Geological Disposal

Generic Operational Environmental Safety Assessment

December 2010
Geological Disposal
Generic Operational Environmental Safety Assessment

December 2010
Conditions of Publication

This report is made available under the NDA Transparency Policy. In line with this policy, the NDA is seeking to make information on its activities readily available, and to enable interested parties to have access to and influence on its future programmes. The report may be freely used for non-commercial purposes. However, all commercial uses, including copying and re-publication, require permission from the NDA. All copyright, database rights and other intellectual property rights reside with the NDA. Applications for permission to use the report commercially should be made to the NDA Information Manager.

Although great care has been taken to ensure the accuracy and completeness of the information contained in this publication, the NDA can not assume any responsibility for consequences that may arise from its use by other parties.

© Nuclear Decommissioning Authority 2010. All rights reserved.


Bibliography

If you would like to see other reports available from NDA, a complete listing can be viewed at our website http://www.nda.gov.uk, or please write to the Library at the address below.

Feedback

Readers are invited to provide feedback to the NDA on the contents, clarity and presentation of this report and on the means of improving the range of NDA reports published. Feedback should be addressed to:

John Dalton,
Head of Communications,
Nuclear Decommissioning Authority (Radioactive Waste Management Directorate),
Curie Avenue,
Harwell Campus,
Didcot,
Oxon,
OX11 0RH, UK
Abstract

In this generic Operational Environmental Safety Assessment, we have assessed potential off-site discharges and the associated radiological doses to members of the public and the environment (including non-human biota) which may arise from activities undertaken during the operational period of a Geological Disposal Facility (GDF). Non-radioactive discharges which may arise from these activities are also considered. Discharges from both the surface facilities and those underground are discussed.

At this stage there is no identified potential site for a GDF, therefore there is no information either about the geology and hydrogeology at a site, or about the detailed design of a GDF that might be constructed there. The information presented in this report is at a generic level independent of geology or a specific design. Some assumptions have been made in order to make illustrative quantitative assessments.

This report forms part of a suite of documents called a Disposal System Safety Case, which considers the safety of radioactive waste transport to a GDF, the safety of the construction and operation of a GDF, and the safety of the GDF in the very long term, after it has been sealed and closed. The generic DSSC consists of an overview report, describing the safety of a geological disposal system; three main documents, one for each of the three components of the overall safety case (the Transport Safety Case (TSC), the Operational Safety Case (OSC), and the Environmental Safety Case (ESC)) and a number of underpinning documents. The generic OESA supports both the ESC and the OSC.
Executive summary

The Nuclear Decommissioning Authority (NDA) has established the Radioactive Waste Management Directorate (RWMD) to manage the delivery of geological disposal for higher activity radioactive wastes, as required under UK Government policy published in the Managing Radioactive Waste Safely (MRWS) White Paper.

A Disposal System Safety Case (DSSC) considers the safety of radioactive waste transport to a Geological Disposal Facility (GDF), the safety of the construction and operation of a GDF, and the safety of the facility in the very long term, after it has been sealed and closed. A DSSC is in the early stages of development, because a site and design have not yet been chosen. We call it a 'generic' safety case, and the strategy to demonstrate safety is also termed 'generic' because it must cover a range of possible disposal environments and facility designs. Nevertheless, this work builds on more than 25 years of experience studying geological disposal and undertaking safety assessments in the UK. It also draws on the extensive body of knowledge and experience in other countries gained through overseas radioactive waste management programmes.

The generic DSSC consists of an overview report, describing the safety of a geological disposal system; three main documents, one for each of the three components of the overall safety case (the Transport Safety Case (TSC), the Operational Safety Case (OSC), and the Environmental Safety Case (ESC)) and a number of underpinning documents.

This report sets out our current understanding of what an Operational Environmental Safety Assessment (OESA) would involve, and supports the ESC and the OSC. We use illustrative disposal concepts to discuss the safety provided by a GDF in a range of potential host geological settings at this generic stage of the DSSC.

This report focuses on the potential radioactive discharges from a GDF during the operational period that may affect members of the public and non-human biota. Non-radioactive discharges from the wastes are also considered in this report. We consider qualitative supporting arguments for the operational environmental safety of a GDF, and also report a quantitative assessment of operational environmental safety.

The operational period is considered to begin when construction of the GDF is started, and ends with the closure of the facility after the completion of backfilling and final sealing. At this early stage of development we focus on aerial discharges, with liquid and solid discharges being qualitatively considered only (noting that such discharges will be managed on a site-specific basis).

Aerial discharges associated with gases generated in packaged wastes have been estimated. Discharges are dominated by gases generated in low level waste and intermediate level waste packages. 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 canisters. Aerial discharges as a result of gas generation within depleted, natural and low-enriched uranium (DNLEU) are expected to be low and not significant in comparison to releases from low level and intermediate level wastes.

The impact on aerial discharges from an Upper Inventory (which has been defined to account for potential increases in volumes that may arise through, for instance, the introduction of new-build nuclear power stations) is discussed in a qualitative fashion.

The total dose to members of the public for the operational period average gas generation rates is calculated to be 0.052 mSv per year. The majority of this dose arises from radon-222 (0.043 mSv), with minor contributions from carbon-14 bearing gas and tritium. This total dose is below the effective dose limit for members of the public of 1 mSv per year. It is also below the maximum dose to individuals which may result from the discharges from any single site, stated in the Environmental Permitting Regulations 2010,
of 0.5 mSv per year, and below the maximum value of dose constraint to members of the public from a new facility of 0.15 mSv per year adopted by RWMD. However, although the dose from carbon-14 and tritium is below the design target of 0.01 mSv per year, the dose from radon-222 exceeds the design target.

Backfilling of the GDF with a cementitious material is expected to take place prior to sealing and closure. Such backfilling has the effect of heating the waste packages as it cures (cement curing is exothermic) and backfilling may make available free water that can be used in e.g. corrosion reactions. Emplacement of a cementitious backfill could therefore contribute to a short-lived enhancement of corrosion rates and hence gas generation rates during the operational period. The rate of gas generation is calculated to peak during the backfilling process, and to be lower at earlier times (which will account for the majority of the operational period).

Studies have also been undertaken using the gas generation rate associated with the backfilling period. The total dose to members of the public for the operational period peak gas generation rates is calculated to be 0.16 mSv per year. The majority of this dose (0.11 mSv per) arises from carbon-14, with contributions from radon-222 and tritium. This total dose is below the effective dose limit for members of the public of 1 mSv per year. It is also below the maximum dose to individuals which may result from the discharges from any single site, stated in the Environmental Permitting Regulations 2010, of 0.5 mSv per year. However, this assessment presents doses to members of the public from radon-222 and carbon-14 that exceed the design target of 0.01 mSv per year, and also the total dose is above the maximum value of dose constraint to members of the public from a new facility of 0.15 mSv per year adopted by RWMD.

The retention of radon-222 (half-life 3.82 days) within a waste package is expressed in terms of an ‘emanation coefficient’, which corresponds to the fraction of radon-222 that is released from a waste package in comparison with the in-package radon-222 generation rate. Radon-222 doses presented in this report assume an emanation coefficient of 2x10^{-3}. This is considered to be a reasonable value to use at this generic stage, although it is known that significant improvements can be made on this through adoption of multiple barriers in the wasteform and waste package. As part of the RWMD Letter of Compliance disposability assessment process, we will ensure that the emanation coefficient for radon-222 is reduced to a level consistent with the needs of the GDF. Existing practice has established that a significant improvement in emanation coefficient is possible to the extent that the associated dose to members of the public can be reduced by up to three orders of magnitude. We have confidence therefore that the design target of 0.01 mSv per year can readily be achieved by application of appropriate packaging measures.

Studies have also been undertaken to investigate the effect on calculated radiological dose of varying the discharge stack height – a significant reduction in radiological doses can be achieved by increasing the discharge stack height. Other such potential mitigation measures are also discussed.

A generic illustrative assessment of the potential doses to non-human biota has been undertaken. The total calculated dose rates for all of the organisms considered are insignificant in consideration of reviews of the effects of ionising radiation on organisms reported by the International Atomic Energy Agency (IAEA) and United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), which give a broad conclusion that exposure of terrestrial organisms to dose rates of 40 μGy per hour, and exposure of aquatic organisms to dose rates of 400 μGy per hour, would be unlikely to lead to observable effects in these populations. The assessment of dose to non-human biota from off site gaseous radioactive discharge from a GDF is therefore concluded to require no further consideration.

Any actual radiological dose from off site discharges from a 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. We are aware of the need to ensure active management of gas in the operational phase of a GDF, to ensure regulatory dose constraints are met. The issues and uncertainties associated with estimates of gas generation are set out in the gas status report. Mitigation measures could be taken to reduce the potential for gaseous discharges from a GDF that would, consequently, also reduce any off site doses.

We will continue to advance our understanding of discharges (both radioactive and non-radioactive) from a facility that may affect members of the public and the environment including non-human biota, by commissioning research and development to address areas of uncertainty in a manner that takes account of their potential significance, and the status and progress of our programme to implement a GDF.

