], DECC and Welsh Assembly Government state (para. 22), *"Where the prospective dose to the most exposed groups of members of the public from discharges from a site at its current discharge limit is below 10 µSv/yr the Environment Agency [or Natural Resources Wales] should not seek to reduce further the discharge limits that are in place, provided that the holder of the [permit] applies and continues to apply BAT [best available techniques]."* The accompanying foot note to para. 22 states, *"... It should be noted that the 10 µSv/yr figure is not as dose target, a dose limit, threshold or a radiation standard. Instead it represents an appropriate level of dose, below which discharge limits should not be reduced further if the operator is continuing to apply BAT."*

3.3 Current international status of assessment work on non-human biota

The GRA [33] states (Requirement R9) that the radiological effects of the GDF on the accessible environment (such as those on non-human species or more general effects such as damaging habitat quality) should be investigated with a view to showing that all aspects of the accessible environment are adequately protected. However, there are no statutory requirements for such assessment and no criteria in UK legislation against which to compare assessment results. Therefore, the RPCM does not include radiological safety criteria relevant to non-human biota or to the wider environment, and safety objectives must be defined in other ways.

Reviews undertaken by the IAEA and United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) of the effects of ionising radiation on organisms (reported respectively as [37; 38]) give a broad conclusion that exposure of terrestrial animals to dose rates of 40 µGy⁵ per hour, and exposure of terrestrial plants and all aquatic organisms

⁵ Gray (Gy) is the unit of absorbed dose, where 1 gray is equal to 1 joule per kilogram. Doses to humans are calculated using the effective dose, which converts grays into sieverts using various tissue weighting factors for each tissue and an organ equivalent dose for each organ. At the present time, no tissue weighting factors or organ equivalent doses have been developed for non-human biota, thus effective doses cannot be calculated. Therefore, doses to non-human biota are quoted as absorbed doses (Gy).

to dose rates of 400 μGy per hour, are unlikely to lead to observable effects in these populations. These values correspond to US Department of Energy benchmarks [39].

Subsequent work conducted under the European Commission (EC) 6th Framework project ERICA [40] derived predicted no-effect dose rate (PNEDR) values for generic freshwater, marine and terrestrial ecosystems (and their sub-organisational levels) exposed to radioactive substances [41]. The object of protection within the ERICA approach is that generic ecosystems should be protected from effects on structure and function under acute and chronic exposure to radionuclides. PNEDR values of 10 μGy per hour for all generic ecosystems were derived through analysis of a large amount of data on the effects of ionising radiation in non-human biota collated in pre-existing databases and generated by experiments conducted within ERICA. The methodology followed EC recommendations for the estimation of predicted no-effect concentrations for chemicals [42]. These values were further endorsed by the EU PROTECT (Protection of the Environment from Ionising Radiation in a Regulatory Context) project [43], and were considered to be current as at the last major update of the ERICA tool in 2014.

In parallel with the EC's ERICA work, the ICRP has developed a framework for the protection of the environment based on derived consideration reference levels (DCRLs) for a number of reference animals and plants (RAPs) [44], and has provided guidance on the application of the protection framework for planned, existing and emergency situations [45]. A DCRL can be considered as a band of dose rate, spanning one order of magnitude, within which there is some chance of deleterious effects from ionising radiation occurring to individuals of that type of RAP. In planned exposure situations (such as that assessed in the OESA), the lower boundary of the relevant DCRL band should be used as the appropriate reference point for protection of different types of biota within a given area during the planning of controls to a source.

The ERICA and ICRP approaches are compatible, with the former based on the protection of ecosystems and the latter based on the protection of individuals within those ecosystems. In most cases the ERICA screening level of 10 μGy per hour is the more stringent; however, for some RAPs in planned exposure situations, the ICRP recommends a lower DCRL of 4 μGy per hour (Table 2).

The ongoing NERC-TREE project is conducting further studies on the effects of acute and chronic exposures on wildlife, but published results are not yet available. Therefore, in this generic OESA, the results of the assessment of radiological impact on non-human biota (presented in section 6.2) are compared against the both the ERICA screening dose rate value of 10 μGy per hour, and (where they are more stringent) the recommended ICRP DCRLs (Table 2).

Table 2 ICRP recommended DCRLs for RAPs relevant to a terrestrial ecosystem [45]

Reference animal or plant	DCRL (mGy/day)	DCRL (μ Gy/h)
Deer	0.1–1	4–40
Rat	0.1–1	4–40
Pine tree	0.1–1	4–40
Grass	1–10	40–400
Bee	10–100	400–4000
Worm	10–100	400–4000

4 Derivation of Off-Site Discharges

The purpose of this section is to consider the potential source terms for off-site solid, liquid and gaseous discharges, based on the wastes to be disposed of within the facility and the activities that will take place during the GDF operational period. Both radioactive and non-radioactive discharges have been considered.

4.1 Solid discharges

The operational activities that may give rise to off-site solid radioactive and non-radioactive discharges are described in the following sections, both for the surface facilities and those underground.

4.1.1 Surface facilities

Solid radioactive discharges

The facilities associated with surface based activities are identified in section 2.3. Solid waste such as plastic, paper, clothing, wood and metallic items will arise from routine monitoring and maintenance activities, and will be disposed of in accordance with best available techniques (BAT). It is currently assumed that any solid radioactive waste arising from activities taking place in the surface facilities will be LLW and will be sent to a Low Level Waste repository for disposal. Disposal within the GDF, which would require provision of appropriate waste packaging and handling facilities within the GDF and definition of appropriate waste acceptance criteria, is a back-up option for such wastes. Any such solid waste arising will need to be compared with waste acceptance criteria and be suitably conditioned and packaged.

Solid non-radioactive discharges

It is expected that there will be no solid non-radioactive (hazardous, non-hazardous and inert wastes as defined under the European Waste Directive) off-site discharges from the surface facilities arising from the wastes themselves. However, it is acknowledged that there will be non-radioactive solid wastes from other sources, such as general office wastes, generated from the surface facilities; these are not considered further in this report and will be considered as part of an EIA at a later stage in the implementation of geological disposal.

4.1.2 Underground facilities

Solid radioactive discharges

Underground activities include the removal of waste packages from shielded waste transport containers in a shielded inlet cell, inspection and monitoring activities, and transfer to and emplacement of waste packages in the appropriate disposal vault or tunnel. If disposal areas are kept open for an extended period, waste package remediation may also take place. As with the surface based activities, such wastes will be disposed of in accordance with BAT; it is currently assumed that any solid radioactive waste arising from activities taking place in the underground facilities will be LLW and will be sent to a Low Level Waste repository for disposal, with disposal within the GDF as a back-up option.

Solid non-radioactive discharges

Similarly, it is expected that there will be no solid non-radioactive (hazardous, non-hazardous and inert wastes as defined under the European Waste Directive) off-site

discharges from the underground facilities arising from the wastes themselves. However, it is acknowledged that there will be non-radioactive solid wastes from other sources, generated from the underground facilities; these are not considered further in this report.

4.2 Liquid discharges

The operational activities that may give rise to off-site liquid radioactive and non-radioactive discharges are described qualitatively in the following sections, both for the surface facilities and those underground. Both aqueous and non-aqueous liquids may be generated and it is assumed that any aqueous liquid generated from an active process will require management under the RSR regime irrespective of the level of radioactivity present.

The nature and extent of liquid discharges (both radioactive and non-radioactive) will be strongly dependent on both the hydrological and hydrogeological regime at a specific site (which will control, for example, groundwater ingress into the facilities), and detailed process and facility design. Neither of these aspects is known or sufficiently well developed at this generic stage for a meaningful estimate of liquid discharges to be made, and no quantitative assessment is therefore attempted in this report.

4.2.1 Surface facilities

Liquid radioactive discharges

A number of the surface facilities, such as decontamination, inspection and maintenance facilities, active laundry and laboratories, may give rise to radioactive liquids requiring disposal. Any radioactive liquid arisings will be collected and treated as appropriate in an active effluent treatment plant (AETP). Treated liquids will be monitored prior to discharge to ensure that all discharges from the site are in compliance with permit limits. Liquid discharges from the surface facilities will be managed as described above. Furthermore, the local aquatic environment and thus the management of liquid radioactive discharges will be highly site-specific, and a more detailed design will also need to be available before meaningful assessments of public doses from liquid effluents can be made. They are therefore not further assessed in this report, although it is noted that it may be possible to derive a bounding estimate for a radioactive liquid effluent source term in the future (section 8.3).

Liquid non-radioactive discharges

Surface drainage will be designed to prevent liquid effluents and run-off from entering the underground part of the facility [21]. It is expected that any non-radioactive liquid discharges from the surface facilities will not be associated with the wastes for emplacement in the GDF and will be considered in a future EIA. Any liquids containing non-radioactive contaminants will be analysed and treated as appropriate prior to discharge from the facility to ensure compliance with relevant standards and permit requirements.

4.2.2 Underground facilities

Liquid radioactive discharges

There are three potential sources of liquid discharges during the operational period that require consideration:

- Liquid radioactive wastes may arise from underground activities such as waste package inspection and maintenance. These will be pumped or transferred in a bowser to the surface and then treated in an AETP and discharged in the same way as liquid effluents arising in surface facilities.

- Depending on the hydrogeological setting, there may be groundwater intrusion into the active areas during the operational period; this will be managed and kept to a minimum level [46]. Liquid ingress will be diverted away from waste packages and collected to minimise the potential for corrosion or contamination. Any groundwater collected from active areas will be monitored and discharged, ensuring compliance with permit limits.
- Depending on the host rock, the groundwater may contain naturally occurring radionuclides. Any such radioactivity is highly site-specific and will be considered on that basis, but could be managed through an AETP if necessary.

Radioactive liquid effluent discharges from the underground facilities will be managed as described above and cannot be meaningfully assessed at the generic stage, as also described above. They are therefore not assessed further in this report.

Liquid non-radioactive discharges

As described above, groundwater intrusion from the host rock into the facility will be managed in the underground facilities during operation [46].

Any groundwater for disposal will be monitored to confirm it is non-radioactive and discharged to ensure compliance with permit limits; this will be through a system separate from any potentially radioactive liquids. Any liquids that may contain polluting or hazardous non-radioactive materials will be analysed and treated as appropriate prior to discharge from the facility to ensure compliance with relevant standards and permit requirements. Therefore any effects from the discharge of non-radioactive liquids from the underground facilities have not been assessed in this report, but will be managed to comply with permit limits.

4.3 Aerial discharges

Aerial discharges may be particulate or gaseous. The design intent is that the GDF itself should be free of contamination; therefore any radiologically significant releases of particulate radioactivity would be the result of accidents. Releases from accidents are not quantified in this generic OESA but will need to be assessed in a future OESA. Offsite releases from accidents are also considered in the generic OSC. Furthermore, any aerial discharges from active facilities will undergo several stages of high efficiency particulate in air (HEPA) filtration. Particulate discharges are therefore expected to be very low under normal operating conditions and potential doses arising from them are not considered further in this illustrative generic assessment.

It is noted that the vast majority of gaseous discharge volume from the GDF will be ventilation air. Although the environmental effects of air discharge (for example air discharged at non-ambient temperature or humidity) will need to be assessed in future (section 8.3), they are not considered to be significant for this generic OESA and the remainder of this assessment focuses on the non-air gases generated by activities at the GDF.

The main mechanisms by which gas is generated in the GDF are reviewed in RWM's Gas Status Report [47]. In the GDF, processes that could generate either large volumes of bulk gases or significant amounts of radioactive gases are:

- Corrosion of metals leading to the release of C-14 and tritium trapped in the metal
- Microbial degradation of organic materials, including the prior hydrolysis of cellulose to smaller organic compounds
- Radiolysis, in particular of water and some organic materials
- Diffusion, notably the release of tritium by solid-state diffusion from metals

- Radioactive decay of radium, which leads to the generation of Rn-222
- The release of radioactive gases containing tritium or C-14 by leaching of irradiated graphite

Rn-222 gas could also potentially be generated from naturally occurring radium in the host rock that forms the walls, floors and ceilings of the underground excavations of the GDF.

The rates at which most of the gases will be generated and released from the wastes are sensitive to environmental factors, which might change with time, such as: the presence of oxygen or water; the presence of hydrogen or chloride ions; and temperature. In particular, the process of backfilling the GDF, which occurs during the operational period, can affect the temperature (through exothermic cement curing reactions) and hence the rate of gas generation.

The operational activities that may give rise to radioactive and non-radioactive emissions to the atmosphere are described in the following sections for both the surface facilities and those underground.

4.3.1 Surface facilities

Aerial radioactive discharges

Operations in a number of the surface facilities will give rise to limited radioactive aerial discharges. For example, package receipt facilities; decontamination, inspection and maintenance facilities; remote handling facilities to allow minor repairs to be undertaken on any incoming transport containers failing acceptance tests; and laboratories may all discharge radioactive gases. All discharges from these facilities will be monitored and any particulates controlled with HEPA filtration.

The surface facilities will allow waste packages to be accepted at the facility, inspected and monitored as appropriate and then transferred from delivery vehicles to a drift wagon or other equipment as appropriate for transfer underground. Although there may be a certain amount of buffer storage before waste packages are able to be transferred underground, the aim is to take the packages directly underground and only a few packages may be stored temporarily at the surface.

Aerial radioactive discharges from the GDF are expected to be dominated by emissions from emplaced wastes underground. Therefore, radioactive aerial discharges from the surface facilities are not explicitly assessed in this report, except to note that discharges will be significantly lower than those assessed from the underground vaults.

Aerial non-radioactive discharges

Operations at the surface will also give rise to limited non-radioactive aerial discharges; for example, there may be a boiler, back-up diesel generators and vehicle emissions. However, as with the radioactive aerial discharges, it is considered that non-radioactive aerial discharges will be dominated by those from the underground facilities, which will include non-radioactive gases generated from emplaced wastes and large volumes of ventilation air as well as power generation and vehicle emissions. Therefore, non-radioactive aerial discharges from the surface facilities are not explicitly assessed in this report, except to note that discharges will be significantly lower than those from the underground vaults.

4.3.2 Underground facilities

Aerial radioactive discharges

The main point of release for aerial discharges from the underground facilities will be through the ventilation system and out via a discharge stack. Any particulates will be controlled with HEPA filtration. The discharge stack height and location will be designed to meet with the requirements specified in [21]. However, these are detailed aspects of the GDF that can only be defined once the actual site topography, height and proximity of the surrounding buildings have been established. As a consequence, for the purpose of assessing the effects of aerial discharges from the GDF in this generic OESA, dispersion modelling has been undertaken using an assumed effective release height⁶ of 15 metres. This height has been chosen for consistency with previous OESAs [12; 48], particularly the factors used in determining dose to non-human biota from radon, which are also used in this assessment (see section 5.2.4). However, subsequent work [49, 50] has determined that an effective release height of 30 metres is more realistic and still conservative. Future OESAs are likely to adopt the assumption of an effective release height of 30 metres, and the implications of doing so are explored semi-quantitatively in section 7 of this report.

Ventilation to all the radioactive areas underground will be supplied via a drift or shaft and discharged through a dedicated ventilation shaft [46]. Aerial discharges from the vaults will be monitored as appropriate prior to discharge. Aerial discharges from construction and operational areas will also be monitored as appropriate for radon emanating from the host rock. Personnel access to underground areas will be controlled to minimise exposure of underground workers.

The main source of aerial discharges from the radioactive areas underground will be gases emanating from the emplaced wastes. Ra-222 emanating from the host rock could also contribute to aerial discharges. These gases are discussed further in sections 4.4 and 4.5.

Aerial non-radioactive discharges

Flammable gases (mainly hydrogen but also possibly methane) generated by the emplaced wastes will be managed, through ventilation, to keep the concentration of these gases below their lower flammability limit in air. Asphyxiant and chemically toxic gases will also be managed through ventilation and, where necessary, controlled access underground to minimise exposure of underground workers. Any non-radioactive particulate material will be controlled by HEPA filtration prior to discharge.

Emissions of polluting, chemically toxic and greenhouse gases have been considered qualitatively in the generic DSSC in the generic Environmental Assessment Report [51] (which addresses the scope of an EIA), supported by a generic carbon footprint analysis [52]. Such emissions will be the subject of further work by RWM in the future (section 8.3). Therefore, non-radioactive aerial discharges from the underground facilities are not assessed further in this report.