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

Abstract iii
Executive summary v
Main abbreviations and acronyms xiii

1 Introduction 1
1.1 Government policy 1
1.2 The wastes 1
1.3 Multiple barrier systems 2
1.4 Illustrative concepts 4
1.5 Disposal System Safety Case 10
1.6 Supporting reports 11
1.7 Aim of this operational environmental safety assessment report 12
1.8 Other relevant assessment reports 14

2 Scope 15
2.1 OESA strategy 15
2.2 Inventory 17
2.3 Discharges 18
2.4 Operations 18

3 Safety objectives 21
3.1 Geological disposal facility design principles 21
3.2 Radiological protection policy manual 21
3.3 Statutory basis 22
3.4 Regulatory guidance and other material 22
3.5 Current international status of assessment work on non-human biota 24

4 Derivation of off site discharges 25
4.1 Solid discharges 25
4.2 Liquid discharges 26
4.3 Aerial discharges 27
4.4 Aerial radioactive discharges – inventory 29
4.5 Derivation of gas generation rates for use in assessment of doses from off site radioactive aerial discharges 32
4.6 Summary - radioactive gas release rates for use in off site dose assessment calculations 33

5 Methodology for assessing doses from off site gaseous discharges 35
5.1 Guidance on methodology 35
5.2 Methodology for assessment of doses to members of the public 35
5.3 Methodology for assessment of dose rates to non-human biota 37

6 Safety analysis 39
6.1 Qualitative safety arguments 39
6.2 Calculated doses to representative public: local resident family 41
6.3 Calculated doses to terrestrial reference organisms 43

7 Discussion 45
7.1 Uncertainties and potential conservatisms 45
7.2 Mitigation measures 47
7.3 Monitoring 48
7.4 Summary 48

8 Future work 51
8.1 Consideration of the Derived Inventory in future OESA assessments 51
8.2 Future research 51
8.3 Further work on non-radioactive discharges 52
8.4 Concept development 52
8.5 Future iterations of the OESA 53
8.6 Statement of confidence 53

Glossary 55

References 69
Main abbreviations and acronyms

National organisations

- CoRWM: Committee on Radioactive Waste Management
- HPA: Health Protection Agency (formerly NRPB National Radiological Protection Board)
- HSE: Health and Safety Executive
- NDA: Nuclear Decommissioning Authority
- Nirex: United Kingdom Nirex Limited
- RWMD: NDA Radioactive Waste Management Directorate

International and overseas organisations

- Andra: French national radioactive waste management agency
- DBE: German company for the construction and operation of waste repositories
- EC: European Commission
- IAEA: International Atomic Energy Agency
- ICRP: International Commission on Radiological Protection
- Nagra: Swiss co-operative for the disposal of radioactive waste
- NEA/OECD: Nuclear Energy Agency / Organisation for Economic Co-operation and Development
- Ondraf/Niras: Belgian agency for radioactive waste and enriched fissile materials
- SKB: Swedish nuclear fuel and waste management company
- US DOE: United States Department of Energy
- UNSCEAR: United Nations Scientific Committee on the Effects of Atomic Radiation

Legislation, regulation, guidance

- EPR 2010: Environmental Permitting (England and Wales) Regulations 2010
- GRA: Guidance on Requirements for Authorisation
- RSA 93: Radioactive Substances Act 1993

RWMD safety cases and safety assessments

- DSSC: Disposal System Safety Case
- EIA: Environmental Impact Assessment
- ESC: Environmental Safety Case
- LoC: Letter of Compliance
- OESA: Operational Environmental Safety Assessment
- PCSA: Post-closure Safety Assessment
- SEA: Strategic Environmental Assessment

Radioactive material types

- DNLEU: depleted, natural and low-enriched uranium
<table>
<thead>
<tr>
<th>Acronym</th>
<th>Full Form</th>
</tr>
</thead>
<tbody>
<tr>
<td>HEU</td>
<td>highly enriched uranium</td>
</tr>
<tr>
<td>HLW</td>
<td>high level waste</td>
</tr>
<tr>
<td>ILW</td>
<td>intermediate level waste</td>
</tr>
<tr>
<td>LLW</td>
<td>low level waste</td>
</tr>
<tr>
<td>Pu</td>
<td>plutonium</td>
</tr>
<tr>
<td>SF</td>
<td>spent fuel</td>
</tr>
<tr>
<td>SILW</td>
<td>shielded intermediate level waste</td>
</tr>
<tr>
<td>U</td>
<td>uranium</td>
</tr>
<tr>
<td>UILW</td>
<td>unshielded intermediate level waste</td>
</tr>
</tbody>
</table>

**Other**

<table>
<thead>
<tr>
<th>Acronym</th>
<th>Full Form</th>
</tr>
</thead>
<tbody>
<tr>
<td>AETP</td>
<td>active effluent treatment plant</td>
</tr>
<tr>
<td>AGR</td>
<td>Advanced Gas-cooled Reactor</td>
</tr>
<tr>
<td>ALARA</td>
<td>as low as reasonably achievable</td>
</tr>
<tr>
<td>BAT</td>
<td>best available techniques</td>
</tr>
<tr>
<td>DSS</td>
<td>Disposal System Specification</td>
</tr>
<tr>
<td>EBS</td>
<td>engineered barrier system</td>
</tr>
<tr>
<td>FEPs</td>
<td>features, events and processes</td>
</tr>
<tr>
<td>GDF</td>
<td>geological disposal facility</td>
</tr>
<tr>
<td>GOSA</td>
<td>Generic Operational Safety Assessment</td>
</tr>
<tr>
<td>MRWS</td>
<td>Managing Radioactive Waste Safely</td>
</tr>
<tr>
<td>PWR</td>
<td>Pressurised Water Reactor</td>
</tr>
<tr>
<td>R&amp;D</td>
<td>research and development</td>
</tr>
<tr>
<td>RPPM</td>
<td>radiation protection policy manual</td>
</tr>
<tr>
<td>SFR</td>
<td>safety functional requirements</td>
</tr>
<tr>
<td>UKRWI</td>
<td>United Kingdom Radioactive Waste Inventory</td>
</tr>
<tr>
<td>WVP</td>
<td>Waste Vitrification Plant</td>
</tr>
<tr>
<td>WIPP</td>
<td>Waste Isolation Pilot Plant</td>
</tr>
</tbody>
</table>
1 Introduction

The Nuclear Decommissioning Authority (NDA) has established the Radioactive Waste Management Directorate (RWMD) to manage the delivery of geological disposal for higher activity radioactive wastes, as required under UK Government policy published in the Managing Radioactive Waste Safely (MRWS) White Paper [1]. This policy also states that the siting of a geological disposal facility (GDF) will be based on a voluntarism and partnership approach. This report is one of a set of assessment and status reports whose purpose is to describe the science and technology underpinning the safety cases for geological disposal of UK higher activity radioactive wastes. This report is the Operational Environmental Safety Assessment (OESA) report, which is a component of the generic Environmental Safety Case (ESC). The OESA describes our current understanding of potential solid, liquid and gaseous discharges from a GDF during its routine operation.

1.1 Government policy

UK Government policy is that geological disposal is the way higher activity radioactive wastes will be managed in the long term: this will be preceded by safe and secure interim storage until a GDF is available and can receive waste. The NDA also has responsibility for the safe and secure storage of its wastes. This responsibility is discharged by other parts of the NDA and is not discussed in this report. In our work we take account of storage arrangements for higher activity wastes and nuclear materials and through co-location try to ensure optimal solutions are developed for their management.

The definition of geological disposal given by the UK Government’s advisory Committee on Radioactive Waste Management (CoRWM) in its recommendations to UK Government in 2006 [2] has been followed through in the UK Government response to those recommendations [3] and in the MRWS White Paper. This is ‘burial underground (200-1,000m) of radioactive waste in a purpose built facility with no intention to retrieve …’. This definition applies to all references to geological disposal in this report.

1.2 The wastes

The MRWS White Paper [1] provides an estimate of the volumes and characteristics of higher activity radioactive waste and other materials that could, possibly, come to be regarded as wastes that might need to be managed in the future through geological disposal. This ‘Baseline Inventory’ [1] is based on the 2007 UK Radioactive Waste Inventory (UKRWI) [4]. It includes materials not currently classified as waste - spent nuclear fuel, and separated plutonium and uranium stocks. However, it excludes low level radioactive waste (LLW) that can be managed under the Government’s ‘Policy for the Long Term Management of Solid Low Level Radioactive Waste in the United Kingdom’ [5].

The Government’s policy (paragraph 3.8 in the MRWS White Paper) is that, pending a decision whether the radioactive materials included in the Baseline Inventory should be declared as waste, we should factor their possible inclusion into the design and development of a GDF. Therefore for planning purposes the Baseline Inventory is used as the basis for developing a disposal system specification [6, 7] and, in turn, GDF engineering designs that meet this specification. These facility designs [8] provide the basis for assessments of the associated safety and environmental, social and economic impacts and for assessments of costs.

---

1 It should be noted that at present the Baseline Inventory is based on 2007 UKRWI figures, and as such, currently contains waste expected to be managed under the Scottish Government’s policy of interim near-surface, near-site storage as announced on 25 June 2007.
The MRWS White Paper sets out the Baseline Inventory in terms of volume and activity, for the different waste types, as shown in Table 1 below. The White Paper emphasises that the figures are only indicative and that the assumptions that support them may change. For design and assessment purposes, more detailed information related to the characteristics of individual waste packages is needed and therefore a ‘Derived Inventory’ has been developed. The ‘Derived Inventory’ is based directly on the Baseline Inventory but with information presented on a ‘per package’ basis and taking account of the effects of radioactive decay (and build-up) as appropriate. In addition to the Baseline, the Derived Inventory has also considered an ‘Upper Inventory’ which accounts for potential increases in waste and material volumes that may arise through for instance, alternative scenarios for reprocessing or the introduction of new-build nuclear power stations. The Derived Inventory is fully described in the Disposal System Technical Specification [7].

### Table 1 The MRWS White Paper Baseline Inventory

<table>
<thead>
<tr>
<th>Materials</th>
<th>Notes</th>
<th>Packaged volume</th>
<th>Radioactivity (At 1 April 2040)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>Cubic Metres</td>
<td>%</td>
</tr>
<tr>
<td>HLW</td>
<td>1, 2, 3, 5</td>
<td>1,400</td>
<td>0.3%</td>
</tr>
<tr>
<td>ILW</td>
<td>1, 2, 5</td>
<td>364,000</td>
<td>76.3%</td>
</tr>
<tr>
<td>LLW (not for LLWR)</td>
<td>1, 2, 5</td>
<td>17,000</td>
<td>3.6%</td>
</tr>
<tr>
<td>Spent nuclear fuel</td>
<td>1, 4, 5</td>
<td>11,200</td>
<td>2.3%</td>
</tr>
<tr>
<td>Plutonium</td>
<td>1, 4, 5</td>
<td>3,300</td>
<td>0.7%</td>
</tr>
<tr>
<td>Uranium</td>
<td>1, 4, 5</td>
<td>80,000</td>
<td>16.8%</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td></td>
<td><strong>476,900</strong></td>
<td><strong>100</strong></td>
</tr>
</tbody>
</table>

Notes:
1. Quantities of radioactive materials and wastes are consistent with the 2007 UK Radioactive Waste Inventory [4].
2. Packaging assumptions for HLW, ILW and LLW not suitable for disposal at the existing national LLWR are taken from the 2007 UKRWM. Note that they may change in the future.
3. The HLW packaged volume may increase when the facility for disposing the canisters, in which the vitrified HLW is currently stored, has been implemented.
4. Packaging assumptions for plutonium, uranium and spent nuclear fuels are taken from the 2005 CorRWM Baseline Inventory [9]. Note that they may change in the future.
5. Radioactivity data for wastes and materials was derived using the 2007 UK Radioactive Waste Inventory, 2040 is the assumed start date for the geological disposal facility.
6. It should be noted that at present the Baseline Inventory is based on UK Inventory figures, and as such, currently contains waste expected to be managed under the Scottish Executive’s policy of interim near-surface, near-site storage as announced on 25 June 2007 [10].

**1.3 Multiple barrier systems**

There are two high-level safety objectives of geological disposal of radioactive waste, namely to isolate the waste from the biosphere and to contain the radionuclides associated with the waste. In order to assure that these objectives of isolation and containment are delivered over the long timescales of interest, geological disposal facilities are designed as multiple barrier systems. This involves designing engineered barriers that will work together and in combination with the natural barrier afforded by the geosphere to prevent radionuclides being released to the surface environment in amounts that could cause harm to life and the environment.

---

2 HLW: high level waste; ILW: intermediate level waste; LLW: low level waste; LLWR: low level waste repository.
As noted above, the multiple barrier concept of disposal addresses two principal objectives with respect to providing safety - the isolation of the wastes and the containment of the radionuclides associated with the wastes:

- By isolation we mean removing the waste from people and the surface environment. Geological disposal at depth in a suitable environment provides isolation by reducing the likelihood of inadvertent and unauthorised human interference. Disposal in a geological environment that is suitably deep and stable over long periods also provides isolation of the disposal facility from the impacts of climatic and other natural environmental events, and shielding of the natural environment from direct radiation from the waste.

- By containment we mean retaining radionuclides within various parts of the multi-barrier system for as long as required by our safety concepts. Radioactive decay will progressively reduce the quantities of radionuclides present in the system. For many radionuclides, disposal concepts can provide total containment until the radionuclides and their daughters decay to insignificant levels of radioactivity within the engineered barrier system. However, the engineered barriers in a disposal facility will degrade progressively over time and gradually lose their ability to provide containment. Further containment is provided by the geological barrier, which acts to delay the movement of any small amounts of long-lived radionuclides that are released from the engineered barrier system. Locating the geological disposal facility in a suitably deep and stable environment protects the engineered barriers, helping them to preserve their containment functions for longer times.

A schematic representation of a multiple barrier system is provided in Figure 1. More specific examples are presented in [11].

The main barriers are:

- The wasteform: This is the form into which the waste is conditioned to make it suitable for disposal.

- The waste container: The conditioned waste is placed in a container (sometimes called a canister), creating what is referred to as the waste package.

- The buffer or backfill: The buffer or backfill in this context refers to material that is placed immediately around emplaced waste containers in a disposal facility. In addition, other types of ‘mass backfill’ will be required to fill excavated access tunnels, shafts or drifts together with sealing systems.

- The geosphere: The geosphere refers to the geological environment in which a GDF is constructed and provides a considerable barrier to the movement of radioactivity back to the accessible environment. In addition to providing isolation, this barrier protects the emplaced wastes from extreme changes that may take place at the Earth’s surface, as a result of either natural causes, as in the example of glaciations, or human actions.

The various components of the multiple barrier geological disposal system contribute to fulfilling the high-level safety objectives of containment and isolation [12] in different ways over different timescales. In line with regulatory guidance in respect of the environmental safety case [13] and with practice in other national programmes (e.g. [14]) we define safety functions for each of the various components of the multiple barrier system which set out what the barrier component should achieve to ensure safety.
1.4 Illustrative concepts

Work that we carried out in 2008 identified a range of possible concepts for geological disposal of ILW/LLW [15], and of high level radioactive waste (HLW) and spent fuel [16], effectively providing us with a catalogue of concepts for consideration. The work drew on previous work in the UK, and disposal programmes in other countries, to identify disposal concepts for generic geological settings (host rock formations and associated geological and hydrogeological conditions).

At the current stage of the programme we are examining a wide range of potentially suitable disposal concepts so that a well-informed assessment of options can be carried out at appropriate decision points in the implementation programme. Drawing from this work we have set out illustrative concepts for three generic geological settings, including the associated variants on rock formations that might overlie the GDF host rock.

We are using these illustrative concepts to:

- further develop our understanding of the functional and technical requirements of the disposal system;
- further develop our understanding of the design requirements;
- support the scoping and assessment of the safety, environmental, social and economic impacts of a GDF;
- support development and prioritisation of our research and development programme;
- underpin our analysis of the potential cost of geological disposal; and
- support assessment of the disposability of waste packages proposed by waste owners.

We have set out the illustrative concepts solely for these purposes. We do not intend that one of these illustrative concepts is necessarily the one that we would use in the relevant geological setting. At this stage, no geological disposal concept has been ruled out. The key aspects of RWMD’s proposed approach to optioneering are described in [17].
We selected the illustrative geological disposal concepts following consideration of the concepts identified in the studies of disposal concepts for ILW/LLW [15] and for HLW and spent fuel [16]. We selected concepts that are well-developed and supported by extensive research and development, and have been subject to detailed safety assessment, regulatory scrutiny and international review [18]. In line with the MRWS White Paper our safety assessment studies at this stage assume that a GDF is a ‘co-located’ facility. This means that the illustrative disposal concept examples for ILW/LLW and HLW/SF shown in Table 2, are assumed to be developed at a single site. This is seen as a reasonable and conservative assumption for assessment purposes at this stage.

The illustrative concepts are listed below and the attached notes present the key reasons why these examples were selected.

### Table 2 Illustrative geological disposal concept examples for different waste types

<table>
<thead>
<tr>
<th>Host rock</th>
<th>ILW/LLW</th>
<th>HLW/SF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Higher strength rocks*</td>
<td>UK ILW/LLW Concept (NDA, UK)</td>
<td>KBS-3V Concept (SKB, Sweden)</td>
</tr>
<tr>
<td>Lower strength sedimentary rock†</td>
<td>Opalinus Clay Concept (Nagra, Switzerland)</td>
<td>Opalinus Clay Concept (Nagra, Switzerland)</td>
</tr>
<tr>
<td>Evaporites‡</td>
<td>WIPP Bedded Salt Concept (US-DOE, USA)</td>
<td>Gorleben Salt Dome Concept (DBE-Technology, Germany)</td>
</tr>
</tbody>
</table>

Notes
a. Higher strength rocks – the UK ILW/LLW concept and KBS-3V concept for spent fuel were selected due to availability of information on these concepts for the UK context.
b. Lower strength sedimentary rocks – the Opalinus Clay concept for disposal of long-lived ILW, HLW and spent fuel was selected because a recent OECD Nuclear Energy Agency report regarding the Nagra (Switzerland) assessment of the concept as state of the art with respect to the level of knowledge available. However, it should be noted that there is similarly extensive information available for a concept that has been developed for implementation in Callovo-Oxfordian Clay by Andra (France), and which has also been accorded strong endorsement from international peer review. Although we will use the Opalinus Clay concept as the basis of the illustrative example, we will also draw on information from the Andra programme. In addition, we will draw on information from the Belgian super container concept, based on disposal of HLW and spent fuel in Boom Clay.
c. Evaporites – the concept for the disposal of transuranic wastes (TRU) (long-lived ILW) in a bedded salt host rock at the Waste Isolation Pilot Plant (WIPP) in New Mexico was selected because of the wealth of information available from this United States Environmental Protection Agency (EPA) certified, and operating facility. The concept for disposal of HLW and spent fuel in a salt dome host rock developed by DBE Technology (Germany) was selected due to the level of concept information available.
d. For planning purposes the illustrative concept for depleted, natural and low enriched uranium is assumed to be same as for ILW/LLW and for plutonium and highly enriched uranium is assumed to the same as for HLW/SF.

The use of generic geological settings does not imply that any specific sites are being considered. The host rock descriptions correspond to three distinct general rock types that are considered potentially suitable to host a disposal facility for higher activity wastes, based on studies carried out in the UK and internationally, and which occur in the UK. They are described as follows:

- **Higher strength rocks** - these would typically comprise crystalline igneous, metamorphic rocks or geologically older sedimentary rocks, where any fluid movement is predominantly through divisions in the rock, often referred to as discontinuities. Granite is a good example of a rock that would fall in this category.

- **Lower strength sedimentary rocks** - these would typically comprise geologically younger sedimentary rocks where any fluid movement is predominantly through the rock mass itself. Many types of clay are good examples of this category of rocks.
• Evaporites - these would typically comprise anhydrite (anhydrous calcium sulphate), halite (rock salt) or other evaporites that result from the evaporation of water from water bodies containing dissolved salts.

These illustrative geological disposal concepts are described in the GDF Design Report [8]. A schematic illustration of a generic GDF comprising multiple barriers is shown in Figure 2.

**Figure 2** Schematic illustration of the layout of a generic GDF (not host rock specific) for all UK higher activity radioactive wastes

---

**1.4.1 Waste packaging**

Some wastes are already being conditioned and packaged in anticipation of geological disposal. We have an important role to support the waste packaging sites through assessment of waste packaging proposals before waste packages are actually manufactured. Through application of the Letter of Compliance disposability assessment process [19] we will work with the waste packager to gain confidence that manufactured waste packages will be compliant with the anticipated needs of transport [20], and with safety cases for disposal in a GDF. This will include demonstration of compliance with the packaging specifications that have been developed on the basis of national and international standards and against the anticipated requirements of transport and disposal, [21,22].