4.4 Gaseous radioactive discharges – inventory

As noted previously, this generic OESA presents an assessment of operational discharges associated with all wastes and materials covered by the 2013 Derived Inventory [11], as described in the Technical Background. The assessment utilises a combination of quantitative and qualitative arguments and methods that are described later in this section.

⁶ The effective release height takes into account the physical discharge stack height, local topography, and efflux parameters (parameters that affect the flow of gas being discharged, such as exit diameter, velocity and buoyancy).

The discussion above concludes that the key radioactive discharges of interest from the GDF will be gaseous discharges from emplaced wastes released via the discharge stack. Relevant gas release data are presented in [53], together with detailed discussion of the assumptions and methodology used to calculate them. The main points are also summarised in the following sections of this report.

4.4.1 Significant radionuclides and radioactive gases

The radioactive gases of potential importance to operational environmental safety are primarily tritium, gaseous molecules containing C-14, and Rn-222 [47].

Although other radionuclides are gaseous species in their own right (for example Kr-85) or can be incorporated into volatile species (for example Se-79 in hydrogen selenide or dimethyl selenide), these are expected to be of relatively little significance [54]. They have therefore not been assessed in this report.

The primary C-14-bearing gaseous species expected to be generated are methane, carbon monoxide and carbon dioxide. The dose per unit release to members of the public differs considerably between these different species; therefore, methane and carbon monoxide are considered individually. It is assumed that carbon dioxide will not be released as a gas, as it is expected to react with cementitious materials present within waste packages to form carbonate (there is very little C-14 in wastes that are specified as unencapsulated) [55], and it is therefore not considered further in this assessment. This assumption may need to be reconsidered if future packaging decisions mean that key waste streams (for example graphite) will not be packaged with cementitious materials.

No speciation is considered for tritium or Rn-222 in the assessment of gas generation and release. All Rn-222 is expected to be generated and released in the form of a monatomic gas, while potential speciation of tritium is considered at the dose assessment stage.

4.4.2 Radionuclide inventory

The 2013 Derived Inventory [11] describes a number of different types of wastes that may be disposed of within the GDF. The wastes can be divided into seven groups as follows, based on the vaults/tunnels in which they will be disposed:

- Unshielded legacy Low and Intermediate Level Waste (ULLW/UIIW)
- Shielded legacy Low and Intermediate Level Waste (SLLW/SILW)
- Waste in robust shielded containers (RSCs)⁷
- Nuclear new build (NNB) UIIW
- NNB SILW
- DNLEU
- HHGW, including HLW, SF, HEU and Pu

The legacy ILW/LLW and the new build ILW are grouped separately because, although some waste from each group could potentially be emplaced in the same vault, the wastes will be emplaced at different times, so in practice will generally be emplaced in different vaults.

Modelling of the generation and release of gas (as reported in [53] and forming the basis for the assessment in this report) was carried out for legacy ULLW/UIIW, legacy SLLW/SILW, and NNB UIIW. Modelling was not carried out for HHGW because such wastes are expected to be packaged in long-lived, unvented containers, so it is assumed

⁷ RSCs are referred to as ductile cast iron containers (DCICs) in the 2013 Derived Inventory.

that no radioactive gas will be released from such waste packages during the GDF operational period [47]. Modelling of Rn-222 generation was also carried out for waste in RSCs, for NNB SILW and for DNLEU. However, modelling of C-14 and tritium generation was not carried out for these wastes because the C-14 and tritium activities of the various materials are much smaller than (generally <1% of) the activities of the same materials in the other ILW, and the evolution of these packages is likely to be similar to other ILW packages. Therefore it is not expected that the rate of C-14 and tritium release from these wastes will significantly contribute to their release rate from the GDF overall.

The discussion below provides details of the source terms within the wastes that are expected to give rise to discharges of each significant gaseous radionuclide from the waste packages during the operational period. Activities are reported at the assumed start date of emplacement in the GDF; that is, 2040 AD for legacy wastes, 2046 AD for waste in RSCs, 2100 AD for NNB wastes, and 2106 AD for DNLEU (see section 4.5 and Table 6 for further discussion).

4.4.3 C-14

Sources of C-14 in the inventory for disposal are dominated by irradiated metals, irradiated graphite, spent ion-exchange resins and organic materials. These may act as sources of C-14-bearing gases, as discussed in the Gas Status Report [47]. This discussion incorporates the findings of recent work by RWM as part of the carbon-14 integrated project [55; 56; 57], which has led to an improved understanding of the production and consequences of C-14-bearing gases from waste emplaced in the GDF, and which forms the basis for this assessment as it relates to C-14.

The proportion of the total C-14 inventory present in each relevant waste material group at 2200 AD (a date selected to ensure that all waste has arisen) is provided in the report on inventory for the carbon-14 integrated project [58]. These data were back-decayed to the assumed emplacement start date to provide the input data required for the release rate calculations, summarised in Table 3.

Table 3 Effective assignment of C-14 activity by material type [53]

Material type	C-14 activity (TBq)		
	Legacy ULLW/ UILW at 2040 AD	Legacy SLLW/ SILW at 2040 AD	NNB UILW at 2100 AD
Stainless steel: 2F03/C	3.00×10^1	–	–
Stainless steel: 2F08	3.91×10^1	–	–
Stainless steel: AP301	–	–	1.20×10^3
Stainless steel: EP302, EP303	–	–	5.55×10^3
Stainless steel: Other waste streams	8.52×10^1	6.01×10^1	6.70×10^{-2}
Mild steel	4.28×10^1	1.40×10^2	1.15×10^0
Zircaloy	2.85×10^1	7.85×10^{-1}	–
Nimonic	2.79×10^1	5.13×10^0	–

Material type	C-14 activity (TBq)		
	Legacy ULLW/ UILW at 2040 AD	Legacy SLLW/ SILW at 2040 AD	NNB UILW at 2100 AD
Magnox (uncorroded): Plates	6.76×10^1	–	–
Magnox (uncorroded): Spheres	–	–	–
Uranium (uncorroded): Plates	9.26×10^0	–	–
Uranium (uncorroded): Spheres	8.76×10^0	–	–
Graphite	7.50×10^2	6.31×10^3	–
<i>GE Healthcare waste*</i>	2.08×10^2	–	–
<i>Other*</i>	7.62×10^1	4.48×10^0	3.14×10^0
Total considered in gas release rate calculations	1.09×10^3	6.52×10^3	6.75×10^3

* GE Healthcare waste is expected to be incinerated (and so assumed not to be disposed of in the GDF). In the case of other wastes, as discussed in [55], there is not expected to be any release of C-14. Therefore, both 'GE Healthcare waste' and 'Other' groups are excluded from calculations of release rates of C-14.

From this data set it is clear that the highest activity for C-14 is contained within the graphite present in SLLW/SILW, and NNB stainless steel waste streams.

4.4.4 Rn-222

Rn-222 is generated from Ra-226, which is present in high concentrations in a limited number of ILW streams. DNLEU in the form of a uranium oxide will also be a source of Ra-226 and hence Rn-222. DNLEU is not in secular equilibrium with its daughter products, and therefore the Ra-226 activity in these waste streams will continue to increase (albeit at a very slow rate) during the period of operation.

The activity of Ra-226 in relevant waste streams at the assumed emplacement start date was obtained from the 2013 Derived Inventory [11]. The rate of Rn-222 release from the waste packages depends on both its generation rate due to decay of Ra-226 and the emanation coefficient for each waste package (see section 4.5.2). In the long term it will also depend on the decay of other parent radionuclides that provide in-growth of Ra-226, but this is not significant over the GDF operational period, so was not considered in this quantitative assessment. The emanation coefficient depends on the encapsulation material used (see section 4.5.2); therefore, Table 4 summarises the Ra-226 activity in groups of waste streams that are assumed to be encapsulated using the same material. Although some legacy ILW, all NNB ILW and all DNLEU are assumed to be encapsulated in cement, these are also separated in the table to provide an indication of the wastes that are calculated to provide significant contributions to the Rn-222 release rate.

Table 4 Ra-226 activity in each waste group at time of initial emplacement [53]

Waste group	Encapsulation material	Ra-226 activity at initial emplacement (TBq)	Year of initial emplacement
Legacy ILW	Polymer*	8.73×10^0	2040 AD
Legacy ILW	Cement	4.00×10^{-1}	2040 AD
Waste in RSCs	Unencapsulated	9.87×10^{-3}	2046 AD
DNLEU	Cement†	5.59×10^{-2}	2106 AD
NNB ILW	Cement	1.35×10^{-6}	2100 AD
Total		9.2×10^0	

* The waste streams that are assumed to be encapsulated in polymer are 1A10, 5C30, 5C52, 6C32, as indicated by recent work on radon emanation from waste packages [59]. For these wastes it is actually assumed that they are first encapsulated in polymer and then in cement.

† Since calculation of the gas generation rates, RWM has improved its assumptions on the packaging of DNLEU. Since this assessment, it is now assumed that DNLEU that is less than 1% enriched in U-235 (ie depleted uranium tails from enrichment and depleted uranium arising from reprocessing of Magnox fuel) will be unencapsulated [11]. The potential effects of this change in assumption are discussed in Section 7.1.1.

Radon emanating from the host rock

As noted earlier, the host rock itself may be a source of Rn-222, generated from Ra-226 that is itself formed as part of the decay chain of naturally occurring U-238. The Rn-222 arising from this source is further discussed and quantified in section 4.5.2.

4.4.5 Tritium

Tritium may be present in ILW metals (for example fuel cladding) as hydrides or as dissolved hydrogen. It will also be present in other materials (for example irradiated graphite) and trapped as tritiated water on desiccants [47].

The proportion of the total tritium inventory present in each relevant waste material group at 2200 AD is provided in the 2013 Derived Inventory [11]. These were back-decayed to the assumed emplacement start date to provide the input data required for the release rate calculations, summarised in Table 5.

The tritium inventory is defined for the same materials as the C-14 inventory, so that the calculations for tritium and C-14 generation can be performed together using the same tool.

Table 5 Effective assignment of tritium activity by material type [53]

Material type	H-3 activity (TBq)		
	Legacy ULLW/ UILW at 2040 AD	Legacy SLLW/ SILW at 2040 AD	NNB UILW at 2100 AD
Stainless steel: 2F03/C	1.06×10^3	–	–
Stainless steel: 2F08	1.46×10^2	–	–
Stainless steel: AP301	–	–	3.88×10^4
Stainless steel: EP302, EP303	–	–	2.07×10^5
Stainless steel: Other containing C-14	6.32×10^2	9.49×10^1	1.47×10^{-4}
Stainless steel: Not containing C-14	2.06×10^1	1.42×10^4	–
Mild steel: Containing C-14	6.95×10^1	1.21×10^3	2.57×10^1
Mild steel: Not containing C-14	3.48×10^{-1}	1.29×10^3	–
Zircaloy	1.51×10^3	3.35×10^1	–
Nimonic	1.33×10^0	2.28×10^0	–
Magnox (uncorroded): Plates	4.07×10^2	–	–
Magnox (uncorroded): Spheres	–	–	–
Uranium (uncorroded): Plates	6.97×10^1	–	–
Uranium (uncorroded): Spheres	1.05×10^3	–	–
Graphite	4.38×10^2	2.53×10^3	–
<i>Other*</i>	9.42×10^2	3.97×10^{-4}	1.26×10^0
Total considered in gas release rate calculations	5.40×10^3	1.94×10^4	2.46×10^5

* The nature of the 'Other' wastes means that it is unlikely that the release rate of tritium will be significant (the main sources of tritium release are expected to be metals and graphite), so the tritium in these wastes has not been included in the calculations. However, the potential tritium release from these wastes can be quantified using simple arguments and comparison with other calculated tritium release rates (see section 7.1.1).

From this data set it is clear that the highest activity of tritium is contained within stainless steel waste streams (particularly legacy SLLW/SILW and NNB UILW), with smaller amounts in mild steel, zircaloy, Magnox, uranium and graphite.

4.5 Derivation of offsite gas release rates

The calculation of release rates of significant radioactive gases (C-14-bearing methane, C-14-bearing carbon monoxide, Rn-222 and tritium) is presented and discussed in [53]. Release rates from emplaced waste were modelled over the operational period for a base scenario (referred to as a 'reference case' in [53]) and three variant scenarios that examine the sensitivity of the results to key parameters.

The operational period is assumed to be the period from first waste emplacement at 2040 AD until closure at 2200 AD. During the final 10 years of operations (2190 AD – 2200 AD), backfilling and closure of all the vaults and the rest of the GDF is assumed to occur. However, since not all waste will be surrounded by backfill until the end of the operational period, it is conservatively assumed that all gas is released into the ventilation system until the end of the operational period (2200 AD). Once all vaults have been backfilled, it is assumed that any gases generated will not be released into the ventilation system but will be retained within the backfill as free gas or dissolved in porewater. This backfilling schedule is based on the assumption for higher strength rock; as noted in section 2, for lower strength sedimentary rock and evaporites backfilling will occur immediately after waste emplacement, and this scenario is accounted for in the staged backfilling variant (see below). The host rock has no other significant implications on gas generation and release from emplaced wastes during the operational period.

Waste was assumed to be emplaced throughout the operational period, in campaigns according to the waste group. The assumed emplacement strategy and an explanation of how this was modelled is provided in [53, section 3.1.1], and summarised in Table 6. This assumed emplacement schedule is also presented in section 3.6.7.1 of the Data Report [17]. In practice, the emplacement will continue on a more or less continuous basis; however, for computational purposes the waste is assumed to be emplaced in batches, with each batch representing the total emplacement during a half-year of the emplacement period.

Table 6 Waste emplacement schedule assumed in gas generation modelling in this generic OESA [53]

Waste group	Emplacement start date	Emplacement duration
Legacy ULLW/UILW	2040 AD	66 years*
Legacy SLLW/SILW	2040 AD	73 years
Waste in RSCs	2046 AD	7 years
NNB UILW	2100 AD	40 years
NNB SILW	2100 AD	40 years
DNLEU	2106 AD	31 years

* The rate of emplacement for legacy ULLW/UILW is assumed to be rapid for the first 24 years of emplacement, then to reduce for the subsequent 42 years of emplacement.

The rates of some of the gas generating processes are dependent on temperature. It is expected that the temperature in the GDF will vary over time. In particular, backfilling has the effect of heating the waste packages as the backfill cures (cement curing is exothermic) and backfilling may make available free water that can be used in, for example, corrosion reactions. Several studies [60; 61] provide information on the expected temperature variation. Based on these, the temperature assumed for the GDF is 35°C during the emplacement stage and 45°C during backfilling and closure.

The mass of water in the waste packages, which could potentially affect the release rates of tritium and C-14 from some waste metals, since some corrosion reactions do not occur in the absence of water, is taken into account in the calculations [53].

The base scenario and variant scenarios (chosen to bound the range of possibilities in parameters where there are significant uncertainties) are described below:

- **Base scenario:** The base scenario was chosen to provide a reasonable generic representation of most aspects of the likely gas generation behaviour. The strategy used in defining this scenario was to use the best estimate case where one is available, and the conservative case otherwise. Hence, it uses the best estimate GDF temperature (as discussed above), the pessimistic assumption that uranium corrodes anaerobically (discussed further under 'Aerobic uranium corrosion variant'), and the baseline backfilling schedule (as described above). Hence, although it is considered likely, the base scenario does include some conservatisms, identified in the following sub-sections and discussed in section 7.1.1.
- **Higher temperature variant:** These calculations assume an upper bound temperature profile rather than the best estimate profile used in the base scenario. For the main operational period, there is insufficient information on how the temperature in a vault may increase above the best estimate value, so this is not varied in this scenario. For the backfilling period, the temperature will depend on the ambient rock temperature at the depth of the GDF. Based on the potential range of rock temperatures, it is assumed that the upper bound temperature for the waste during backfilling, which is used in the variant calculations, is 60°C.
- **Aerobic uranium corrosion variant:** This scenario assumes that uranium corrodes under aerobic conditions rather than anaerobic conditions as assumed in the base scenario. In the base scenario calculations, it is assumed that conditions in packages containing uranium are anaerobic at all times after emplacement in the GDF. However, it is not certain that conditions for uranium waste will be anaerobic during the operational period (particularly if the grout does not contain blast furnace slag). The rate of uranium corrosion is substantially lower for aerobic conditions than for anaerobic conditions, which could have a substantial effect on the overall C-14 and tritium release rates. Variant calculations have been performed assuming that the uranium corrodes under aerobic conditions during the operational period. The base scenario and variant calculations are intended to bound the range of possible uranium corrosion behaviours.
- **Staged backfilling variant:** In the base scenario calculations, backfilling is assumed to occur during the final 10 years of the operational period, after waste has been emplaced in all vaults. Staged backfilling is an alternative strategy in which a vault is backfilled as soon as it has been filled with waste. Backfilling occurs at a separate time for each vault rather than over a single period as in the base scenario calculations. A key consideration in interpreting the gas release results for the staged backfilling variant calculations is whether the gases released from waste packages are released into the GDF ventilation system, as it is this release that is relevant to this assessment. This is uncertain for these variant calculations as it is not clear whether a vault will be ventilated once backfilling has been completed. Given this, two cases are presented based on two alternative assumptions that

bound the releases via the GDF ventilation system. In staged backfilling variant A, it is assumed that gases are released into the GDF ventilation system from all vaults throughout the operational period. In staged backfilling variant B, it is assumed that gases are released into the GDF ventilation system only from vaults for which backfilling has not been completed.