Through application of this process we ensure that the two waste package related barriers – the wasteform and the waste container – are considered and developed in the context of the developing safety cases for transport and disposal, what we collectively describe as the Disposal System Safety Case (DSSC) and which is described further in Section 1.5. In

---

3 SF: spent fuel; UILW: unshielded ILW; SILW: shielded ILW
cases where we are satisfied that the waste package will be compliant with the safety cases this is signified by the issue of a Letter of Compliance.

For wastes and materials for which packaging plans have not been fully developed, it has been necessary for the purposes of the DSSC to make assumptions regarding the wasteform and waste container, and the characteristics and performance of these barriers. These assumptions are summarised below and greater detail is provided in the GDF Design Report [8].

1.4.1.1 ILW plus depleted, natural and low enriched uranium

ILW and that LLW destined for geological disposal is assumed to be packaged in such a manner that will ensure compliance with the generic waste package specification [21]. This will generally involve the use of one of the standardised waste containers defined in [7] and a conditioning process that ensure adequate immobilisation of the radionuclides associated with the waste.

Packaging options for disposal of depleted, natural and low enriched uranium (DNLEU) are being investigated. At present, for the purposes of GDF planning and development of safety cases, we assess a design in which DNLEU is disposed of in the ILW disposal area, compacted in the form of a uranium oxide and packaged into 500 litre drums (Figure 3) or 3 cubic metre boxes [7]. Packaging solutions for ILW and LLW are shown in Figure 4. These packaging assumptions are consistent with the MRWS White Paper [1], but it is recognised that the actual packaging may be different and has not been optimised.

**Figure 3**  Cutaway of 500 litre drum showing drum and waste
Figure 4  Standard package containers for ILW/LLW
1.4.1.2 HLW, spent fuel, highly enriched uranium and plutonium

At this stage, a number of assumptions have been made on the conditioning and packaging processes for HLW, spent fuel, highly enriched uranium and plutonium in order to provide a baseline and to assess their disposability:

- HLW is conditioned by immobilising it in borosilicate glass (the process of vitrification) inside stainless steel canisters known as Waste Vitrification Plant (WVP) canisters. The packaging assumption, at this stage of GDF development, is that these will be overpacked in high-integrity containers prior to disposal;

- In the event that spent fuel is declared a waste it is assumed that fuel assemblies will be overpacked into high-integrity disposal containers. Pressurised Water Reactor (PWR) fuel would be packaged directly in the form of complete fuel assemblies. Advanced Gas-cooled Reactor (AGR) fuel would be dismantled with the graphite sleeves, support grids, and braces being processed separately as ILW, and the remaining fuel pins being consolidated into bundles within a stainless steel ‘basket’ which is overpacked;

- In the event that plutonium and highly enriched uranium are declared as waste it is assumed that these are processed by conversion into a titanium-based ceramic puck, with multiple pucks placed in a stainless steel can. These cans would then be encapsulated in glass in a large canister (similar to a WVP canister) and then packaged in a high-integrity container for disposal.

The above packaging assumptions are consistent with the MRWS White Paper [1], although it is recognised that actual packaging will need to be optimised and may be different. Packaging for HLW and spent fuel are shown in Figure 5.

Materials suitable for the construction of the high-integrity waste container are currently being investigated. For HLW and spent fuel, initial work [22] considered packaging in a waste container manufactured from copper, similar to that currently envisaged for spent fuel disposal in Sweden and Finland. Subsequent work has reviewed iron-based waste containers [23], other possible candidate container materials [24] and engineered barrier system options [25]. The length and diameter of the waste container is currently specified to accommodate UK WVP canisters, and spent AGR and Sizewell ‘B’ PWR fuels.
1.5 Disposal System Safety Case

A safety case demonstrates safety using a collection of arguments and evidence, including the evaluation of consequence and risk in a safety assessment. A Disposal System Safety Case (DSSC) will provide evidence to demonstrate that the disposal system will be safe to operate, will remain safe after it is closed and will meet all applicable regulatory safety requirements. It will include a statement about the approach to safety and the safety assessment that has been undertaken, along with supporting information.

The safety case for the disposal system is currently at an early stage of development, because a site and design for a GDF have not yet been chosen. We call it a generic safety case because it must cover a range of possible disposal environments and facility designs. The generic DSSC [11] describes in broad terms why we are able to have confidence in the safety of the transport of the waste to a GDF [26], the construction and operation of the facility [27] and the long-term safety after it is closed [28].

The generic DSSC consists of a hierarchy of documents. A part of this hierarchy includes a set of status reports. Figure 6 illustrates the generic DSSC document hierarchy and indicates how each status report relates to one or more of the Tier 1 safety case reports. These draw on Tier 2 safety assessments which include calculations for illustrative disposal concepts. These provide the basis for a ‘reference case’ to enable us to provide benchmark safety assessment calculations as an input to the disposability assessments of waste packaging proposals. The status reports provide the appropriate data for these calculations and support the overall safety case.
The evolving safety case will explain and justify the functions provided by each barrier. It will identify the time periods over which the barriers are expected to perform their safety functions in relation to various radionuclides and also the alternative or additional safety functions that operate if a barrier does not fully perform.

Figure 6 Disposal System Safety Case document hierarchy

1.6 Supporting reports

A set of status reports has been produced that relates to key research topics:

- reports on package evolution [29], near-field evolution [30], and geosphere [31], describing the understanding of the role and evolution of the barriers;
- reports on radionuclide behaviour [32], and gas generation and migration [33] describing the release and movement of materials through the multi-barrier system;
- reports on criticality safety [34] and one on waste package accident performance [35] addressing the control of low probability events and their outcome; and
- a report on biosphere describing what we think a future biosphere may look like and how radionuclide uptake might be expected to take place [36].

Each status report is based on extensive research and development evidence and understanding to support the safety arguments that are made in the three main safety case reports of the generic DSSC. Where applicable, the status reports justify key parameters (at an appropriate level) that are used and referenced by the DSSC safety
assessment documents [e.g. 37, 38, 39, 40, 41, 42]. Each status report also describes the uncertainties associated with a particular research and development topic and identifies any gaps in knowledge and the work that remains. These gaps will be addressed through our research and development strategy [43] as part of the work programme to implement geological disposal that is described in ‘Geological Disposal: Steps towards implementation’ [44].

The status reports are an integral part of the DSSC and will be submitted to regulators as part of our voluntary scrutiny agreement. It is anticipated that the status reports will also be widely circulated to all interested stakeholders. We have written each status report to inform all those who take an interest in our work within a particular research topic. Each status report has supporting technical and scientific references and, therefore, provides an overview and introduction to the published scientific literature for each topic. However, the status reports are technical documents and therefore in full, or in part, may require the reader to have some existing knowledge regarding the various research topics of interest.

The status report that is particularly applicable to the OESA is the gas status report. As mentioned above, there are also a number of underpinning technical reports to the DSSC. One of these reports which supports the OESA is the radioactive wastes and assessment of the disposability of waste packages report [45].

1.7 Aim of this operational environmental safety assessment report

As part of the generic DSSC, the OESA addresses the environmental safety of potential radioactive discharges and non-radioactive (i.e. resulting from the chemically hazardous properties of the waste) discharges arising from the operational phase of a GDF. It supports the ESC and the OSC. It considers the radiological impact to members of the public and to non-human biota⁴. The report is intended to cover aspects required to be considered under the Environmental Permitting Regulations 2010 [46] and the Ionising Radiations Regulations 1999 [47].

The approach to production of the OESA is shown in Figure 7. The process shown applies to potential radioactive and non-radioactive discharges to the aerial, aquatic and terrestrial environments. Although the control measures associated with a GDF will largely be based on passive features, operational functions will require active safety features. We will address this combination of passive and active features within the OESA to demonstrate an appropriate level of overall environmental safety for the operational period of a GDF.

⁴ The term ‘non-human biota’ refers to flora and fauna.
The purpose of the generic OESA is to provide an illustrative quantitative indication of off site doses associated with the operational phase of a GDF. These doses are herein compared with relevant criteria set out in the RWMD Radiological Protection Policy Manual [48], and will facilitate the development of an understanding of the importance of various factors which lead to the derived results (data, assumptions, etc.).

The OESA builds upon and develops the off site dose assessment presented in the earlier safety case (published in 2003) that we refer to as the Generic Operational Safety Assessment (GOSA) [49].

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

- **Section 2, Scope:** this section briefly sets out the scope of what is covered within this assessment report.
- **Section 3, Safety objectives and design criteria:** this section sets out RWMD policy and the principles against which the results of this assessment are judged.
- **Section 4, Source term:** this section describes how the off site discharges have been derived based on the wastes and activities assessed, as set out in Section 2.
- **Section 5, Assessment of off site discharges:** this section describes the methodology for calculating off site doses based on the discharges from the site derived in Section 4. This section includes a description of the illustrative exposure scenarios assumed for representative members of the public.
Section 6, Safety analysis: this section presents calculated doses to representative members of the public and non-human biota and where appropriate compares these doses with applicable design criteria and safety objectives.

Section 7, Discussion: this section discusses the results in context with the design criteria and safety objectives. Uncertainties and conservatisms in the assessment and potential mitigation measures that could be employed are also discussed.

Section 8, Future work: this section describes the future work in this area that is either ongoing or planned to address issues raised by this and related assessment reports.

A glossary provides definitions of key technical terms used in this report.

1.8 Other relevant assessment reports

This report assesses the off site impact of routine discharges of waste from the site. The hazards during the construction period including non-radiological safety are discussed in Volume 1 of the Generic Operational Safety Assessment [39]. Off site doses received as a result of direct radiation shine from sources on the site are assessed in Volume 2 of the Generic Operational Safety Assessment [40]. Fault conditions resulting in consequences off site are considered in Volume 3 of the Generic Operational Safety Assessment [41]. Consequences during the post closure phase of GDF implementation are assessed in the Post-closure Safety Assessment [37], which together with this report supports the ESC.
2 Scope

The Environmental Permitting Regulations [46] refer to the disposal of radioactive waste where ‘disposal’ is defined as referring to discharges whether into water, air, sewer, drain or otherwise. To avoid confusion with the disposal of waste within the facility, in this report we have used the term ‘discharges’ to refer to disposals off site. This generic Operational Environmental Safety Assessment (OESA) therefore considers the radiological exposure of individuals resulting from off site discharges during the operational period of a GDF. Radiological exposure of non-human biota is considered along with members of the public. Non-radioactive discharges are also considered.

The operational period of a GDF is considered to begin when construction of the facility is started, and ends with the closure of the facility after the completion of backfilling and final sealing. During the operational period it is anticipated that further construction activities could take place at the same time as waste emplacement, and, possibly, closure of parts of the facility that have already been filled with waste.

In this generic OESA we do not present an assessment of discharges that could occur during the GDF construction phase prior to waste emplacement (e.g. drilling liquids, leachates associated with spoil heaps etc), although note the need to do this when we move from generic to site-specific as the MRWS process proceeds. Work supporting the Operational Safety Case, e.g. the Generic Operational Safety Assessment – Volume 1 Construction and non-radiological safety assessment, specifically looks at the hazards that could occur from activities during the construction period and demonstrates our awareness of the issues for future consideration.

Illustrative geological disposal concept examples for a GDF constructed in a higher strength host rock, in a lower strength sedimentary host rock, and in an evaporite host rock, as adapted for UK conditions, are described in [8] and are considered in the generic DSSC. Although the disposal inventory would be invariant between these concept examples, GDF operations would need to be adapted to host rock properties. For example, for planning purposes, a notional period of 10 years has been included in [8] for the backfilling of access tunnels, sealing and closure of a GDF, irrespective of the host rock type. However, for a GDF in a higher strength host rock, backfilling of the vaults is assumed to occur once all emplacement operations have ceased, whereas for a GDF in a lower strength sedimentary host 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 [8]. Such differing approaches to the way a GDF is operated dependent on host rock could affect the rate of discharges to be assessed in an OESA. GDF operations could 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 the OESA – qualitative assessment, management of uncertainty, and quantitative assessment – is described below.

5 ‘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.
2.1.1 OESA: qualitative assessment

Our demonstration of understanding of the geological disposal system during the operational period comes from a variety of sources:

- Key inputs come from our own R&D programme and from previous R&D work by Nirex. As we move forward, we expect further understanding relevant to operational environmental safety to be developed, e.g. with respect to waste package behaviour during the operational period, the geological environment, and the transport of potential radiological and non-radiological contaminants at the particular site(s) under consideration.

- We contribute to the UK National Dose Assessment Working Group [50], which aims to bring together people and organisations with responsibility for, or an interest in, the assessment of radiation doses to the public from the operation of nuclear facilities.

- We participate in several international initiatives concerned with reducing uncertainties in environmental modelling. For example, we are a member of the BIOPROTA forum [51], 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 (e.g. carbon-14), and is relevant to both the OESA and PCSA.

- We also follow and contribute to other international programmes concerned with environmental modelling. For instance, the IAEA sponsors the Environmental Modelling for Radiation Safety (EMRAS) project [52]. The first EMRAS programme concluded its activities in 2007, and a second programme is now running in the period 2009-2011.

In Section 6, we use this understanding to summarise qualitative supporting arguments for operational environmental safety of a GDF, to discuss the potential for off site release of contaminants and public exposure for a GDF, 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 6 is at a level of detail appropriate to this generic stage of a GDF implementation programme, and is supported by our 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 our 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 a GDF;

- uncertainties in contaminant migration routes beyond a GDF; and

- uncertainties in calculating potential consequences to exposed individuals and other receptors.
2.1.3 OESA: quantitative assessment

The conceptual models that underpin the OESA need to consider the most important features of the waste, processes that could potentially lead to releases of radioactivity and other hazardous constituents during operation of a 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. During the operational period, radiological impacts need to be considered for representative members of the critical group. Characterisation of critical groups includes consideration of local food consumption and occupancy habits, based on identifying small groups, or in some cases individuals, with habits likely to result in a dose as a result of GDF operations. The International Commission on Radiological Protection (ICRP) [53] has defined the term representative person as a person who may be hypothetical, but whose habits (e.g. consumption of foodstuffs, breathing rate, location, usage of local resources) are typical of a small number of individuals representative of those most highly exposed. The ICRP notes that the habits of the representative person should not be the extreme habits of a single member of the population.

It is not possible to state the particular computational tools that would be used in later stages of the MRWS Site Selection 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. We intend to use industry-standard tools that are available at the time each update of the OESA is conducted.

2.1.4 Using the OESA

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

At the current stage of the MRWS Site Selection Process, the generic OESA presents 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

In this generic OESA we present an assessment of operational discharges associated with all wastes and materials covered by the Baseline Inventory described in Section 1.2. The assessment utilises a combination of quantitative and qualitative arguments which are described in detail in Section 4.

In Section 1.2, the derivation of the Derived Inventory which is used throughout the DSSC documentation suite as the underpinning dataset for design and safety assessment studies, is described. In the case of the OESA quantitative assessments, we have been required to assess gas generation from packaged LLW and ILW. Such calculations are not available to us based on the Derived Inventory defined in [7] but for its predecessor based on the 2004 UK Radioactive Waste Inventory. This will lead to some differences in the results but would not lead us to draw significantly different conclusions.
As we develop the OESA further we will commission further work to update the calculations based on the latest Derived Inventory to confirm that the results are not significantly different.

In addition to the Derived Inventory, the DSTS [7] also describes an Upper Inventory which has been developed to enable consideration to be given to the uncertainty in volume and composition, from consideration of alternative scenarios e.g. for fuel reprocessing, nuclear power generation and waste conditioning [54]. The impact of the Upper Inventory is assessed in a qualitative fashion.

In addition to the emplaced radioactive inventory, the host geology may also be a source of naturally occurring radon. We do not at this stage have any indication of the radioactivity that may arise from naturally occurring radon and note the need to assess the implications of this source when we produce a site specific OESA.

2.3 Discharges

Solid, liquid and gaseous radioactive discharges potentially arising from the routine operations described in Section 2.4 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 a 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 a GDF, e.g. groundwater discharges during construction. These would be considered 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 MRWS process.

Strategic Environmental Assessment (SEA) and EIA form part of the environmental assessments which are undertaken at the appropriate stages of a GDF programme in compliance with the relevant directives [55]. These assessments help to identify the potential environmental, social and economic impacts of implementing a GDF.

The waste storage and packaging requirements, described in Section 1.4.1, ensure that discharges, both radioactive and non-radioactive and as associated directly with the wastes, are appropriately minimised. This is discussed further in Section 4.

2.4 Operations

As described in Section 1.4, different illustrative disposal concept examples have been defined for three generic geological settings. These concepts assume co-location of ILW/LLW and HLW and spent fuel (and materials that might be declared as waste) in a single facility. The illustrative geological disposal concept examples 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.
- ILW/LLW Disposal Area
  - Including a shielded inlet cell (or a separate inlet cell for each vault) where unshielded package(s) would be removed from the shielded waste transport container and where inspection and monitoring activities would be undertaken.
- HLW/Spent Fuel Disposal Area
  - Each HLW/Spent Fuel waste container is assumed to be taken underground in its transport container within a shielded wagon. The waste container would then be removed from the transport container in a reception area and then transported to the disposal tunnel for disposal.

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.

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 contaminants before they reach the surface, such that any releases to the atmosphere are below regulatory limits.
- Depending on the geological setting, there may be groundwater intrusion into the active areas during the operational period; this would be managed and kept to a minimum level. Any liquid ingress would be diverted away from waste packages and collected to minimise the potential for corrosion or contamination.
- Once all wastes are emplaced, a decision could be taken to close the GDF. Backfilling with a cementitious material could take place prior to sealing and closure. Such backfilling has the effect of heating the waste packages as it cures (cement curing is exothermic) and backfilling may make available free water that can be used in e.g. corrosion reactions. Emplacement of a cementitious backfill emplacement could therefore contribute to a short-lived enhancement of corrosion rates and hence gas generation rates during the operational period.

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.
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 (e.g. building rubble, non-radioactive scrap). These will be considered in a future version of the OESA.
3 Safety objectives

RWMD 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 [56] and the Radiological Protection Policy Manual [48]. The Disposal System Specification (DSS) [6, 7] 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 [56] define a consistent set of principles which apply to all elements of a geological disposal facility and associated waste transportation 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. As part of the engineering design process, more detailed design-specific SFRs will be developed with input from hazard identification and assessment processes. 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 safety functional requirements as part of the iterative design process described in the DSS.

3.2 Radiological protection policy manual

The RWMD Radiological Protection Policy Manual (RPPM) [48] sets out the radiological protection policy and criteria within and against which all work undertaken by RWMD directed towards the implementation of geological disposal is to be undertaken. The policy and criteria presented in the RPPM are consistent with statutory radiological protection requirements and applicable guidance. Conformance with the provisions of the RPPM 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.

For the OESA the relevant criteria against which the results of this assessment will be compared are presented in Table 3.