It is noted that these variant scenarios explore the sensitivity of gas releases to operational and host rock-related factors, rather than potential changes in the inventory for disposal. As discussed in section 2.2, inventory uncertainty is managed through separate (generic DSSC-wide) inventory scenarios and a modular approach to assessments, whereby the individual contributions of different types of waste are determined. These are identified in the following discussion (and in more detail in [53]), allowing the effects of inventory scenarios on radioactive gas releases to be discussed qualitatively in section 7.1.1.

4.5.1 C-14-bearing gases

There are a number of uncertainties regarding the release of C-14 from corrosion of metals. In this assessment, the C-14 in metal wastes is all cautiously assumed to be released as gas as the metal corrodes, whereas only a fraction may be gas in practice. It is also assumed to be evenly distributed throughout the metal, whereas the true distribution is unknown. Furthermore, the resulting dose will depend on the gaseous species formed, and there is considerable uncertainty about the form in which C-14 will be released from corrosion of metals. Release as methane and carbon monoxide is considered in [53]; as discussed earlier, carbon dioxide is assumed not to be released as a gas, as it is expected to react with cementitious materials present within waste packages to form carbonate [55]. The dose per unit release is several orders of magnitude higher for carbon monoxide than methane (section 5.1.2), so in the subsequent discussion and quantitative dose assessments, it is cautiously assumed that all C-14 released from corrosion of metals is in the form of carbon monoxide. The implications of these uncertainties and assumptions, most of which are conservative, are discussed further in section 7.1.1.

The release of C-14 from graphite can be rapid (acute), slow (chronic) or zero. The acute period of relatively rapid C-14 release is not relevant to calculations for the GDF, as this release will occur when the graphite is initially packaged. Only the subsequent chronic release of C-14 from irradiated graphite is of potential interest to the OESA. A parameterisation exercise for the graphite model was undertaken as part of the carbon-14 integrated project [62]. The values for the fraction of C-14 available for slow release, and the rate constant of slow release, as well as the speciation of the released C-14 (all used in the calculation of release rates) are taken from this.

Figure 3 shows the release of C-14-bearing carbon monoxide over the operational period for the base scenario. The main component is from the legacy ULLW/UILW throughout the operational period. The peak release rate is 0.43 TBq per year and occurs at 2064 AD (the end of the 24-year rapid emplacement period for legacy ULLW/UILW). At around 2100 AD, the release rate decreases by a factor of ~4 to ~0.1 TBq per year until the backfilling period, when it rises again to close to initial levels. The key processes contributing to the release are corrosion of the uranium and Magnox.

Figure 4 shows the release of C-14-bearing methane over the operational period for the base scenario. The main component is from the legacy SLLW/SILW throughout the operational period. The peak release rate is 0.012 TBq per year and occurs at 2112 AD (the end of the first emplacement period for legacy SLLW/SILW). The key process contributing to the release is graphite degradation.

Figure 3 Summary of calculated release rates for C-14-bearing carbon monoxide during the GDF operational period base scenario, assuming C-14 released from metals is all carbon monoxide

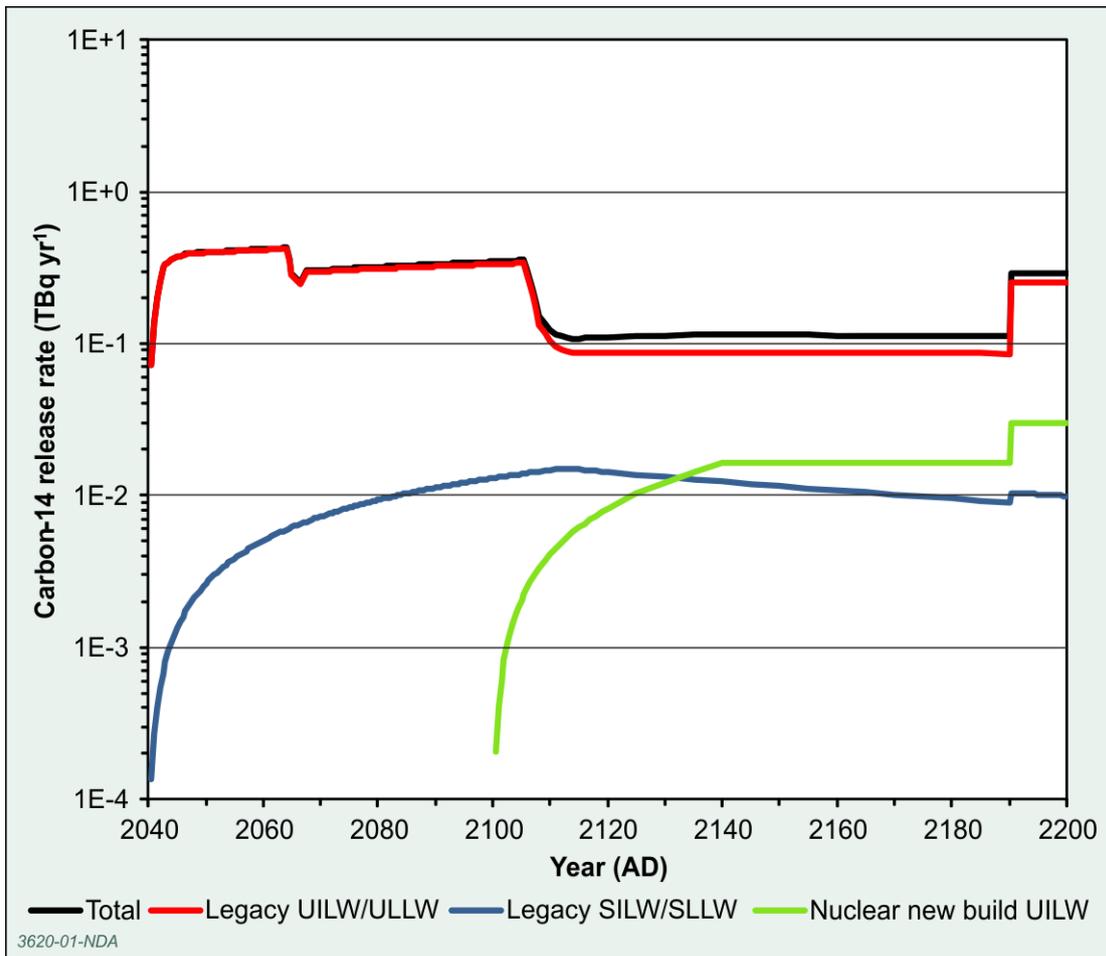
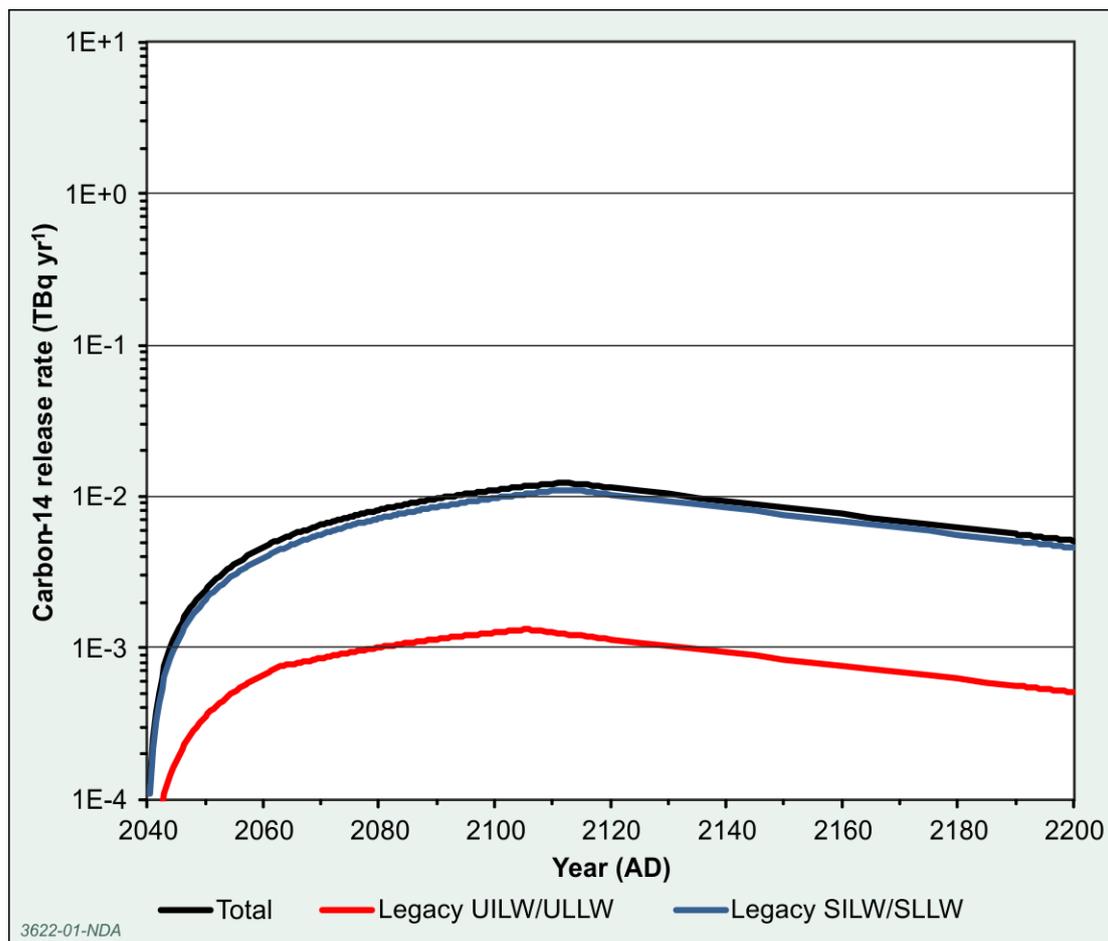


Figure 4 Summary of calculated release rates for C-14-bearing methane during the GDF operational period base scenario, assuming C-14 released from metals is all carbon monoxide



In the higher temperature variant, the results only differ after 2190 AD. The effects of the higher temperature between 2190 and 2200 AD are to increase the release rate from corrosion of Magnox by about a factor of five, from corrosion of stainless steels by about a factor of two, and from corrosion of mild steels by about 40%. This means that, in this scenario, the peak release rate of C-14-bearing carbon monoxide now occurs at 2190 AD and is 1.3 TBq per year. The magnitude and timing of the peak release rate of C-14-bearing methane are unchanged. Plots of the release rates over time for the variant scenarios can be found in [53].

In the aerobic uranium corrosion variant scenario, uranium corrosion is less rapid and continues throughout the operational period, rather than the uranium in each package completely corroding shortly after emplacement. As a result the C-14-bearing carbon monoxide release rate from uranium corrosion increases gradually during emplacement, then reduces gradually, and the peak rate from this source is only about half of the peak rate in the base scenario. This means that the peak release rate in this scenario occurs at the start of the backfilling period (2190 AD), and is lower than the peak in the base scenario at 0.37 TBq per year. The magnitude and timing of the peak release rate of C-14-bearing methane are unchanged.

In the staged backfilling variant scenarios, the peak release rate of C-14-bearing carbon monoxide is slightly higher than in the base scenario and is relatively insensitive to which of variants A and B is assumed (0.51 TBq per year at 2061 AD and 0.48 TBq per year at 2052 AD respectively), since the rapid corrosion of uranium means that most of the peak release

occurs from a single vault and the temperature is slightly higher (due to earlier backfilling) at the time of peak release rate. For C-14-bearing methane, the peak release rate is 0.016 TBq per year at 2116 AD for staged backfilling variant A, and 0.0031 TBq per year at 2055 AD for staged backfilling variant B.

4.5.2 Rn-222

The rate of generation of Rn-222 within the wasteform can be calculated readily from the activity of Ra-226 present. However, this information is not sufficient, as it will take some time for the generated Rn-222 to migrate out of a waste package and, due to its short half-life (3.82 days), much of the Rn-222 will decay within the waste package. To allow for this, an emanation coefficient can be specified that is the ratio of Rn-222 escaping from the waste package to that generated by decay of Ra-226.

The emanation coefficient will depend on the waste material, the encapsulation material (if any), and their geometries, as well as the rate of bulk gas generation. Recent work has defined best estimate values for the emanation coefficient based on wastes packaged using either a cement encapsulant or a polymer encapsulant [59] and assuming no bulk gas generation. It is noted that the values defined do not take any credit for the time taken for Rn-222 to migrate through the waste itself, and are based on somewhat cautious assumptions in terms of the geometry of the encapsulation materials.

Given the dependence of the emanation coefficient on the encapsulation material, the Ra-226 inventory has been divided based on the expected encapsulation arrangements of each waste stream (see section 4.4.4). Three encapsulation arrangements have been identified, with emanation coefficients determined based on best estimate values from recent work [59]:

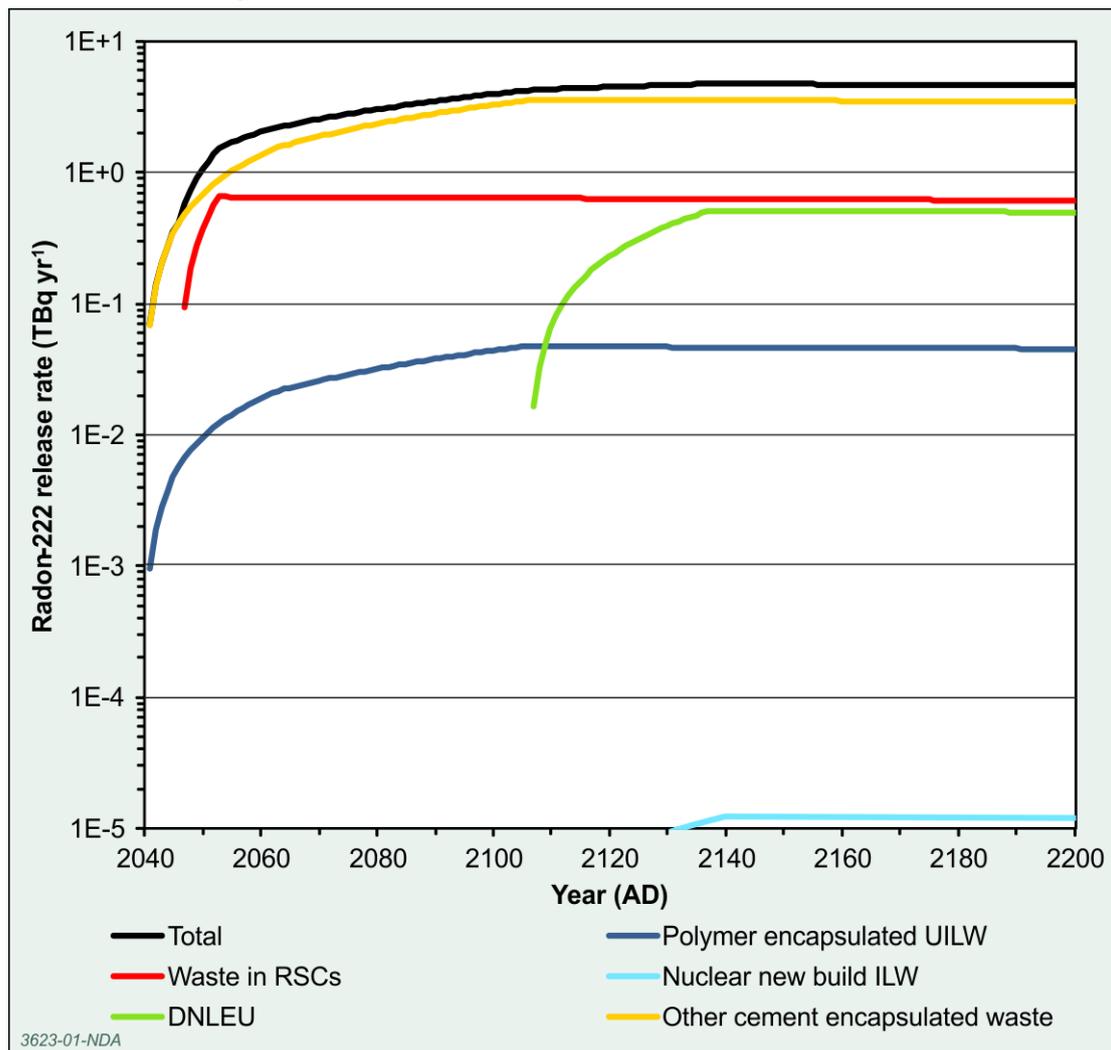
- Encapsulation in cement – emanation coefficient: 0.14
- Encapsulation in polymer, followed by encapsulation in cement – emanation coefficient: 8.40×10^{-5} (product of the emanation coefficients for encapsulation in each of the materials)
- Packaging without encapsulation – emanation coefficient: 1

These emanation coefficients, assumed to be constant over the operational period, have been incorporated into the calculations of Rn-222 release presented in [53]. In practice, these will vary between individual wastes, as well as with factors such as temperature, although the values have been chosen to be cautious.