Table 3 Dose limits and targets for normal operation to be applied during the period of licensing and/or environmental permitting

<table>
<thead>
<tr>
<th>Dose (mSv/y)</th>
<th>Effective dose limit(^{(1)})</th>
<th>Maximum value of dose constraint to members of the public from a new facility(^{(2)})</th>
<th>Design target(^{(3)})</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>0.15</td>
<td>0.01</td>
<td></td>
</tr>
</tbody>
</table>

Notes:

1. The statutory effective dose limits are those laid down by the Ionising Radiations Regulations 1999 [47], Regulation 11 and Schedule 4. In addition the Environmental Permitting (England and Wales) Regulations 2010 [46] and the Radioactive Substances (Basic Safety Standards) Regulations (Northern Ireland) 2003 [57] require that the sum of the doses resulting from the exposure of any member of the public to ionising radiation does not exceed the dose limits set out in Council Directive 96/29/Euratom (the EC Basic Safety Standards Directive) [58].
2. The Environmental Permitting (England and Wales) Regulations 2010 and the Radioactive Substances (Basic Safety Standards) Regulations (Northern Ireland) 2003 require the regulators to have regard to the following maximum doses to individuals which may result from a defined source, noting that any member of the public may be exposed to more than one source or site, for use at the planning stage in radiation protection:

- 0.3 mSv per year from any source from which radioactive discharges are made; or
- 0.5 mSv per year from the discharges from any single site.

The GRA [13] notes that for the operational and active institutional control phases, the Health Protection Agency (HPA) 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 RPPM we have adopted the lower constraint recommended by the HPA as it represents a lower design target.

3. The Health and Safety Executive’s Safety Assessment Principles [59] provide a Basic Safety Objective of 0.02 mSv in a calendar year to ‘any person off the site’. The 2009 Statutory Guidance to the Environment Agency states that where the prospective dose to the most exposed group of members of the public is below 0.01 mSv/y, the Environment Agency should not seek to reduce further the discharge limits that are in place, provided that the operator applies and continues to apply Best Available Techniques. The lower target provided in the 2009 Statutory Guidance to the Environment Agency is used as the design target.

3.3 Statutory basis

As discussed above, the RWMD RPPM has been developed to be consistent with statutory radiological protection requirements and applicable guidance. The relevant statutory requirements relating to the limits and targets set out in Table 3 and hence to this assessment are:

- Council Directive 96/29/Euratom [58], laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionising radiations;
- Health and Safety at Work etc Act 1974 (HSWA74) [60];
- Nuclear Installations Act 1965 (as amended) (NIA65) [61];
- Ionising Radiations Regulations 1999 (SI 1999/3232) (IRR99) [47];
- The Radioactive Substances Act 1993 (RSA93) [62];
- Environmental Permitting (England and Wales) Regulations 2010 (SI 2010/675) [46];
- Radioactive Substances (Basic Safety Standards) Regulations (Northern Ireland) 2003 [57].

3.4 Regulatory guidance and other material

In addition to the legislation listed above, international recommendations and guidance and UK guidance has been issued which is of relevance to this assessment. These are discussed below.

ICRP103, Recommendations of the ICRP, 2007 [53]

The International Commission on Radiological Protection (ICRP) provides guidance on protection against ionising radiation. The legal limits for radiation exposure contained in the IRR99 were based on its recommendations published in 1990 (ICRP60). The ICRP recommendations have been revised through ICRP103 but there are no changes to the recommended dose limits.
Application of the 2007 Recommendations of the ICRP to the UK, Advice from the Health Protection Agency, 2009 [63]

The Health Protection Agency (HPA) is responsible for advising government departments on radiological protection. The HPA 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.


Issued by the Health and Safety Executive (HSE), the principles are intended for use as a basis for the Nuclear Installations Inspectorate’s own safety assessment work. However, the HSE recognises that it is helpful for licensees under the NIA65 to have knowledge of the safety principles against which their plant will be judged.

Radioactive Substances Regulation (RSR) – Environmental Principles, 2009 [64]

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 on Requirements for Authorisation (GRA) documents (see below) and take account of relevant RSR Environmental Principles.

Statutory Guidance to the Environment Agency concerning the regulation of radioactive discharges into the environment, 2009 [65]

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 a suitable underpinning to the Statutory Guidance.


The geological disposal GRA was issued by the Environment Agency and the Northern Ireland Environment Agency. 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. The new regulations include provisions broadly similar to those of RSA 93 but within a common framework for environmental permitting across different regulatory regimes.
Radiological Protection Objectives for the Land-based Disposal of Solid Radioactive Wastes, Advice from the Health Protection Agency, 2009 [66]

The HPA 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.

National Dose Assessment Working Group (NDAWG) [50]

NDA RWMD is a member of NDAWG. The aim of NDAWG is to bring together people and organisations with responsibility for and/or an interest in, the assessment of radiation doses to the public from the operation of the nuclear industry and from minor users of radioactivity. The main focus of the work of NDAWG is past, present and future authorised discharges and direct radiation.

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

There are some countries that set national criteria and procedures for the protection of wildlife (e.g. USA and Australia). However, other countries (e.g. France and the UK) have no regulatory requirements for assessing the impact of radionuclides on wildlife. Reviews undertaken by the International Atomic Energy Agency (IAEA) and United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) of the effects of ionising radiation on organisms (reported respectively as [67,68]) give a broad conclusion that exposure of terrestrial organisms to dose rates of 40 μGy per hour, and exposure of aquatic organisms to dose rates of 400 μGy per hour would be unlikely to lead to observable effects in these populations. There is currently international work being undertaken to establish data and methodologies for the conduct of assessments of the effect of ionising radiation on non-human biota that can used as part of the normal regulatory activity [69].

The results of the assessment of radiological impact on non-human biota are presented in Section 6.
4 Derivation of off site discharges

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

4.1.1.1 Solid radioactive discharges

The facilities associated with surface based activities are identified in Section 2. Solid waste such as plastic, paper, clothing, wood and metallic items may arise from routine monitoring and maintenance activities. It is currently assumed that any solid radioactive waste arising from activities taking place in the surface facilities will be LLW and would be disposed of within the GDF. Therefore there would be no solid radioactive waste discharged off site from the surface facilities.

4.1.1.2 Solid non-radioactive discharges

Similarly, it is expected that there will be no solid non-radioactive but otherwise hazardous waste discharged off site 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 herein.

4.1.2 Underground facilities

4.1.2.1 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. As with the surface based activities, it is currently assumed that most radioactive solid waste arising from these activities would be LLW (there could conceivably be some ILW) and would be disposed of within the GDF. This would require provision of appropriate waste packaging and handling facilities within the GDF and definition of appropriate waste acceptance criteria. Any such solid waste arising would need to be compared with on-site waste acceptance criteria and following suitable conditioning and packaging is assumed to be determined to be compliant and accepted for disposal on site. Therefore there would be no solid radioactive waste from the underground facilities discharged off site.

4.1.2.2 Solid non-radioactive discharges

Similarly, it is expected that there will be no solid non-radioactive but otherwise hazardous waste discharged off site 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 herein.
4.2 Liquid discharges

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

4.2.1 Surface facilities

4.2.1.1 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 would be collected and treated as appropriate in an Active Effluent Treatment Plant (AETP). Treated liquids would be monitored prior to discharge to ensure that all discharges from the site are kept within acceptable limits. Liquid discharges from the surface facilities will be managed as described above and are not expected to be significant, and are therefore not further assessed in this report. This will be confirmed in future updates.

4.2.1.2 Liquid non-radioactive discharges

Surface drainage will be designed to prevent liquid effluents and run-off from entering the underground part of the facility [7]. Any liquids which may contain hazardous non-radioactive materials will be analysed and treated as appropriate prior to discharge from the facility to ensure compliance with relevant standards. Therefore, it is expected that there will be no liquid non-radioactive discharges from the surface facilities associated with the wastes for emplacement in a GDF.

4.2.2 Underground facilities

4.2.2.1 Liquid radioactive discharges

There are two 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 geological setting, there may be groundwater intrusion into the active areas during the operational period; this would be managed and kept to a minimum level. Any liquid ingress would be diverted away from waste packages and collected to minimise the potential for corrosion or contamination. Any groundwater collected would be monitored and discharged to ensure compliance with applicable limits; this is expected to be through a system separate to that relating to any potentially radioactive liquid wastes.

Radioactive liquid effluent discharges from the underground facilities would be managed as described above and are not expected to be significant, and are therefore not assessed further in this report.

4.2.2.2 Liquid non-radioactive discharges

As described above, groundwater intrusion from the host rock into the facility would be managed in the underground facilities during operation [8].

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

4.3 Aerial discharges

The components within a GDF that have the potential to generate gases and are of most relevance to gas production would include:

- the wastes, including both metals (e.g. steels, Zirkaloy, Magnox, aluminium and uranium) and organic materials (e.g. cellulose and synthetic polymers such as polyvinylchloride (PVC)) \[4\];
- the waste encapsulants, such as a cement grout or organic polymer;
- the container materials, including iron or steel (the most likely to be used for ILW / LLW containers) and some of a variety of metals, including steel, copper, titanium and nickel alloys, being considered for containers for other higher activity wastes \[24\];
- the buffer or backfill materials (through radiolysis of associated water); and
- structural materials, such as steel reinforcement used in underground construction.

The main mechanisms by which gas could be generated in a GDF have been reviewed in a joint EC/NEA status report \[70\]. These are metal corrosion, radiolysis and microbial degradation. Thus, in a 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 carbon-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;

but would also include:

- diffusion, notably the release of tritium by solid-state diffusion from metals;
- radioactive decay of radium, which leads to the generation of radon-222; and
- the release of radioactive gases containing tritium or carbon-14 by leaching of irradiated graphite.

The rates at which most of the gases will be generated 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. The process of backfilling the GDF, as may occur during the operational period, could affect the temperature and hence rate of gas generation.

The operational activities which may give rise to off site aerial radioactive and non-radioactive discharges are described in the following sections for both the surface facilities and those underground.
4.3.1 Surface facilities

4.3.1.1 Aerial radioactive discharges

Operations in a number of the surface facilities will give rise to limited 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 would be monitored and any particulates controlled with High Efficiency Particulate in Air (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.

It is considered that maximum aerial discharges would occur from the underground disposal vaults when all waste packages have been transferred underground and emplaced. Therefore, radioactive aerial discharges from the surface facilities are not explicitly assessed in this report, except to note that discharges would be significantly lower than those assessed from the underground vaults when containing the full inventory for disposal.