It is noted that an emanation coefficient is not needed for C-14 or tritium because the decay of these radionuclides during migration out of a waste package is insignificant, due to their much longer half-lives.

In the base scenario, Rn-222 release rates reach a peak of 4.7 TBq per year at 2138 AD (Figure 5). There are initial increases in release rates due to emplacement for each waste group, following which release rates are virtually constant over the duration of the operational period. The main contribution to the release rate is from cement-encapsulated legacy ILW, with additional significant contributions from waste in RSCs and DNLEU.

Figure 5 Breakdown by waste group of calculated release rates for Rn-222 during the GDF operational period base scenario



None of the variant scenarios considered in [53] affect the release rates of Rn-222, with the exception of staged backfilling variant B. In this scenario the peak release rate of Rn-222 is 0.94 TBq per year, reached at 2052 AD. This is much lower than for the base scenario, since there are only contributions from a few vaults at any time rather than from all vaults once emplacement is complete.

Radon emanating from the host rock

Recent work [63] has derived indicative rates of Rn-222 emanation from generic host rocks. These are used in this assessment to provide a first approximation of the potential contribution of Rn-222 emanating from the host rock to the overall release of Rn-222 from the GDF.

Indicative emanation rates for specific vaults are presented in [63] for generic higher strength rock and lower strength sedimentary rock, and used to derive indicative emanation rates per square metre of rock face. These can be multiplied by the total underground excavation surface area (including all vaults, tunnels, drifts and shafts) of the GDF as currently designed for higher strength rock and lower strength sedimentary rock, to give the overall rate of Rn-222 emanation from the host rock across the entire GDF (Table 7; section 3.8.3 of the Data Report [17, Table 7]). It is noted that this is conservative, as

construction, waste emplacement and closure will be phased so that only part of the total excavation area will be open at any one time for most of the operational period. No emanation rate has been derived for generic evaporite host rock; evaporites tend to have extremely low uranium contents, so the Rn-222 emanation is expected to be vanishingly small [63].

Table 7 Emanation of Rn-222 from generic host rocks

Generic host rock	Emanation rate of Rn-222 per m ² (TBq per year)	GDF excavation surface area (m ²)	Emanation rate of Rn-222 from entire GDF (TBq per year)
Higher strength rock	3.22×10^{-8}	6.60×10^6	2.13×10^{-1}
Lower strength sedimentary rock	1.61×10^{-8}	5.20×10^6	8.37×10^{-2}

These rates are at least an order of magnitude lower than the release rates of Rn-222 from wastes (Figure 5). To simplify the number of calculations, the assessment of dose in this generic OESA uses the emanation rate of Rn-222 associated with the GDF in higher strength rock, as Table 7 and [63] show that this assumption is bounding of other generic host rocks.

It is noted that there is significant uncertainty in many of the parameters underlying these indicative emanation rates, resulting in a large uncertainty on the emanation rate values. These are discussed further in section 7.1.1.

Total release rates of Rn-222 from the GDF

The total release rate of Rn-222 from the GDF at any time during the operational period is the sum of the Rn-222 released from the emplaced wastes (which varies over time, as discussed in [53] and above) and the Rn-222 emanating from the host rock (which is conservatively assumed in this assessment to be constant over the operational period). Thus, in the base scenario and all variants except for staged backfilling variant B, the peak release of Rn-222 is estimated to be 4.9 TBq per year at 2138 AD, of which 0.2 TBq per year (4%) is estimated to derive from the host rock. In staged backfilling variant B, the peak release rate is estimated to be 1.2 TBq per year at 2052 AD, of which 0.2 TBq (17%) is estimated to derive from the host rock.

4.5.3 Tritium

As for C-14, there is uncertainty regarding the release of tritium from corrosion of metals. In this assessment, the tritium in metal wastes is assumed to be evenly distributed throughout the metal. It is all cautiously assumed to be released as gas as the metal corrodes, whereas only a fraction may be gas in practice. These assumptions are discussed further in section 7.1.1.

Tritium can also be released from metals due to solid-state diffusion through the metal. The modelling of tritium diffusion as incorporated into the calculation of release rates is discussed in [53, section 3.2.2].

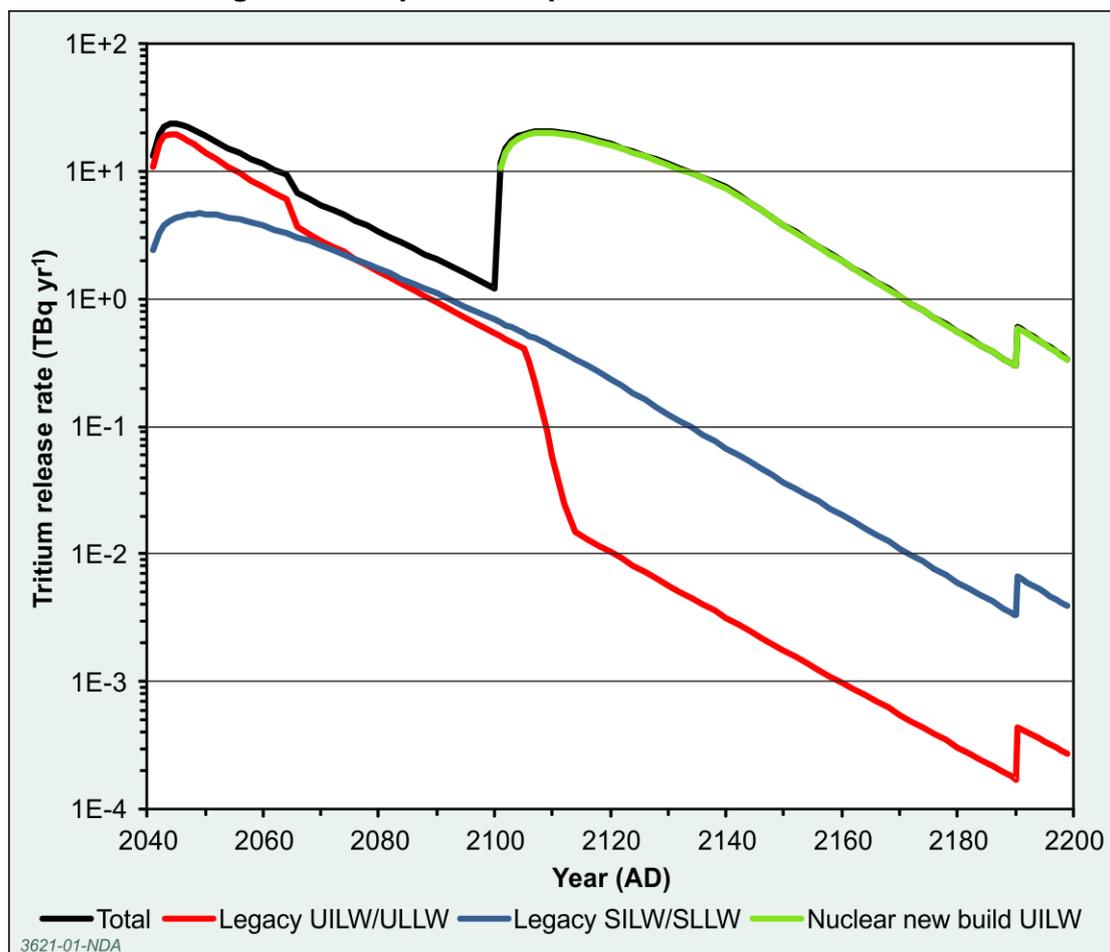
In many cases, the release of tritium from metals will be predominantly due to either corrosion or diffusion. As noted in [53], calculations of tritium release due to corrosion and diffusion cannot be performed together in the model used. Therefore, the release rate of tritium from metals is approximated by assuming that only the process giving the greatest release rate occurs for each of the steel and Zircaloy material groups (to avoid 'double counting' of the tritium inventory). For Magnox and uranium wastes, there are no data to

calculate the release rates due to diffusion. However, the corrosion rates of Magnox metal and uranium are several orders of magnitude greater than those for steel wastes, so the release of tritium due to corrosion is expected to be relatively rapid, irrespective of the rate of release due to diffusion.

Tritium release from graphite is also uncertain. As for C-14, the period of acute release is not relevant, so is not considered. From the experimental data available, the fraction of tritium available for release cannot be determined, so it is cautiously assumed that all the tritium may be released. The rate constant for slow release is estimated using the most recent measurements for gaseous tritium release from Oldbury graphite [64].

In the base scenario, tritium release rates for each waste group increase during the initial years of waste emplacement as more waste is emplaced (Figure 6). From part-way through the emplacement period for each waste group, the release rates begin to reduce due to radioactive decay. At 2190 AD, the increase in temperature associated with backfilling results in a small step increase in the tritium release rate. There are significant contributions to the tritium release rate from all waste groups during different parts of the operational period. The maximum release rate is 24 TBq per year, and occurs at 2044 AD (near the beginning of the emplacement period for the legacy ULLW / UILW). There is also a peak of similar magnitude (21 TBq per year) at 2108 AD (near the beginning of the emplacement period for the NNB UILW). The key processes contributing to these releases are corrosion of uranium and diffusion from stainless steel.

Figure 6 Summary of calculated release rates for tritium containing gases during the GDF operational period base scenario



For the higher temperature variant scenario, the results only differ after 2190 AD, where the rates are higher than for the base scenario, but still low overall. There is no change to the magnitude or date of the peak release rate.

In the aerobic uranium corrosion variant scenario, uranium corrosion is less rapid and continues throughout the operational period, rather than the uranium in each package completely corroding shortly after emplacement. As a result, the tritium release rate from uranium corrosion increases more gradually during the initial years of waste emplacement, before reducing as a result of radioactive decay. The peak release rate of tritium in this scenario is 21 TBq per year at 2109 AD, slightly lower and occurring slightly later than in the base scenario.

In the staged backfilling variant scenarios, the peak release rate for tritium is higher than in the base scenario because the higher temperatures resulting from backfilling occur before there has been any inventory reduction due to radioactive decay. The peak release rate is relatively insensitive to which of variants A or B (36 TBq per year and 34 TBq per year respectively, both at 2043 AD) is assumed, since the rapid corrosion of uranium means that most of the peak release occurs from a single vault.

4.6 Summary: radioactive gas release rates for use in off-site dose assessment calculations

Based on the analysis described above, Table 8 summarises the peak release rates of radioactive gases from the GDF (including Rn-222 emanation from the host rock) in the base scenario. Table 9 summarises the peak release rates of radioactive gases from the GDF in a conservative bounding case; this takes the highest peak release rate for each radionuclide across all variant scenarios and combines them in a single case. It is noted that this bounding case addresses uncertainties in GDF operations and conditions rather than uncertainties in the Derived Inventory, which are discussed qualitatively in section 7.1.1. Peak release rates in both the base scenario and the bounding case are presented in section 3.4.4 of the Data Report [17].

Peak release rates in both the base scenario and the bounding case are used subsequently in this report in the dose assessment calculations for gaseous emissions to the atmosphere. However, it is noted that these peak release rates only occur for a period of a few years; over the remainder of the operational period, the release rates will be somewhat lower (as shown in Figures 3 to 6). The resultant doses will be correspondingly lower and this is discussed in section 6.

Table 8 Peak radioactive gas release rates during the operational period of the GDF in the base scenario

Radioactive gas	Peak release rate for operational period base scenario (TBq per year)	Time of peak release
C-14-bearing CO	0.43	2064 AD
C-14-bearing CH ₄	0.012	2112 AD
Rn-222	4.9	2138 AD
Tritium	24	2044 AD

Table 9 Peak radioactive gas release rates during the operational period of the GDF in the bounding case

Radioactive gas	Peak release rate for operational period bounding case (TBq per year)	Time of peak release	Variant scenario giving rise to bounding case
C-14-bearing CO	1.3	2190 AD	Higher temperature
C-14-bearing CH ₄	0.012*	2112 AD	Higher temperature
Rn-222	4.9	2138 AD	Base scenario
Tritium	36	2043 AD	Staged backfilling A

* The release rate of C-14-bearing methane in the staged backfilling variant A is slightly higher than this at 0.016 TBq per year. However, in that case the release rate of C-14-bearing carbon monoxide is only 0.51 TBq per year. The bounding case shown above reflects the situation in which the combined release of C-14 is at its peak.

5 Methodology for Assessing Doses from Gaseous Emissions to the Atmosphere

Based on the consideration of operational source terms presented in section 4, emissions of GDF-derived radioactive gases to the atmosphere are considered to be of most significance with regard to a potential impact on the public and on the environment (non-human biota).

This section describes the methodologies for the assessment of doses that could potentially be received from operational emissions of GDF-derived radioactive gases to the atmosphere for two endpoint receptors – the public and non-human biota. These methodologies have been applied to the radioactive gas release rates presented in section 4, and the results are presented and discussed in section 6 .

5.1 Methodology for assessment of doses to members of the public

The Environment Agency, Scottish Environment Protection Agency and the Department of Environment in Northern Ireland, in collaboration with the Food Standards Agency and Health Protection Agency, have developed and published principles and guidance for the prospective assessment of public doses [35]. An initial radiological assessment methodology has also been developed by the Environment Agency [65; 66]. There are two volumes of this report; the first outlines the approach with some illustrative examples, and the second is the more detailed methodology report.

The Environment Agency guidance [35; 65] recommends a staged approach to dose assessments:

- Stage 1: Initial assessment using default data (as provided in [65] and [66]); if assessed dose is greater than the regulatory constraint, proceed to Stage 2
- Stage 2: Initial assessment using refined data (for example using the default data with scaling factors to account for local dispersion conditions); if assessed dose is greater than the regulatory constraint, proceed to Stage 3
- Stage 3: Site-specific assessment accounting for all relevant specific local conditions

It is not a requirement to use this staged approach; however, in the absence of a specific site, a stage 1 or 2 approach provides indicative and likely conservative estimates of dose. A stage 2 approach has been followed for the generic assessment presented in this report.

5.1.1 Public receptors

In order to calculate the dose to members of the public from gas discharges during the operational period, it is first necessary to define what is meant by the public and which receptor(s) should be used.

The GRA [33] states (requirement R5):

“During the period of authorisation of a disposal facility for solid radioactive waste, the effective dose from the facility to a representative member of the critical group should not exceed a source-related dose and a site-related constraint.”

The Environment Agency initial methodology [65; 66] explains the concept of the critical group as *“intended to be representative of those individuals in the population expected to receive the highest dose”*, based on recommendations from the ICRP [67]. However, the term ‘critical group’ has been superseded in ICRP guidance [28] by the term ‘representative person’, defined as a person who may be hypothetical, but whose habits (for example

consumption of foodstuffs, breathing rate, location, usage of local resources) are typical of a small number of individuals representative of those most highly exposed, noting that they should not be the extreme habits of a single member of the population. The Environment Agency has adopted this terminology [35].

Different members or groups of the public (based on their location and habits) may be chosen as representative persons, depending on the circumstances. The Environment Agency methodology [66] considers seven groups:

- For discharges to air, a 'local inhabitant family'
- For discharges to estuary and coastal water, a 'fisherman family'
- For discharges to a freshwater river, an 'angler family' and an 'irrigated food consumer family'
- For discharges to a public sewer, 'sewage treatment workers', a 'farming family', and 'children playing in brook'

The local inhabitant family is the relevant group for gas discharge from the GDF to the atmosphere in this generic OESA.

Doses can be assessed for representative persons of different ages, typically adults, 10-year-old children, 1-year-old infants, and new-born babies, as well as fetuses and embryos. In the Environment Agency methodology [66], the term 'offspring' is used to refer collectively to the embryo, foetus and new-born baby (following ICRP convention [68]).

5.1.2 Dose per unit release factors

The relationship between a radionuclide emission or discharge rate and the resultant dose received by a receptor can be expressed by a dose per unit release (DPUR) factor⁸. DPUR values have been calculated by the Environment Agency for 100 different radionuclides. The value of a DPUR depends on the radionuclide, the exposure pathway, and the location, habits and age of the representative person. The exposure pathways considered also depend on the location and habits of the representative person. The exposure pathways considered by the Environment Agency [65] for discharges to air for the local resident family are:

- Inhalation of radionuclides in the effluent plume
- External irradiation from radionuclides in the effluent plume and deposited to the ground
- Consumption of terrestrial food incorporating radionuclides deposited to the ground

The Environment Agency methodology report [66] describes how the DPUR values were calculated using the 1998 version of the software PC-CREAM, developed by the National Radiological Protection Board (now Public Health England) [69]. PC-CREAM is based on an EC methodology for assessing the radiological consequences of routine releases to the environment [70]. It contains modules that are used to calculate the transfer of radionuclides through different environments and the food chain; the module PLUME is used for atmospheric environments. PC-CREAM and its underlying dispersion models are robust, fit for purpose, and have been verified against environmental data [71, 72].

The Environment Agency DPUR values for C-14 have been superseded by recent work [73], which uses a more realistic representation of the discharge pathway for releases of

⁸ The term 'DPUR' is used here in preference to the equivalent 'Dose Release Ratio (DRR)' used in some other contexts, in order to reflect the terminology used by the environmental regulators. The term 'Effective Dose Factor' is used in recent work on carbon-14 [73], but is not used here to avoid confusion.

C-14-bearing gases from the GDF during operations. In particular, different C-14-bearing gases are considered separately (methane and carbon monoxide are those relevant to this assessment), and an effective release height of 15 metres is included in the factors calculated (in the Environment Agency methodology, the default DPUR values assume releases at ground level but can be scaled for different effective release heights; see section 5.1.3). Updated DPUR values for C-14 have been calculated for three age groups: infant, child and adult [73].

The DPUR values used in this assessment are presented in Table 10 (tritium and Rn-222) and Table 11 (C-14-bearing methane and carbon monoxide). These are discussed in section 3.10.3 of the Data Report [17].

Table 10 Dose per unit release values for a local resident family for tritium and Rn-222

Radionuclide	Age group	DPUR ($\mu\text{Sv}/\text{year}$ per Bq/year of discharge to the atmosphere, for releases at ground level)		
		Terrestrial food consumption ^{2,3}	External irradiation ^{2,4}	Inhalation ²
Tritium ¹	Offspring	5.6×10^{-13}	0	1.4×10^{-12}
	Infant	7.9×10^{-13}	0	5.9×10^{-13}
	Child	4.0×10^{-13}	0	8.6×10^{-13}
	Adult	3.8×10^{-13}	0	9.2×10^{-13}
Rn-222	Offspring	0	9.5×10^{-16}	0
	Infant	0	4.5×10^{-16}	2.4×10^{-10}
	Child	0	5.7×10^{-16}	1.9×10^{-10}
	Adult	0	9.5×10^{-16}	1.3×10^{-10}

1. Tritium is assumed to be in an organic molecule – this provides the highest factor and is therefore conservative
2. Values are defined in [66, Tables 3, 4, 5, 6]
3. Values for each of eight food groups are summed to provide the required values
4. Two values relating to external exposure are summed to provide the required values

Table 11 Dose per unit release values for a local resident family for C-14-bearing methane and carbon monoxide [73, Table 6]

Radioactive gas	DPUR accounting for all pathways ($\mu\text{Sv}/\text{year}$ per Bq/year of discharge to the atmosphere, for an effective release height of 15 m)		
	Infant	Child	Adult
C-14-bearing methane	1.5×10^{-14}	8.7×10^{-15}	9.0×10^{-15}
C-14-bearing carbon monoxide	1.2×10^{-10}	7.1×10^{-11}	7.2×10^{-11}

5.1.3 Scaling factors

Default assumptions in the Environment Agency methodology are that the release from the GDF is at ground level, the local resident is located at a conservative distance of 100 metres from the release point and food is produced at a distance of 500 metres from the release point.

However, at stage 2 of the methodology, scaling factors can be applied to dose calculations to account for site-specific dispersion conditions. For atmospheric discharges, a scaling factor is applied to account for the effective release point being higher than the ground level (where exposure is assumed to occur), which, as discussed previously, can occur as a result of discharge stack height, efflux parameters, local topography, or a combination of these factors. The higher the effective release height above ground level, the lower the dose from gaseous discharges will be.

Scaling factors may vary for different exposure pathways, depending on the habits assumed for the representative person. The Environment Agency guidance [65] assumes that the location of inhalation and external irradiation is nearer to the discharge point (100 metres) than the assumed food source (500 metres). As a result, the scaling factor is the same for the inhalation and external dose pathways, but different for the food pathway.

In this assessment, it is assumed that all radioactive gases from the GDF during the operational period will be released from an effective release height of 15 metres (as discussed in section 4.3.2). Using [65, Figure 2], this gives scaling factors of 0.09 for inhalation and external irradiation, and 0.45 for food consumption. These are applied to DPURs for tritium and Rn-222, but not to C-14-bearing methane and carbon monoxide, as the updated DPURs for these gases already include scaling for a release height of 15 metres (using the same scaling factors as the Environment Agency methodology). The new DPUR values for C-14-bearing gases assume residency and food production at distances of 100 metres and 500 metres from the release point respectively, the same assumptions as made in the Environment Agency methodology.

5.1.4 Calculation of annual dose

For each representative person, the dose from each radioactive gas (tritium, Rn-222, C-14-bearing methane and C-14-bearing carbon monoxide) is calculated separately.

For tritium and Rn-222, the following equation is used, as detailed in the Environment Agency methodology [65]:

$$\text{Dose } (\mu\text{Sv per year}) = [A \times B \times E] + [A \times C \times F] + [A \times D \times F]$$

Where

- A = Radioactive gas release rate (Bq per year)
- B = Food DPUR*
- C = External DPUR*
- D = Inhalation DPUR*
- E = Food dose scaling factor (no units)
- F = Inhalation and external dose scaling factor (no units)

* DPUR units are $\mu\text{Sv}/\text{year}$ per Bq/year of discharge to the atmosphere.

For C-14-bearing methane and C-14-bearing carbon monoxide, a single updated DPUR accounts for all pathways and an effective release height of 15 metres. Therefore, the dose for each representative person is given by the following equation (based on that used in the Environment Agency methodology [65]):

$$\text{Dose } (\mu\text{Sv per year}) = A \times G$$

Where

- A = Radioactive gas release rate (Bq per year)
- G = DPUR accounting for all exposure pathways ($\mu\text{Sv}/\text{year}$ per Bq/year of discharge to the atmosphere)

An annual dose from gaseous emissions for each representative person can be calculated by summing the individual annual doses of each radioactive gas [65].

The resultant calculated doses are presented in section 6 and discussed in section 7.1.

5.2 Methodology for assessment of dose rates to non-human biota

There is no accepted national guidance (equivalent to the Environment Agency methodology for assessing dose to members of the public) for the assessment of dose rates to non-human biota.

A significant amount of work has been undertaken by international organisations and collaborative EC-funded projects regarding the assessment of radiological impacts on non-human biota, including the development of assessment tools. Results and tools from these projects have been used in assessments by RWM in the past, particularly the ERICA assessment tool [for example 74; 48]. This tool was an outcome of the EC 6th Framework project ERICA (completed in 2007) [40], but is now maintained independently and is freely available online. For the assessment of the dose from Rn-222 and its daughters, RWM has previously used an approach developed on behalf of the Environment Agency [75], combined with the ERICA tool. As discussed in section 3.3, the ERICA tool is compatible with the ICRP approach to protection of the environment developed concurrently.

At a high level, potential receptors and pathways followed by radionuclides from an initial gaseous discharge into the wider environment can be summarised as follows:

- Fauna – inhalation, ingestion (including water uptake), aerosol skin contact
- Flora – metabolic absorption (respiration, photosynthesis), uptake from soil, surface absorption

These receptors and pathways are considered in further detail in the following sections.

5.2.1 The ERICA approach and assessment tool

The ERICA approach and associated tool provides a complete assessment package, including elements related to environmental management, risk characterisation and impact assessment, as well as dose calculation. Full details of the approach are provided in the D-ERICA summary report [40], which also acts as a handbook for the tool. Comprehensive help documentation within the tool provides further (version-specific) guidance, assisting the user in making appropriate choices and inputs, and interpreting the outputs. This assessment uses version 1.2 of the ERICA tool, released in 2014 and incorporating major updates⁹.

The ERICA tool allows assessments to be undertaken at three tiers, each successive tier giving the user more control over the assessment. Tier 1 is designed to be simple and conservative, requiring a minimum of data input by the user and making extensive use of default data and options, whereas tiers 2 and 3 allow more user interaction. Screening criteria are included in tiers 1 and 2 that allow the user to exit the assessment process while being confident that effects on biota are low or negligible, and that no further action is required. Where effects cannot be shown to be negligible, or there are other reasons for carrying out a more detailed assessment, the assessment may continue to a higher tier. At tier 1, results are generated in the form of risk quotients¹⁰, while at tiers 2 and 3 both risk quotients and dose rates are reported. A tier 2 assessment is used in this generic OESA, in order to determine dose rates that can be reported and directly compared with the PNEDR values (and, where they are more stringent, DCRL values) set out in section 3.3.

5.2.2 Reference organisms

Assessments of the potential impact of radiation on non-human biota are undertaken by identifying the nature of the ecosystem (ERICA offers terrestrial, marine or freshwater ecosystems, consistent with those considered in the ICRP approach) and a range of characteristic 'reference organisms' within it [16]. The reference organism approach is analogous to the representative person approach applied for human radiological protection (section 5.1.1). A range of such organisms is generally chosen to encompass different trophic levels¹¹ and variations in lifestyle, geometry and uptake characteristics that determine exposure. As discussed in section 3.3, the ICRP framework uses RAPs, which are encompassed by and comparable to the reference organisms used in ERICA; the geometries of the ICRP RAPs are used as defaults for some reference organisms in the ERICA tool, and the same methodology has been used to develop the databases.

In the ERICA tool, the range of default reference organisms used depends on the ecosystem selected (freshwater, marine or terrestrial). For an assessment of atmospheric discharges from the GDF, a terrestrial ecosystem is most relevant, and for a terrestrial ecosystem the ERICA default reference organisms are as follows (equivalent ICRP RAPs are indicated in italics):

⁹ A subsequent version of the ERICA tool, v.1.2.1, was released in February 2016, but included only minor updates applicable to radionuclides and ecosystems not relevant to this assessment.

¹⁰ A risk quotient is defined in ERICA as: "*A measure of the risk calculated by each contaminant to an organism. For radioactive substances it is defined by the activity concentration of a given radionuclide in soil, water or air divided by the environmental media concentration limit [i.e. the environmental concentration of a radionuclide which would give rise to a dose rate of concern] for that radionuclide*" [40].

¹¹ The term 'trophic levels' refers to a group of organisms that occupy the same position in the food chain.

- Amphibian
- Annelid
- Arthropod – detritivorous (*worm*)
- Bird
- Flying insects (*bee*)
- Grasses and herbs (*grass*)
- Lichens and bryophytes
- Mammal – large (*deer*)
- Mammal – small burrowing (*rat*)
- Mollusc – gastropod
- Reptile
- Shrub
- Tree (*pine tree*)

A database of default values for the attributes necessary to assess dose to these reference organisms, such as occupancy factors¹², geometry, radioecology parameters (for example concentration ratio) and dose conversion coefficients, is built into the ERICA tool. These can be edited where site-specific information is available, but for this generic assessment the default values for all attributes were applied.

It is noted that impacts on sensitive species, indicator species, and endangered or locally important species may not be fully represented by assessments undertaken for these reference organisms. Should such species be relevant at a specific site, additional assessments (and/or specific dose rate criteria) may be necessary. Tiers 2 and 3 of the ERICA tool allow users to exclude reference organisms from the assessment and/or define additional reference organisms, thus more accurately representing a specific biota. However, for this generic OESA, no site-specific information is available and the default list is considered to be sufficiently representative of a terrestrial ecosystem.

5.2.3 Assessment of dose rates to non-human biota from tritium and C-14 using ERICA

The ERICA tool has an in-built air dispersion model that allows doses from tritium and C-14 to be calculated directly from the peak radionuclide release rates presented in section 4. The model is based on the IAEA Gaussian plume model applied to assess the dispersion of long-term atmospheric releases [76]. This model is widely accepted for use in radiological assessment activities, and is considered appropriate for representing the dispersion of either continuous or long-term intermittent releases within a distance of a few kilometres of the source.

In addition to the release rates for the selected radionuclides, ERICA requires a number of other parameters in order to calculate environmental concentrations of radionuclides, which are listed below. For this generic assessment, the default values included within the ERICA tool have been used where there is no justification for using alternative values.

¹² Occupancy factors describe the fraction of time spent by an organism at different locations in its given habitat (in a terrestrial ecosystem, these are on soil, in soil and in air).

- Release height: in this assessment, it is assumed that all radioactive gases from the GDF during the operational period will be released from an effective release height of 15 metres (as for the assessment of dose to members of the public)
- Distance to the receptor: in this assessment, the receptor is assumed to be located 300 metres from the discharge point, for consistency with the 2010 generic OESA; this is considered to be both representative and reasonably conservative for a scoping assessment, although it is noted that more conservative assumptions could be made [48] (the most appropriate receptor distance for assessments of dose to non-human biota in subsequent OESAs will be determined based on site-specific conditions and designs)
- Wind speed: in this assessment, the ERICA default value of 2 m/s is used
- Fraction of time wind blows towards the receptor: in this assessment, the ERICA default value of 0.25 is used
- Surface soil density: in this assessment, the ERICA default value (for soils other than peat) of 260 kg/m² is used
- Duration of discharge: in this assessment, the duration of discharge is assumed to be equal to the GDF operational period. This is assumed to be 160 years (2040 AD to 2200 AD, including a nominal 10-year closure period)

The calculation of radionuclide concentrations in biota and the resultant dose rates requires additional information, including concentration ratios and dose conversion coefficients. For this generic assessment, the default values for the terrestrial environment and the reference organisms included within ERICA have been used. These are discussed in the Data Report [17, section 3.10.4].

Calculated dose rates are presented in section 6.2.

5.2.4 Assessment of dose rates to non-human biota from Rn-222

The ERICA tool does not include the necessary parameters to undertake dose assessments for Rn-222, and a different method must therefore be used.

In the 2010 generic OESA [12], dose rates from Rn-222 and its daughters were derived using the generic methodology set out in [75]. Details of its application to the OESA are provided in [48]. In this approach, dose per unit concentration values were derived for each reference organism and combined with atmospheric concentrations of Rn-222 (derived from modelling using the software PC-CREAM 08) to give dose rates to non-human biota located 300 metres from the release point resulting from a release from an effective release height of 15 metres.