4.3.1.2 Aerial non-radioactive discharges

As with the radioactive aerial discharges, it is considered that maximum aerial discharges would occur from the underground disposal vaults when all waste packages have been transferred underground and emplaced. Therefore, non-radioactive aerial discharges from the surface facilities are not explicitly assessed in this report, except to note that discharges would be significantly lower than those assessed from the underground vaults when containing the full inventory for disposal.

4.3.2 Underground facilities

4.3.2.1 Aerial radioactive discharges

The main point of release for aerial discharges from the underground facilities would be through the ventilation system and out via a discharge stack. The discharge stack height and location would be designed to meet with the requirements specified in [7]. However, these are detailed aspects of a GDF that can only be defined once the actual site topology, height and proximity of the surrounding buildings have been established. As a consequence, for the purpose of assessing the effects of aerial discharges from a GDF, dispersion modelling has been undertaken using an assumed discharge stack height of 15 metres.

Ventilation to all of the radioactive areas underground would be supplied via a drift or shaft and discharged through a dedicated ventilation shaft [8]. Potentially radioactive aerial discharges from the vaults would be monitored prior to discharge. Ventilation to construction areas, when construction is taking place in parallel with operational activities, would be supplied via fresh air from a shaft and discharged from a separate shaft and segregated from potentially radioactive gaseous discharges.

The main source of aerial discharges from the radioactive areas underground would be gases emanating from the emplaced wastes. These gases are discussed further in Sections 4.4 and 4.5.
4.3.2.2 Aerial non-radioactive discharges

Flammable gases (mainly hydrogen but also possibly methane) generated by the emplaced wastes would be managed, through ventilation, to keep the concentration of these gases below their lower flammability limit in air. Asphyxiant and chemotoxic gases would also be managed through ventilation and, where necessary, controlled access underground to minimise exposure of underground workers.

Non-radioactive gases released off site are therefore expected to be greatly diluted in the ventilation gases. Any non-radioactive particulate material would be controlled by HEPA filtration prior to discharge. Therefore, non-radioactive aerial discharges from the underground facilities are not assessed further in this report.

4.4 Aerial radioactive discharges – inventory

As noted previously this generic OESA presents an assessment of operational discharges associated with all wastes and materials covered by the Baseline Inventory described in Section 1.2. The assessment utilises a combination of quantitative and qualitative arguments and methods which are described later in this section. As described in Section 2.2 the quantitative assessment of aerial discharges from LLW and ILW is based on gas generation calculations undertaken for the 2004 Radioactive Waste Inventory. This will lead to some differences when compared to equivalent calculations based on the Baseline Inventory defined in the DSTS [7] but these are not expected to be significant and would not lead us to draw significantly different conclusions. This will be confirmed in future issues of the OESA where the calculations will be repeated for the Baseline Inventory dataset.

In addition to the Derived Inventory, the DSTS [7] also describes an Upper Inventory which has been developed to enable consideration to be given to the uncertainty in volume and composition, from consideration of alternative scenarios e.g. for fuel reprocessing, nuclear power generation and waste conditioning [54]. The impact of the Upper Inventory is assessed in a qualitative fashion.

The discussion above concludes that the key radioactive discharges of interest from a GDF would be aerial discharges released via the discharge stack from the total waste inventory when emplaced in the underground facilities. The rate of gas generation and hence aerial radioactive discharge can be conservatively calculated using gas generation data for the total inventory of waste to be disposed of in a GDF.

HLW, spent fuel, plutonium and HEU are assumed in the generic DSSC to be packaged in high integrity, unvented disposal canisters. Radioactive discharges of gaseous radionuclides from these disposal canisters would not be expected under normal operating conditions [33] and therefore do not contribute to the aerial discharge from the GDF.

LLW, ILW and DNLEU are assumed in the DSSC to be packaged in containers that would be vented. Aerial discharges associated with gases generated in LLW and ILW containers are assessed in this report; the key radionuclides are identified as tritium, carbon-14 and radon-222. DNLEU does not generate tritium or carbon-14 although will generate radon-222 through radioactive decay. The generation of radon-222 from this source will be small in comparison to the source-term from LLW and ILW and as described later these stocks of uranium can be packaged in a form that will allow the radon to decay before it is released from the package and into the ventilation circuit.

Future updates to the OESA will consider this radon source in a quantitative assessment. We do not present a quantitative assessment relating specifically for the Upper Inventory: the additional tritium and carbon-14 inventory associated with the ILW and LLW in the Upper Inventory is not a significant gas generation source term during the aerobic GDF operational period. The impact of radon in the Upper Inventory is not thought to represent...
a significant challenge to the GDF due to the availability of packaging technologies that can provide significant hold-up of radon in the waste package. Future updates to the OESA will consider the issue of radon in the Upper Inventory in a quantitative assessment.

Recent calculations [71] of the generation rates of gases from ILW and LLW in a GDF in a higher strength host rock have been made based on an update to the 2004 UK radioactive waste inventory data [72]. The results of these calculations and their underpinning assumptions are summarised in the gas status report [33] and form the basis of the assessment reported in this generic OESA.

We note that there are other radionuclides, including those that are gaseous species in their own right (e.g. krypton-85) or that can be incorporated into volatile species (e.g. selenium-79 in hydrogen selenide or dimethyl selenide), these are expected to be of less significance [73] and have therefore not been assessed in this report.

The discussion below provides details of the source terms within the wastes that could give rise to gaseous discharges from the packages during the operational period.

### 4.4.1 Carbon-14

There are three main intermediate level waste types leading to generation and release of carbon-14: irradiated metals, irradiated graphite and organic wastes. To make a conservative assessment basis we make the pessimistic assumption that these releases are in the form of methane rather than the more reactive form of carbon dioxide which would be trapped by carbonation of grout in the waste packages [74].

In [33], it is noted that gas generation calculations assume carbon-14 is released from the corrosion of irradiated metals and the congruent reaction of carbon-14 bearing carbides in the form of methane, and the rate of release is assumed to be proportional to the corrosion rate. The release of carbon-14 from irradiated graphite is assumed to be proportional to its degradation rate and that 1% of the carbon-14 released would be in the form of methane.

Microbial degradation of organic molecules containing carbon-14 may result in the formation of carbon dioxide and methane. Carbon-14 bearing methane is also formed from the radiolysis of some organic molecules. Widespread methanogenesis is unlikely during the operational period although the possibility of localised niches within some individual waste packages exists. Carbon dioxide generated by microbial action is assumed to be trapped by the wasteform grouts.

As discussed in Section 1.4, we undertake assessment of the disposability of proposed waste packages before they are manufactured and work with the packager to ensure that the barriers provided by wasteform and container meet the anticipated needs for disposal in a GDF [45]. In recognition of our disposability assessment process, the quantified assessment reported herein assumes that release of carbon-14 from the small quantities of waste that comprise small organic molecules is not significant since it is assumed that the limited number of relevant wastes would be specifically conditioned and packaged such that significant quantities of carbon-14 bearing methane would not be released during the operational period.

The ‘effective’ radionuclide inventory for carbon-14 at 2040 was calculated in the gas generation calculations [71] to have an activity of $1.54 \times 10^3$TBq for UILW and $6.41 \times 10^3$TBq for SILW/LLW. These activities were calculated using [72]. As noted in [71] these activities were then multiplied by the fraction of carbon-14 present in each waste material to derive a partitioning of carbon-14 activity at 2040. From this data set it is clear that the highest activity for carbon-14 is contained within the graphite present in the SILW/LLW, with smaller amounts of activity in the stainless and mild steel. The activity of the wastes are contained within;
• graphite,
• stainless steel,
• mild steel,
• zircaloy,
• nimonic alloy,
• magnox,
• uranium,
• corroded magnox,
• corroded uranium, and
• non-metals, various types.

4.4.2 Radon-222

Radon-222 is generated from radium-226, which is present in high concentrations in a limited number of intermediate level waste streams. In this assessment, such radium-bearing wastes have been assumed to be packaged in an appropriate manner to reduce their radon emissions (e.g. encapsulation in a low permeability material). As discussed above, the disposability assessment process facilitates the production of waste packages that are expected to be compliant with the needs of disposal in a GDF.

The ‘effective’ radionuclide inventory for radium-226 at 2040 was calculated in the gas generation calculations \[71\] to have an activity of 2.32x10^3TBq for UILW. There is calculated to be no ‘effective’ radium-226 inventory for SILW/LLW at 2040. These activities were calculated using \[72\].

DNLEU in the form of a uranium oxide (see Section 1.4.1.1) will also be a source of radium-226 and hence radon-222. DNLEU is not in secular equilibrium with its daughter products^{6}, and therefore the radium-226 activity in these waste streams will continue to increase during the period of operation. However, being in the oxide form, the releases are expected to be a small fraction of releases from the more mixed and heterogeneous forms present in ILW. Furthermore, when and if uranium stocks are-conditioned and packaged for disposal, the LoC disposability assessment process will be applied to ensure that the barriers provided by the wasteform and waste package are consistent with the requirements of the disposal safety case.

Based on the above reasoning, releases from DNLEU are not quantified further in this assessment. As noted above, future updates to the OESA will consider this radon source in a quantitative fashion to confirm the robustness of this approach.

4.4.2.1 Radon emanating from the host rock

In addition to the waste, the host rock itself may be a source of radon-222. This is a site-specific issue. As noted earlier in Section 2, radon-222 emanating from the host rock is not considered in this generic OESA but will be addressed at a later site-specific stage.

4.4.3 Tritium

Tritium may be present in ILW metals (e.g. fuel cladding) as hydrides or as dissolved hydrogen. It would also be present in other materials (e.g. irradiated graphite) and trapped

\(^6\) DNLEU is processed uranium i.e. it is not in the state it would be naturally occurring in the environment.
as tritiated water on desiccants. The major contributor to the generation of tritium during the operational period is the aerobic corrosion of metallic uranium.