The atmospheric concentrations, and therefore the dose rates, derived in this way are a function of the release rate of Rn-222 assumed in the 2010 generic OESA. Although the Rn-222 release rate has been updated in this generic OESA (section 4), the methodology for determining dose rates from release rates remains valid and updated dose rates can be calculated by scaling the previously derived dose rates to the new release rate of Rn-222. The first stage of this calculation is the derivation of dose rates per unit release of Rn-222 for each reference organism from the information presented in [12] and [48]; these are listed in Table 12.

Table 12 Dose rates per unit release of radon, calculated from dose rates in Table 9 of [12, Table 8; 48]

Reference organism	Dose rate ($\mu\text{Gy/h}$) per unit release of Rn-222 of 1 Bq/s continuously per year from an effective release height of 15 m
Amphibian	Not determined
Annelid ¹	4.18×10^{-8}
Arthropod – detritivorous ²	8.42×10^{-8}
Bird	1.17×10^{-8}
Flying insects	4.94×10^{-8}
Grasses and herbs	2.29×10^{-7}
Lichens and bryophytes	2.29×10^{-7}
Mammal – large ³	1.14×10^{-8}
Mammal – small burrowing ⁴	2.73×10^{-8}
Mollusc – gastropod ⁵	Not determined
Reptile	1.09×10^{-8}
Shrub	2.29×10^{-7}
Tree	2.29×10^{-7}

1. In the 2010 generic OESA, this organism was called 'Soil invertebrate', reflecting terminology used in an earlier version of ERICA (the same is true for the following organisms)
2. Previously 'Detritivorous invertebrate'
3. Previously 'Mammal (deer)'
4. Previously 'Mammal (rat)'
5. Previously 'Gastropod'

Dose rates can then be calculated by multiplying the values in Table 12 by the Rn-222 release rate derived in section 4. The resultant calculated doses, arising from the decay of Rn-222 and including contributions from its daughters, are presented in section 6.2.

6 Quantitative Assessment Results

This section reports doses from gaseous emissions to the atmosphere as calculated by applying the methodologies presented in section 5 to the rates of emissions of GDF-derived radioactive gases presented in section 4. Two endpoint receptors, the public (section 6.1) and non-human biota (section 6.2), are considered. The results are discussed further, in the context of safety analysis, in section 7.1.

6.1 Calculated doses to representative public: local resident family

Table 13 presents doses calculated using the peak gas release rates for the operational period base scenario to the local resident family receptor group (see section 5.1.1).

It can be seen from Table 13 that, for each radioactive gas, the highest annual dose across all pathways is experienced by the infant age group of the local resident family. The discussion below therefore relates to the doses received by this age group.

The maximum calculated annual dose from gaseous emissions to members of the public during the operational period (assuming the base scenario gas release scenario) is 0.17 mSv per year. The majority of this arises from Rn-222 (0.11 mSv per year), with minor contributions from C-14-bearing carbon monoxide (0.052 mSv per year) and tritium (~0.01 mSv per year) and a negligible contribution from C-14-bearing methane. Almost all of the Rn-222 contribution is received via the inhalation pathway.

It is noted that these peak annual doses are hypothetical and conservative, since they are based on peak release rates that will, in practice, occur at different times for each radioactive gas, rather than being coincident (section 4.5). They also assume that the resident family is located at the same place all the time, which is not likely in practice. Additionally, it is noted that peak doses from each gas will only be received over a short period (a few years, coincident with the peak release of each radioactive gas) during the operational period of the GDF. For the remainder of the time, doses are expected to be significantly lower.

A semi-quantitative consideration of the two major contributors to dose (Rn-222 and C-14-bearing carbon monoxide) illustrates the conservative nature of this assessment. Figure 3 shows that the peak release rate (and therefore dose, which linearly scales with release rate) of C-14-bearing carbon monoxide occurs at 2064 AD but is approximately the same from 2040 AD until ~2100 AD, when it decreases by a factor of ~4. Conversely, Figure 5 shows that the release rate (and dose) from Rn-222 up to 2100 AD is less than or equal to ~75% of the peak. Using these observations and the peak doses reported above, to a first approximation, the annual dose up to 2100 AD will be on the order of 0.05 mSv per year from C-14-bearing carbon monoxide and 0.08 mSv per year from Rn-222 (a sum of 0.13 mSv per year). The annual dose from 2100 AD (excluding the backfilling period) will be on the order of 0.01 mSv per year from C-14-bearing carbon monoxide and 0.11 mSv per year from Rn-222 (a sum of 0.12 mSv per year). For the 10-year backfilling period, the release rates of both C-14-bearing carbon monoxide and Rn-222 are close to their peak, so the dose from gaseous emissions is likely to be closer to the peak calculated dose of 0.17 mSv per year.

These conservatisms, and those discussed in section 7.1, should be taken into account in comparing illustrative doses calculated for a generic site with regulatory limits and guidance levels. The calculated illustrative dose from gaseous emissions of 0.17 mSv per year is below the legal dose limit for members of the public of 1 mSv per year (Table 1), but exceeds the source-related dose constraint to members of the public from a new facility of 0.15 mSv per year adopted by RWM as described in Section 3.2.3. The dose from gaseous emissions, as well as the individual doses from C-14-bearing carbon monoxide and Rn-222, also exceeds the Basic Safety Objective (BSO) of 0.02 mSv per year.

Table 13 Calculated annual doses from gaseous emissions to the local resident family receptor group using peak gas release rates for the operational period base scenario

Radioactive gas	Discharge (Bq per year)	Age group	Food dose (mSv per year)	External dose (mSv per year)	Inhalation dose (mSv per year)	All pathways dose (mSv per year)
Tritium	2.4×10^{13}	Offspring	6.0×10^{-3}	0	3.0×10^{-3}	9.1×10^{-3}
		Infant	8.5×10^{-3}	0	1.3×10^{-3}	9.8×10^{-3}
		Child	4.3×10^{-3}	0	1.9×10^{-3}	6.2×10^{-3}
		Adult	4.1×10^{-3}	0	2.0×10^{-3}	6.1×10^{-3}
C-14-bearing methane	1.2×10^{10}	Infant	DPUR accounts for all pathways, therefore dose not calculated for individual pathways			1.8×10^{-7}
		Child				1.0×10^{-7}
		Adult				1.1×10^{-7}
C-14-bearing carbon monoxide	4.3×10^{11}	Infant	DPUR accounts for all pathways, therefore dose not calculated for individual pathways			5.2×10^{-2}
		Child				3.1×10^{-2}
		Adult				3.1×10^{-2}
Rn-222	4.9×10^{12}	Offspring	0	4.2×10^{-7}	0	4.2×10^{-7}
		Infant	0	2.0×10^{-7}	1.1×10^{-1}	1.1×10^{-1}
		Child	0	2.5×10^{-7}	8.4×10^{-2}	8.4×10^{-2}
		Adult	0	4.2×10^{-7}	5.7×10^{-2}	5.7×10^{-2}
All gases		Infant	Not calculated for individual pathways			1.7×10^{-1}
		Child				1.2×10^{-1}
		Adult				9.5×10^{-2}

Table 14 presents doses calculated using the bounding case gas release scenario explained in section 4.6. As doses to the infant age group have been shown to be bounding, only doses to infants are included in the table. Again, the conservatism noted above and discussed in section 7.1 should be taken into account when comparing these illustrative calculated doses with regulatory criteria.

Table 14 Calculated annual doses from gaseous emissions to the local resident family receptor group (infant age group) using peak gas release rates for the operational period bounding case

Radioactive gas	Discharge (Bq per year)	Age group	Food dose (mSv per year)	External dose (mSv per year)	Inhalation dose (mSv per year)	All pathways dose (mSv per year)
Tritium	3.6×10^{13}	Infant	1.3×10^{-2}	0	1.9×10^{-3}	1.5×10^{-2}
C-14-bearing methane	1.2×10^{10}	Infant	DPUR accounts for all pathways, therefore dose not calculated for individual pathways			1.8×10^{-7}
C-14-bearing carbon monoxide	1.3×10^{12}	Infant	DPUR accounts for all pathways, therefore dose not calculated for individual pathways			1.6×10^{-1}
Rn-222	4.9×10^{12}	Infant	0	2.0×10^{-7}	1.1×10^{-1}	1.1×10^{-1}
All gases		Infant	Not calculated for individual pathways			2.8×10^{-1}

The maximum resulting dose to members of the public is calculated in this scenario to be 0.28 mSv per year. The contribution from Rn-222 is unchanged from the base scenario at 0.11 mSv; the additional dose arises from tritium and C-14-bearing carbon monoxide. This dose from gaseous emissions is below the effective dose limit for members of the public of 1 mSv per year. However, it is above the source-related dose constraint to members of the public from a new facility of 0.15 mSv per year adopted by RWM, and significantly exceeds the BSO of 0.02 mSv per year. As for the base scenario, these calculations are illustrative at this stage and incorporate significant conservatism, as well as representing a bounding case that is not likely to reflect reality.

6.2 Calculated dose rates to terrestrial reference organisms

Table 15 presents dose rates to non-human biota arising from peak gas releases from the GDF during the operational period according to the bounding case, using the methodology described in section 5.2.

Table 15 Calculated dose rates from gaseous emissions to the non-human biota receptor group using peak gas release rates in the bounding case

Organism	Dose rate (μ Gy per hour)			
	Tritium	C-14	Rn-222	All gases
Amphibian	3.62×10^{-2}	4.09×10^{-2}	Not determined	7.71×10^{-2}
Annelid	3.62×10^{-2}	1.31×10^{-2}	6.51×10^{-3}	5.58×10^{-2}
Arthropod (detritivorous)	3.62×10^{-2}	1.31×10^{-2}	1.31×10^{-2}	6.24×10^{-2}

Organism	Dose rate (μGy per hour)			
	Tritium	C-14	Rn-222	All gases
Bird	3.62×10^{-2}	4.23×10^{-2}	1.83×10^{-3}	8.03×10^{-2}
Flying insects	3.39×10^{-2}	1.31×10^{-2}	7.69×10^{-3}	5.47×10^{-2}
Grasses and herbs	3.62×10^{-2}	2.71×10^{-2}	3.56×10^{-2}	9.89×10^{-2}
Lichens and bryophytes	3.61×10^{-2}	2.75×10^{-2}	3.56×10^{-2}	9.92×10^{-2}
Mammal – large	3.62×10^{-2}	4.23×10^{-2}	1.77×10^{-3}	8.03×10^{-2}
Mammal – small burrowing	3.62×10^{-2}	4.23×10^{-2}	4.25×10^{-3}	8.27×10^{-2}
Mollusc – gastropod	3.62×10^{-2}	1.31×10^{-2}	Not determined	4.93×10^{-2}
Reptile	3.62×10^{-2}	4.23×10^{-2}	1.70×10^{-3}	8.02×10^{-2}
Shrub	3.62×10^{-2}	2.71×10^{-2}	3.56×10^{-2}	9.89×10^{-2}
Tree	3.62×10^{-2}	4.11×10^{-2}	3.56×10^{-2}	1.13×10^{-1}

There are similar contributions to the calculated dose rate from the dose rate associated with tritium and C-14 for all species, with (in most cases) a smaller contribution from Rn-222. The highest dose rates are received by trees, followed by lichens and bryophytes, grasses and herbs, and shrubs. The lowest dose rates are received by molluscs, followed by flying insects and annelids.

Calculated dose rates from gaseous emissions for all organisms in the bounding case scenario are significantly below both the ERICA screening dose rate value of $10 \mu\text{Gy}$ per hour for a terrestrial ecosystem, which reflects PNEDR values derived as part of the ERICA project, and ICRP DCRLs for relevant RAPs (section 3.3).

It is noted that there are recognised uncertainties in dose assessments to non-human biota (for example, estimates of concentration ratio values ranging over four orders of magnitude). However, these bounding case dose rates have been calculated using current best practice methodology and conservative/best estimate input parameters; given that the highest calculated dose rate ($0.1 \mu\text{Gy}$ per hour to trees) is over an order of magnitude lower than the most stringent screening value (DCRL of $4 \mu\text{Gy}$ per hour), it is considered that the assessment of dose to non-human biota from gaseous emissions to the atmosphere from the GDF requires no further consideration in this report.

As noted earlier and in section 8.2, RWM is aware that further work on the assessment of doses to non-human biota is ongoing, and that best practice methodology and input parameters may change in the future. Any such changes will be incorporated into future updates of the OESA.

7 Safety Analysis and Discussion

This section:

- Discusses the quantitative assessment results for the gas pathway presented in section 6, including uncertainties, conservatisms and potential mitigation measures (section 7.1)
- Summarises qualitative supporting arguments for the operational environmental safety of the GDF (section 7.2)
- Summarises the role of monitoring during the operational period (section 7.2.1)

7.1 Analysis of quantitative assessment (gas pathway)

Annual doses to members of the public (local resident family receptor group) resulting from peak gaseous emissions to the atmosphere during the operational period in the base scenario and bounding case scenario are calculated respectively to be 0.17 mSv per year (dominated by Rn-222) and 0.28 mSv per year (dominated by C-14-bearing carbon monoxide and Rn-222) (see Table 13 and Table 14).

The following discussion identifies some of the conservatisms and uncertainties in both the conversion of gaseous releases to doses and the derivation of the gaseous source term, in particular as potentially affecting the rate of gas generation and release in the GDF operational period.

Also in this section, potential mitigation measures are identified that could be employed, should it be considered necessary when more detail about the facility design and potentially exposed groups becomes available at later stages of the implementation programme.

7.1.1 Uncertainties and potential conservatisms

Uncertainties and potential conservatisms are discussed first by topic then the site-related uncertainties that have been identified throughout the report across topics are collated.

Dose assessment methodology

For the assessment of doses to members of the public reported in section 6, the local family receptor group is assumed to be located 100 metres from the gas release point, consuming food grown at 500 metres from the gas release point. These assumed distances are very dependent on the site characteristics, but it should be noted that a receptor group living farther from the gas release point will receive a lower dose during the operational period (as any gaseous emissions will be subject to a greater degree of dispersion over the longer travel distance). This decrease in dose as a result of an increase in distance from the release point has been demonstrated in previous assessments [15]. Doses are also affected by assumptions of receptor group behaviour (including residency time, breathing rates and food consumption rates), and environmental factors such as average wind conditions; such factors are site-specific.

Doses are also strongly dependent on the effective release height, which, as noted earlier, is a function of both design-specific factors, including stack height and efflux parameters, and site-specific factors such as topography. Until a specific site and detailed GDF design are available, there is uncertainty regarding the effective release height. As discussed in section 4.3.2, this assessment has assumed an effective release height of 15 metres for consistency with previous operational environmental safety assessments [12, 48], but parallel operational safety work [49] supports changing the assumed effective release height to 30 metres, a value that has been robustly underpinned [50, Appendix B]. The

design-specific factors affecting effective release height can be readily adjusted in order to mitigate doses, and a scoping calculation assessing the effect on doses of assuming an effective release height of 30 metres is presented in section 7.1.2.

There are also uncertainties in the DPUR values for C-14-bearing gases, particularly for carbon monoxide, which are explained in detail in [73]. The uncertainty in the carbon monoxide DPUR arises largely because the uptake of carbon monoxide by plants (relevant to the ingestion pathway) is poorly constrained, and conservative assumptions have therefore been applied. Further work to reduce this uncertainty could be beneficial, as this assessment has shown that carbon monoxide releases are potentially significant contributors to dose.

Inventory

As noted in section 2.2, RWM treats uncertainties in the Derived Inventory through the consideration of inventory scenarios, summarised in the Technical Background. These scenarios can be grouped into those that lead to more of the same inventory, those that lead to less of the same inventory, and those that change waste package characteristics. The modular approach (determining the individual contribution of different waste types) adopted in the assessment of gas release rates presented in section 4 of this report, and to a greater level of detail in [53], allows the likely impact of these scenarios to be evaluated qualitatively.

Scenarios that lead to more or less of the same inventory only affect the peak doses presented in this generic OESA if they concern waste streams that make significant contributions to releases of relevant radionuclides. For example, if graphite were excluded from the inventory for disposal in the GDF, the peak release of C-14-bearing methane would decrease, but there would be a negligible effect on the more radiotoxic C-14-bearing carbon monoxide (because the main contributors to its release are corrosion of uranium and Magnox) and the overall dose would not be significantly reduced. In addition, uncertainties in the radionuclide inventories of those waste streams that make significant contributions to the peak release rates of each radionuclide will result in similar uncertainties in the peak release rates. This could be up to a factor of ten in some cases [53].

Scenarios that change waste package characteristics only have an effect on the OESA dose calculations if they alter the gas generation and release behaviour of waste packages during the operational period. For example, changes to the reactors used in the new build programme resulting in SF with different characteristics would have no effect because SF is assumed to be packaged in unvented containers from which no gas is released. However, alternative packaging assumptions resulting in a change to the encapsulation of waste packages containing radium could affect the rate of Rn-222 release. Thus, the change in packaging assumptions for depleted uranium (described in Section 4.4.4) is expected to result in an increase in Rn-222 release. The emanation coefficient for the affected streams increases from 0.14 to 1 (ie by a factor of ~7). Therefore the total Rn-222 release rate from DNLEU is expected to increase by less than a factor of 7.

Uncertainties in gas release rates affecting all gases

The effects of some major uncertainties in radioactive gas release rates have been explored in the variant scenarios reported in [53] and explained in section 4.5, although (to simplify the number of calculations and results presented) these have not been individually translated into doses. For example, if uranium corrosion is assumed to take place under aerobic rather than anaerobic conditions, peak release rates of tritium and C-14 are reduced and doses will also be reduced. A number of other uncertainties exist, and the effects of these are discussed below and in subsequent sub-sections.

It has been assumed in this assessment that gases released from waste packages will enter the GDF ventilation system and be discharged to the environment until the end of the operational period at 2200 AD. However, this may not be the case; for example, gases could react with other GDF components, such as engineered barriers, before entering the ventilation system. Additionally, as soon as backfilling has commenced, gases may be prevented from entering the ventilation system from the backfilled vaults. Accounting for this could reduce the release rates of some gases during the backfilling period, although it would not affect the peak release rates of gases as currently modelled in the base scenario as these occur before the start of the backfilling period. Inputs to the model (including corrosion rates) that result in calculated peak release rates occurring prior to the start of backfilling are discussed in [53]; however, if these corrosion rates are overly conservative and peak release rates do in fact occur during backfilling, then accounting for the possible prevention of gases being released to the ventilation system from backfilled vaults will result in lower peak doses. It is further noted that the base scenario represents a major conservatism for lower strength sedimentary rock and evaporite, because staged backfilling is already planned for these host rock types (as discussed in section 4.5).

Conversely, in this assessment the waste inventory in each waste group is assumed to be emplaced uniformly over the emplacement period. In practice, for many waste streams, emplacement will occur during a small fraction of the overall emplacement period. If the waste streams that provide the main contributions to the peak C-14 or tritium release rates are emplaced over a shorter period than assumed, the calculated peak C-14 or tritium release rate will increase. The overall scheduling of emplacement is also important, particularly for tritium; because of its relatively short half-life, delaying emplacement until later in the operational period would result in lower releases due to prior decay.

In addition, there are uncertainties in the temperature, pH and other conditions expected within waste packages during the operational period beyond those explored in the variant scenarios, which could have an effect on the release rates of radioactive gases. These are likely to vary according to site-specific factors, as well as over time in response to natural and operational processes.

C-14

Making the conservative assumption that all C-14 released from metals is in the form of carbon monoxide, the peak release rate of C-14-bearing carbon monoxide in the base scenario is estimated to be 0.43 TBq per year. This gives rise to a dose of 0.052 mSv per year using the approach noted and applied in sections 5 and 6. The peak release rate of C-14-bearing carbon monoxide in the bounding case (resulting from the higher temperature variant scenario) is 1.3 TBq per year, giving rise to a dose of 0.16 mSv per year. In both cases, the dose from C-14-bearing methane is negligible in comparison.

The C-14-bearing gas release rate calculations contain a number of pessimisms. In particular, there are significant uncertainties around how much C-14 will be released from metals as gas, and in what form. If all C-14 released from metals is assumed to be in the form of methane instead of carbon monoxide, the peak base scenario dose from carbon monoxide is reduced by over an order of magnitude to 1.4×10^{-3} mSv per year, while the dose from methane increases from 1.8×10^{-7} to 6.5×10^{-6} mSv per year and can still be considered negligible. Equivalent doses for the bounding case are 1.4×10^{-3} mSv per year for carbon monoxide and 2.0×10^{-5} mSv per year for methane. No sensitivity calculations have been performed to assess the effect on dose if not all C-14 in metals were to be released as a gas, but this assumption is considered to be conservative and it is possible that only a fraction may be released as gas [53]; in this case doses would be significantly lower.

Conversely, it has been assumed that C-14 is distributed uniformly throughout all the metal wastes. In practice, it could be concentrated near to the surface of some metals, which

could significantly increase the calculated peak C-14 release rates or could result in release prior to disposal at the GDF.

Rn-222

The calculated dose presented for Rn-222 in section 6 (0.11 mSv per year in both the base scenario and the bounding case) is dependent on the assumed emanation coefficients for wastes in different types of encapsulant presented in section 4.5.2. As a result of these revised assumptions regarding polymer encapsulated waste, the modelled release of Rn-222 from the GDF is now dominated by other streams containing Ra-226, primarily cement encapsulated legacy ILW (Figure 5). However, in practice there is expected to be a wide range of emanation coefficients for Rn-222 from cement encapsulated wastes (which will also vary with factors such as temperature), and the values presented are expected to be cautious [59]. Possible further work to improve understanding of radon emanation coefficients is set out in [59]. Further work could also be undertaken to better understand hold up of radon within the waste itself (for which no credit is currently taken) and hence the rate of off-site discharge. It is expected that such work would lead to reduced (more realistic) calculated doses due to Rn-222.

There is also significant uncertainty in many of the parameters underlying the indicative emanation rates of Rn-222 from the host rock (presented in Table 7), resulting in an uncertainty up to several orders of magnitude on the emanation rate values [63]. These uncertainties include:

- the uranium content of the host rock
- the thickness of the layer of rock, damaged by excavation and stress relief, from which Rn-222 is released
- the mass of solid rock (which depends upon its density), within this damaged layer
- the percentage of the generated radon that is actually released, noting that this release coefficient depends on a number of factors including the type of host rock, the nature and extent of any excavation damage and the presence or otherwise of lining, shotcrete, or similar

Site- and design-specific studies in the future are expected to yield more realistic results because these parameters can be measured or estimated with a much lower level of uncertainty than in the generic case. In particular, it is noted that the assumption that all excavated areas of the GDF will have exposed rock surfaces throughout the duration of the operational period is likely to be pessimistic. In practice, shotcrete lining in operational excavations will reduce the emanation of Rn-222, excavation will be phased so that the whole GDF will only be excavated towards the end of the operational period, and some underground areas may be closed before others are excavated. It is noted that both the parameters listed above and the way in which the GDF will be operated (in terms of lining and phased excavation) are highly specific to the host rock.

Tritium

Uncertainties in the modelling of tritium release from the GDF could affect the calculated doses for the base scenario (0.0098 mSv per year) and the bounding case (0.015 mSv per year) in either an upwards or a downwards direction.

As for C-14, it is conservatively assumed that all tritium will be released from metals as gas, whereas only a fraction may be gas in practice; the results are cautious in this respect and doses from tritium may be significantly lower in practice.

Conversely, it has been assumed that tritium is distributed uniformly throughout all the metal wastes. In practice, it could be concentrated near to the surface of some metals,

which could significantly increase the calculated peak tritium release rates or could result in release prior to disposal at the GDF.

It is possible that the peak tritium release rate would be higher if release from Magnox was assumed to be by diffusion rather than corrosion [53]. This was not calculated owing to lack of data, but a scoping calculation indicated the potential for release by diffusion to be important. There is also uncertainty in the diffusion rate of tritium from stainless steel (a key contributor to the release of tritium), but this is not expected to have a significant effect on dose [53].

The distribution of the tritium inventory detailed in Table 5 indicates that (for legacy ULLW/UIIW) a significant inventory of tritium (about 15% of that in legacy ULLW/UIIW) is present in waste materials other than metals and graphite. It is anticipated that tritium release from these other waste materials will not provide a significant contribution to the overall tritium release rate from the waste. However, it is possible that (some of) this tritium could be released at a significant rate. An indication of the potential significance of this inventory can be obtained by comparing the tritium inventory in these materials with the calculated tritium release rates for other materials. By this method it is estimated that such releases could potentially increase the calculated peak tritium release rate by up to ~20 TBq per year, although this is considered unlikely [53]. In this extreme case, peak doses to members of the public would be 0.018 mSv per year for the base scenario and 0.023 mSv per year for the bounding case (that is, around the BSO).

The short half-life of tritium also means that the timing of disposal in relation to waste arising will have an effect on releases and therefore doses, as noted above.

Site-related uncertainties

Several of the uncertainties discussed above and elsewhere in this report result from the fact that a site for the GDF has not yet been identified. They include:

- the host rock, which has implications for natural radon emanation from underground excavations and backfilling strategy
- the surface and groundwater regime, which will significantly influence the extent and nature of both radioactive and non-radioactive liquid effluents
- the location and behaviour of the nearest human residents, and local populations of non-human biota, which will affect the assumptions made in assessing dose to these receptors
- local topography and site layout, which will control the effective release height and the minimum distance of the receptors to the gas release point
- other environmental factors, such as average wind conditions

Many of these uncertainties can be mitigated through the design of the GDF, as discussed in the following section.

7.1.2 Mitigation measures

Additional steps could be taken to reduce the potential for gaseous discharges from the GDF and the off-site doses associated with them.

One of the key mitigation measures relates to the packaging of the wasteforms before they are dispatched to the GDF. Wastes that are accepted for disposal at the GDF must be treated, conditioned or packaged in such a way as to render them:

- Passively safe, such that they can be stored safely with the minimum need for actively managed safety systems, monitoring or prompt human intervention

- Capable of safe handling during storage, transport and emplacement in the GDF
- 'Disposable', so that they can be shown to be compliant with all the relevant regulations and safety cases for transport to and disposal in the GDF [77]

This is undertaken through RWM's Disposability Assessment process, which exists to minimise the risk that waste packages will not be suitable for disposal by assessing packaging proposals against the existing safety case envelope [77]. There is potential to develop specific packaging solutions for waste streams that make a large contribution to the gases of concern in this assessment. In accordance with RWM's safety integrated design process, as set out in RWM's Engineering Design Manual, any significant issues identified in the development of the safety case (including this generic OESA) can be discussed with the waste packagers, to ensure that appropriate mitigation measures can be put in place prior to dispatching the waste to the GDF. Packaging advice at an individual level is not expected to have a significant impact on future OESA results, with the exception of high radium-bearing streams, which are the most significant contributors to dose. Doses from such packages can be reduced by encapsulation in polymer, as discussed in [59]; this advice has already been incorporated into RWM's Waste Package Specification and Guidance Documentation on the packaging of radon generating wastes [78].

In this assessment, the radioactive discharges are assumed to be dispersed from an effective release height of 15 metres, which, as discussed earlier, is now considered to be unrealistically conservative. Increasing the effective release height will increase the dispersion of any gaseous discharges prior to them reaching ground level, and hence reduce the estimated doses to members of the public. This can be readily achieved either by increasing the physical stack height or by tuning the efflux parameters (or a combination). Using [65, Table 2], an effective release height of 30 metres gives a scaling factor for the food exposure pathway of 0.18 and for the inhalation and external exposure pathways of 0.015, which leads to the doses from tritium and Rn-222 shown in Table 16. The reduction in dose from Rn-222 by an order of magnitude is particularly significant, as this is the highest contributor to dose from gaseous emissions. A similar reduction is expected in the dose from C-14-bearing carbon monoxide, although this has not been quantified here as adjusting the updated DPURs would require the different exposure pathways to be separated. It is clear, however, that increasing the effective release height would result in a significant reduction to public dose. Therefore, further work to undertake a full assessment of public dose using a 30 metre effective release height would be of benefit and would remove an unrealistic conservatism in the assessment of dose resulting from gaseous emissions from the GDF. The actual physical stack height and efflux parameters (both controls on effective release height) will be determined through application of BAT and optimisation at the time of site-specific design development.

Table 16 Calculated annual doses to the local resident family receptor group (infant age group) from gaseous emissions from an effective release height of 30 metres

Radioactive gas	All pathways dose (base scenario, mSv /year)	All pathways dose (bounding case, mSv/year)
Tritium	3.6×10^{-3}	5.4×10^{-3}
Rn-222	1.8×10^{-2}	1.8×10^{-2}

Increasing the minimum distance from the discharge point to members of the public (assumed in this assessment and in the Environment Agency methodology to be 100 metres) will also result in reduced public doses. This could be readily achieved

through GDF layout design, for example by placing the discharge stack near the centre of the facility or extending the boundary fence. As for effective release height, such aspects will be determined through application of BAT and optimisation at the time of site-specific design development.

As explained in section 4.5, the base scenario gas release rates used in this assessment assume that the backfilling of the GDF vaults happens within a ten-year period, once all the waste is emplaced and just prior to closing the facility. However, the staged backfilling variant scenarios indicate that significantly lower peak release rates of Rn-222 and C-14 (the main contributors to dose from gaseous emissions) could be achieved by an alternative backfilling strategy, such as backfilling each vault as soon as emplacement is complete for that vault. It is noted that this reduction only occurs if it is assumed that gases generated in backfilled vaults are not released into the GDF ventilation system but are retained within the backfill as free gas or dissolved in porewater. It is further noted that staged backfilling is already planned for lower strength sedimentary rock and evaporite, and is only a variant scenario for higher strength rock.

7.2 Qualitative safety arguments

RWM's high-level strategy for ensuring operational environmental safety is to eliminate hazards during normal operation of the GDF and, where this is not possible, to provide protection to control environmental impacts. The generic requirements on the GDF to meet this safety strategy are set out in the Disposal System Specification – Part B [21]. The application of safety management systems, encompassing sound operating procedures and the use of suitably trained, qualified and experienced staff, also has a major part to play in ensuring high standards of safety. The following paragraphs present examples of supporting evidence to build confidence in this generic OESA.

The wastes will be packaged before transport to the GDF in accordance with detailed waste package specifications. Strict quality assurance requirements are imposed to ensure these packaging specifications are met. Furthermore, all waste packages will be assessed through a waste acceptance process prior to dispatch and acceptance into the GDF.

The packaged waste has a number of inherent safety features. The wastes are solid, or have been solidified before transport to the GDF. Most wastes will have been encapsulated in a stable matrix¹³. The wastes do not contain liquids or pressurised gases, and are packaged in appropriate containers. Any wastes that are expected to evolve significant gas (for example Magnox metal) will be packaged in containers having filtered vents to prevent the containers from becoming pressurised. This limits the potential for dispersal of radioactive material in the event of an accident but can lead to the release of radioactive gases as assessed in this report.

The most hazardous waste packages, including HLW/SF and some types of ILW, will be transported within reusable shielded transport containers that comply with the performance requirements for a 'Type B' transport container, as defined in the IAEA transport regulations [79]. These regulations stipulate that a Type B transport container must remain intact even under conditions of severe impact, fire or immersion. Therefore, no significant releases of radioactivity during their normal transport and handling are expected. The unshielded waste packages will not be removed from their Type B transport containers until after they have been taken underground.

¹³ Spent fuel – if declared a waste – is unlikely to be encapsulated in a matrix, as the radioactivity is already present in the form of a stable oxide fuel pellet. Additionally, waste in RSCs may not be encapsulated as these containers are designed to provide all safety functions required for transport and disposal without the need for encapsulation.

Less hazardous ILW/LLW will be packaged mainly in 'Type 2 industrial packages', again as defined in the IAEA transport regulations. The Type 2 industrial packages currently in use are large steel, concrete-lined boxes that are suitable for both transporting the waste and disposing of it. The safety of these packages during transport and handling is provided largely by regulatory limitations on the quantities of radionuclides that can be placed in the container. Such packages require a letter of compliance as they also serve as disposal packages. Radioactive gases may be released during buffer storage through filtered vents in the packages, and will be fed through the ventilation system and released to the atmosphere from the discharge stack. However, there will only ever be a relatively small number of such packages in the surface facilities at any one time, so only minor environmental discharges are expected at any time, which will be monitored. Such releases are not included in the calculation of releases from emplaced wastes in section 4.5, but as noted in section 4.3.1, they are expected to be insignificant in comparison.

Monitoring of wastes after arrival on site will be carried out in an enclosed building. This will detect any faulty transport packages so that they can be quickly isolated and appropriate remedial action taken. Any liquid effluents, such as rainwater on the surface of transport containers, will be collected and monitored for radioactive content. Any radioactive effluents will be treated so that any off-site discharges of radioactivity are within the discharge authorisation.

Waste will be transported underground using an inclined access tunnel (drift) or, possibly, a shaft. The Type B reusable transport containers and the Type 2 industrial packages will continue to contain the wastes during transfer from the surface facilities to the underground facilities. The only potential for releases off site during normal operations is radioactive gases from any vented Type 2 industrial packages. Again, there will only ever be a relatively small number of such packages being transferred underground at any one time, so only minor environmental discharges are expected, which will be monitored.

There will be potential for public exposure from off-site release of radioactive particulates as packages are lifted from their transport containers. However, as noted in section 4.3, such releases are expected to be minimal and will be reduced by use of HEPA filters at any discharge points of the underground ventilation system. There will also be potential for release of any radioactive gases that may have built up inside transport containers. These gases cannot be removed by HEPA filters, but will be monitored and could be mitigated if required as a result of fault analysis. The unloading and emplacement of other types of waste package will be undertaken using approaches specific to waste type and packaging. In particular, there may be less potential for release of radioactive particulates during the emplacement of Type 2 industrial packages in disposal vaults as there will be no need to unload them from transport containers.

Maintenance activities will have the potential to generate effluents and off-site discharges. Protection of the public and the environment will be achieved by carrying out such work in a purpose-built structure with suitable effluent collection and monitoring systems.

The design and operation of the GDF will ensure that environmental conditions in the underground facilities during the operational period are maintained so as to ensure that the waste packages remain in as good a condition as possible prior to GDF closure. However, if the GDF is developed in higher strength rock, there may be an extended period when the disposal areas are kept open prior to backfilling (for example to provide for ease of retrievability). In this case, significant degradation of some waste packages could occur, such that package remediation might be necessary if RWM were required to retrieve or otherwise move the wastes. Facilities will be maintained for package remediation and, as for other kinds of maintenance activities, the work will be carried out in a purpose-built structure with suitable effluent collection and monitoring systems.

Monitoring of the underground facilities during the operational period will detect any emissions or leakages of radioactivity. Any liquids will be collected, examined for radioactivity, and remediated if necessary in the surface facilities. Radioactive particulates in the air will be trapped by the use of HEPA filters at discharge points of the ventilation system. RWM's monitoring programmes will allow informed decisions to be made about management of discharges and the waste packages in the disposal areas. However, it may not be practicable to prevent the environmental discharge of radioactive gases from vented ILW/LLW containers that may build up over time in the disposal areas. In addition, radon gas from naturally occurring sources of radioactivity in the host rock may enter the underground facilities. For these reasons, the quantitative assessment of operational environmental safety in this generic OESA focuses on the potential for (and environmental safety implications of) discharges of radioactive gases from the underground facilities, recognising that any radioactive gas emissions to the atmosphere will be in compliance with permit limits.

Many of the activities described above are similar to those carried out at existing nuclear facilities in the UK for which there are proven standards and management systems and many years of good operating experience:

- UK waste stores, for example for vitrified HLW at Sellafield and ILW stores around the country, are fitted with handling, monitoring, filtration and fire protection systems similar to those required for the GDF
- LLW is safely transported to and disposed of at the near-surface disposal facility, the Low Level Waste Repository near the village of Drigg in West Cumbria, using purpose-built containers that must meet the IAEA transport regulations for a Type 2 industrial package; these are similar in design to the Type 2 industrial packages planned for ILW/LLW transport to the GDF
- The IAEA has produced systematic and comprehensive information on gaseous radioactive waste management systems, providing extensive guidance and best practice examples [80], including the UK's Thermal Oxide Reprocessing Plant; other UK facilities, such as the Magnox Encapsulation Plant at Sellafield, also have significant operating experience of managing releases of contaminated gas
- Many operating facilities (nuclear and non-nuclear) in the UK and internationally have radioactive waste management arrangements in place to manage, abate and monitor solid, liquid and gaseous radioactive emissions to the environment to demonstrate regulatory compliance
- Naturally occurring radon emanating from the host rock is a common issue that has been and is currently successfully managed in mines and underground civil engineering projects in the UK and internationally

7.2.1 Monitoring

Any operational releases of radioactivity to the environment from the GDF will require permitting under the RSR regime. Demonstrating compliance with regulatory limits through the application of BAT to monitoring will be a requirement of such a permit. In addition, the GRA requires that the developer or operator of a disposal facility for solid radioactive waste carries out a programme of site investigation and site characterisation to provide baseline information for the environmental safety case, and subsequently monitors for changes caused by construction, operation and closure of the facility [33].

Although a specific site has not yet been identified for the GDF, RWM is considering what monitoring may be needed to meet these requirements and is working towards a coherent monitoring strategy. It is anticipated that a significant amount of monitoring will be required prior to construction of the GDF to inform the development of a site-specific DSSC, including components such as the OESA.

Once a site for the GDF has been selected, site-specific monitoring activities can be defined. After the construction of the GDF, monitoring will be undertaken during the operational period to establish the levels of the discharges from the facility and demonstrate compliance with the regulatory limits set out in section 3.

The monitoring programme for the operational period will include the definition of trigger values of the monitored parameters beyond which action will be taken, and action plans to investigate, control, mitigate and/or remediate (as necessary) any releases approaching the regulatory limits.

8 Future Work

8.1 Consideration of the Derived Inventory in future OESA assessments

The Derived Inventory will continue to be developed with information obtained from waste producers and the Disposability Assessment process. RWM will continue to improve its understanding and address assumptions about gas generation and release from all the packaged waste and materials in the Derived Inventory. Once a site for the GDF is identified, a site-specific OESA will be produced that is based on the then-current UK radioactive waste inventory.

8.2 Future research

RWM has recognised the need to determine in detail the short term dynamics (relevant to the operational period) of gas evolution from waste packages, and to accurately assess dose to members of the public and non-human biota arising from such gas releases. Significant work in these areas has been undertaken by RWM since the 2010 generic OESA was published, but work is still ongoing. Some of the work in progress or planned is noted as follows:

- The assumptions discussed in this report relating to gas generation and release rates of C-14 are being investigated through tasks described in the Science and Technology Plan [81]. Tasks 201 – 213 encompass studies on release from irradiated metals, Tasks 226 – 231 encompass similar studies on irradiated graphite and Tasks 241 – 242 examine releases from other studies. System modelling of C-14 is addressed by Tasks 251 – 253.
- RWM plans to investigate further the role of alternative encapsulants for radium-bearing wastes (for example polymers) in delaying the release of Rn-222 from waste packages [81, Task 261].
- Work has been undertaken to develop the methodology for assessing non-radioactive discharges from the GDF [82]. This work will be drawn upon and developed to support quantitative assessments of such discharges.
- RWM will continue to play an active role in the BIOPROTA, MODARIA and NERC-TREE programmes (discussed in section 2.1.1) and their successors, through which work will continue to be undertaken on the effects of radionuclide releases on the biosphere. Methods for assessing dose to non-human biota, and relevant assessment criteria, will be kept under review. The areas of research are described in [81, Tasks 11 – 20].
- RWM will consider the case for further work noted in this report as being potentially beneficial. This includes work to:
 - fully quantify the effect on doses to members of the public of increasing the effective release height to a more realistic value of 30 metres
 - constrain DPURs for C-14-bearing gases, especially carbon monoxide
 - improve understanding of radon emanation coefficients from excavated surfaces within the GDF
 - derive a bounding source term for radioactive liquid effluent releases and off-site doses arising from them

8.3 Further work on non-radioactive discharges

During the GDF operational period, there will be activities that have the potential for off-site effects, other than those arising from the wastes. For example, machinery, both at the surface and underground, will generate waste lubricants and solids, and combustion engine vehicles will generate exhaust fumes. In addition, ventilation air may be discharged at non-ambient temperature and/or humidity. These discharges will be assessed to determine the local consequences and will be subject to optioneering with respect to the 'carbon footprint' of the facility. Such considerations form part of both the Operational Safety Case [4] and environmental assessments [51; 52].

Further work will be undertaken to develop the work that has already been reported on the assessment of non-radioactive discharges, such as any chemically hazardous gases released from the waste. This work will be developed in line with the SEA and EIA reports to ensure appropriate consistency.

8.4 Concept development

The set of assumptions used for the generic illustrative assessment presented in this report is based on the illustrative disposal concepts for the design of the GDF described in [46]. Assumptions have been made regarding the management of groundwater intrusion into areas for the receipt of the wastes, and regarding control measures such as filtration. There are uncertainties in these assumptions. However, as RWM moves towards site selection, these assumptions will become site-specific and assessments can then be refined accordingly.

In accordance with RWM's safety integrated design process, as set out in RWM's Engineering Design Manual, any significant issues identified in the development of the safety case (including this generic OESA) can be discussed with the waste packagers, to ensure that appropriate mitigation measures can be put in place prior to dispatching the waste to the GDF. Proposed changes are controlled by RWM's change management procedure, which ensures that the changes are appropriately underpinned and their potential significance to safety or the environment is assessed before implementation.

The assessment of the impact of gaseous emissions to the atmosphere will continue to be reviewed as further details of the site and disposal facility become available. The results of this assessment will be considered alongside the results of on-site radiological assessments, to ensure that doses to both members of the public and workers are as low as reasonably achievable (ALARA)¹⁴. The results of assessments are used in the development of the disposal system concept as described above. Consideration through the management arrangements described will ensure that, by making doses to workers ALARA, they do not compromise the off-site doses, and vice-versa.

Work is being undertaken to consider alternative options for the implementation of geological disposal; radiological protection is a consideration in these optioneering studies. Detailed optioneering studies will be carried out at the appropriate time (when more details are known about the actual systems and processes once a site is identified), and will need to demonstrate that BAT and optimisation is applied (in accordance with RSR requirements) to ensure that public and environmental impacts, as well as doses to workers, are ALARA.

¹⁴ The term 'ALARA' is used here in preference to the equivalent 'as low as reasonably practicable (ALARP)' normally used to describe doses to workers, in order to reflect the terminology used by the environmental regulators.

8.5 Future iterations of the OESA

As discussed above in this section, further work will be undertaken that is relevant to issues covered in the OESA, and this assessment will subsequently be iteratively updated and re-published.

9 Summary

This section summarises the assessments undertaken, the results and conclusions with respect to the operational environmental safety of the GDF.

RWM is aware of the need to ensure active management of gas in the operational phase of the GDF, through the application of BAT and the principle of optimisation, to ensure that regulatory dose constraints are met and doses to the public are minimised. This management will include assessment of anticipated releases; this generic OESA has explained RWM's assessment approach and described the methodology for:

- Assessing the non-radioactive discharges from the GDF
- Assessing the discharges from the GDF both for surface facilities and those underground
- Calculating doses to members of the public located off site from the GDF
- Calculating doses to non-human biota located off site from the GDF

Parts of this assessment approach have been illustrated by example at this generic stage. As a result, RWM has high confidence that doses to members of the public and the environment from the GDF can be modelled, and that the necessary experience exists within RWM to assimilate information about a site and incorporate this into relevant models, when it becomes available.

This report focuses on the potential radioactive gaseous emissions from the GDF during the operational period until the completion of all backfilling and sealing activities, and presents an illustrative quantitative assessment, using conservative assumptions, of their potential effects on members of the public and non-human biota. At this early stage of development, non-radioactive discharges, and radioactive solid and liquid discharges, from the GDF, are considered qualitatively, noting that such discharges (and any doses arising from them) will be highly site-specific and will be managed on a site-specific basis. Additional qualitative supporting arguments for the operational environmental safety of the GDF are also presented.

Emissions associated with gases generated in packaged wastes have been estimated. Modelling of discharges from legacy ULLW/UILW, legacy SLLW/SILW and NNB UILW has been undertaken for all significant radioactive gases (C-14-bearing methane and carbon monoxide, tritium, and Rn-222). Additionally, modelling of discharges from waste in RSCs, NNB SILW and DNLEU has been undertaken for Rn-222. Discharges are not expected from packaged HLW, SF, Pu or HEU, as these wastes and materials are assumed to be packaged in high integrity, unvented disposal containers. Potential discharges of naturally-occurring Rn-222 from the host rock are also considered.

A gas release base scenario (expected to provide a reasonable generic representation of most aspects of the likely gas generation behaviour) and a bounding case scenario (taking the highest peak release rate for each radionuclide across three variant scenarios) are considered in a quantitative assessment of dose. In the base scenario, backfilling of the vaults is assumed to occur after all emplacement operations have ceased, as currently scheduled for the GDF in a higher strength rock. In the bounding case scenario, the highest peak release rate for tritium arises from a staged backfilling variant, and the highest peak release of C-14 arises from a higher temperature variant. The peak release rate for Rn-222 is not increased in any of the variant scenarios, so it is the same in both the base scenario and the bounding case.

Illustrative, conservative calculations of dose to members of the public have been undertaken. The annual dose to members of the public from peak gas release rates during the operational period according to the base scenario, based on conservative assumptions

appropriate to this generic stage, is calculated to be 0.17 mSv per year. The majority of this dose arises from Rn-222 (0.11 mSv per year, almost all via the inhalation pathway), with minor contributions from C-14-bearing gases and tritium.

The annual dose to members of the public from peak gas release rates during the operational period according to the bounding case, again based on conservative assumptions, is calculated to be 0.28 mSv per year. The contribution from Rn-222 is unchanged from the base scenario at 0.11 mSv; the additional dose arises from tritium and C-14-bearing carbon monoxide.

The calculated doses to members of the public from gaseous emissions are below the legal dose limit for members of the public of 1 mSv per year but above the source-related dose constraint of 0.15 mSv per year and the BSO of 0.02 mSv per year. Therefore more work is needed before it can be demonstrated that the dose will be below the identified constraints.

Significant uncertainties and conservatism in the calculations, particularly relating to the modelling of radioactive gas release from the GDF, are identified, and a semi-quantitative discussion of their potential implications is presented. Scoping calculations have been undertaken to investigate the effect on calculated radiological dose of increasing the effective release height to a more realistic value, and indicate that a significant reduction in radiological doses can be achieved. Other potential mitigation measures are also discussed.

A generic illustrative assessment of the potential dose rates to non-human biota, assuming the bounding case gas release scenario, has also been undertaken. The calculated dose rates for all the organisms considered are insignificant compared to predicted no-effect dose rate values of 10 μ Gy per hour derived in the EC ERICA project. The assessment of dose to non-human biota from radioactive gaseous emissions to the atmosphere from the GDF is therefore concluded to require no further consideration at this stage.

Any actual radiological dose from gaseous emissions to the atmosphere from the GDF will be determined by site-specific factors, and will be a function of actual gaseous discharge rates during each year of GDF operation in combination with local environmental factors and the location and habits of exposed groups.

This report will be updated, in line with updates to the DSSC, as part of each major stage of the GDF development programme. Over time, the design options under consideration and the choices that will have to be made will change from an emphasis on strategy to one on implementation. This approach is consistent with a staged development and approval process.

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Glossary

A glossary of terms relating to the generic DSSC can be found in the Technical Background. The following terms are specific to this report.

Term	Definition
AETP	Active effluent treatment plant
discharge	Term used in this report to refer to disposals or emissions of solid, liquid or gaseous substances off site, to avoid confusion with disposal of waste within the GDF.
DPUR	Dose per unit release
DRR	Dose release ratio
effective release height	Gaseous discharges from the GDF will be released via discharge stack, the height and location of which can be varied. However, other site- and design-specific factors can also affect the dose received as a result of such discharges. The effective release height takes into account the physical discharge stack height, local topography, and efflux parameters (parameters that affect the flow of gas being discharged, such as exit diameter and velocity).
ERICA	Environmental Risk from Ionising Contaminants: Assessment and Management
PNEDR	Predicted no effect dose rate: dose rate value, derived as part of the EC ERICA project, below which a specified generic ecosystem is expected to be protected from effects on structure and function under chronic exposure to radionuclides.
RPCM	Radiological Protection Criteria Manual
SFR	Safety functional requirement
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation



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