The 'effective' radionuclide inventory for tritium at 2040 was calculated in the gas generation calculations [71] to have an activity of $2.54 \times 10^{5}$ TBq for UILW and $2.57 \times 10^{4}$ TBq for SILW/LLW. These activities were calculated using [72]. From this data set it is clear that the highest activity of tritium is contained within the stainless steel present in the SILW/LLW with smaller amounts in mild steel. The activity of the wastes are contained within:

- stainless steel,
- mild steel,
- zircaloy,
- aluminium,
- magnox, and
- uranium.

4.5 Derivation of gas generation rates for use in assessment of doses from off site radioactive aerial discharges

4.5.1 Carbon-14 bearing gases

The possible generation rates of carbon-14 in methane during the operational phase are discussed in the gas status report [33], based on the calculations and underpinning assumptions reported in reference [71]. These gas generation rates are considered herein. As noted in Section 4.4.1, we assume that release of carbon-14 from wastes comprising small organic molecules does not need to be considered.

For most of the operational period of a GDF, the rate of carbon-14 bearing methane generation for all sources combined is estimated to be about 0.5 TBq per year.

This generation rate is calculated to increase if vaults are backfilled with a cementitious material. Such backfilling has the effect of heating the waste packages as it cures (cement curing is exothermic) and backfilling may make available free water that can be used in e.g. corrosion reactions.

The assumptions used in reference [71] lead to a maximum generation rate of 6 to 8 TBq per year on backfilling. These are currently expected to represent likely upper bounds during the operational period prior to closure of the facility [33]. Immediately after the closure of a GDF (a period excluded from consideration in the OESA), the rate of carbon-14 gas generation peaks at about 8 to 10 TBq per year [33, 71]. On the basis of the calculations reported in reference [71], and acknowledging the uncertainties in the rate of gas generation, we have used a selection of values for the carbon-14 bearing gas generation rates during the operational period:

- 0.5 TBq per year (representative of the operational period average gas generation rate); and
- 6 TBq per year (representative of the operational period peak gas generation rate).

We also consider a carbon-14 bearing gas generation rate of 10 TBq per year – this is taken herein to be a conservative bounding gas generation rate, noting that this does not imply such a generation rate is expected to occur in the timeframe considered by the OESA.

---

7 Higher values in the range 20 to 30 TBq per year arose during the post-closure phase in some variant calculations.
4.5.2 Radon-222

In the case of radon-222, a generation rate of approximately 1,000 TBq per year has been estimated for a radium-226 inventory of 23.2 TBq at 2040, as reported in [71] on the basis of the updated 2004 inventory [72]. However, this generation rate is conservative and not appropriate for estimating release to the atmosphere, as it does not include any retention and decay of radon-222 within waste packages. Retention of radon-222 by waste packages can reduce release rate significantly as a consequence of its relatively short half-life (3.82 days).

The retention of radon-222 within a waste package is expressed in terms of an ‘emanation coefficient’, which corresponds to the fraction of radon-222 that is released from a waste package in comparison with the in-package radon-222 generation rate. [75]. In determining the rate of off-site release of radon-222 for this generic OESA, we have taken radon-222 in-package retention into account by assuming an emanation coefficient of 2x10^-3 [76]. For a total radon-222 generation rate of 1,000 TBq/year, this results in an off-site discharge release rate of 2 TBq per year, which is used for assessment purposes in this generic OESA. As we discuss later, there is the potential to package radon-bearing wastes in a form that provides improved ‘hold-up’ and which offers the potential to justify a much reduced emanation rate.

As noted earlier, additional radon-222 associated with the DNLEU is not considered significant in comparison to discharges from LLW and ILW, and is not considered in the quantified assessment. The robustness of this approach will be considered in a future update of the OESA.

4.5.3 Tritium

The possible generation rates of tritium during the operational phase are discussed in the gas status report [33] based on the calculations reported in reference [71].

Assuming a GDF operational history as per that discussed in Section 4.5.1 for carbon-14 bearing gases, and acknowledging the uncertainties in the rate of gas generation, we have used a selection of values for tritium generation rates during the operational period:

- 1-2 TBq per year (representative of the operational period average gas generation rate); and
- 10 TBq per year (representative of the operational period peak gas generation rate).

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

Based on the analysis described above, Table 4 summarises the average, peak and bounding radioactive gas release rates that are used subsequently in this report in the dose assessment calculations for off site discharges.

Table 4 Radioactive gas release rates

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Average release rate for operational period (TBq per year)</th>
<th>Peak release rate for operational period (TBq per year)</th>
<th>Bounding release rate (TBq per year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Carbon-14</td>
<td>0.5</td>
<td>6-8</td>
<td>10</td>
</tr>
<tr>
<td>Radon-222</td>
<td>2</td>
<td>2</td>
<td>2</td>
</tr>
<tr>
<td>Tritium</td>
<td>1-2</td>
<td>10</td>
<td>10</td>
</tr>
</tbody>
</table>
5 Methodology for assessing doses from off site gaseous discharges

5.1 Guidance on methodology

Based on the consideration of source terms presented in Section 5, off site operational discharges of GDF-derived radioactive gases 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 off site operational discharges of GDF-derived radioactive gases for two endpoint receptors - the public and non-human biota. Doses that are calculated by applying these methodologies to the rates of gaseous discharges noted in Section 4 are reported in Section 6.

In relation to the calculation of dose to the public, the ICRP [53] has defined the term ‘representative person’ as a person who may be hypothetical, but whose habits (e.g. 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, 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 [77,78]. 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 National Dose Assessment Working Group (NDAWG) has recommended this approach for the assessment of prospective public dose during the operational period of a licensed facility [79].

The Environment Agency guidance [77] recommends a staged approach to the assessment of critical group doses⁸. The first stage consists of a simple and cautious assessment of the critical group dose (initial radiological assessment), and the second stage takes into account local dispersion in the air or water as appropriate.

The first stage of the Environment Agency guidance [77] provides a system for undertaking an initial cautious prospective assessment of the dose arising from sources of radioactive waste discharged to the environment. It is not a requirement to use this approach; however, in the absence of a specific site it provides a good initial assessment methodology to follow to gauge off site doses to members of the public during the operational period of a GDF. This approach has been followed for the generic OESA presented in this report, for releases to the environment.

For doses to non-human biota, a separate assessment has been undertaken using the ERICA [80] (Environmental Risk from Ionising Contaminants: Assessment and Management) assessment tool; this is described in Section 5.3.

5.2 Methodology for assessment of doses to members of the public

To assess off site doses to members of the public, it is necessary to establish the off site operational discharges of GDF-derived radioactive gases (this is described in Section 4 of this report). The key radionuclides that have been identified are tritium, radon-222 and carbon-14; the assessment methodology used to calculate the radiological impact of gaseous species containing these radionuclides is described herein.

---

⁸ The term 'critical group' has now been replaced with 'representative person' as defined in [53].
The Environment Agency guidance has calculated the Dose Per Unit Release (DPUR) for 100 different radionuclides. Supporting the Environment Agency guidance is a methodology report [78] that describes how the DPUR values were calculated using the 1998 version of the NRPB model PC CREAM [81]. PC CREAM is based on a methodology for assessing the radiological consequences of routine releases to the environment published by the European Commission [82]. The model contains different modules (e.g. FARMLAND and PLUME) which are used to calculate the transfer of radionuclides through e.g. atmospheric and coastal environments, and the food chain. PC CREAM and its underlying dispersion models are robust, fit for purpose and have been verified against environmental data [83,84,85].

The Environment Agency guidance provides DPUR values that have been calculated for different groups of the public. The exposure group considered in the OESA is the resident family who lives close to the operational GDF.

The exposure pathways considered by the Environment Agency methodology when deriving a radiological dose are:

- inhalation of radionuclides in the effluent plume
- external irradiation from radionuclides in the effluent plume
- external irradiation from radionuclides in the effluent plume deposited to the ground
- consumption of terrestrial food incorporating radionuclides deposited to the ground.

The Environment Agency methodology calculates DPURs for seven different groups of the public (based on their location and habits) and to four age groups (including the foetus).

In this report, doses to members of the local resident family who may receive doses as a result of discharges from a GDF have been calculated. The term “offspring” has been used to collectively denote the embryo, foetus and newborn child [86]. The DPUR for the most limiting age group for each radionuclide (that is the age group who would receive the largest dose) is presented in Table 5.

### Table 5  Dose per unit release factors for local resident family

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>DPUR for worst age group local resident family (µSv per year, per Bq per year of discharge to atmosphere, for a release at ground level)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Terrestrial food consumption DPUR</td>
</tr>
<tr>
<td>Tritium</td>
<td>5.7x10^{-13}</td>
</tr>
<tr>
<td>Radon-222</td>
<td>0</td>
</tr>
<tr>
<td>Carbon-14</td>
<td>3.3x10^{-11}</td>
</tr>
</tbody>
</table>

Key assumptions in the Environment Agency methodology are that the release from a GDF is at ground level, the local resident is assumed to be 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.

Stage 2 of the Environment Agency guidance [77] allows for some scaling to be applied to account for the dispersion conditions. It is assumed in this assessment that the release height from the discharge stack of a GDF is 15 metres. The Environment Agency provides separate scaling factors for inhalation and external dose pathways, and the food pathway. This is because the location of exposure of the local inhabitant is assumed to be nearer the release point than the location of the food source. The scaling factors for a discharge stack height of 15 metres are 0.09 for inhalation and external irradiation, and 0.45 for food consumption.
The gaseous discharges from the source term (as discussed in Section 4) can then be multiplied by the scaled DPUR factors, in order to calculate doses to members of the public from terrestrial food consumption, external irradiation and inhalation pathways. The equation for doses to members of the public is presented below:

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

\(A = \) Gaseous discharge (Bq per year)
\(B = \) Food DPUR
\(C = \) External DPUR
\(D = \) Inhalation DPUR
\(E = \) Food dose scaling factor
\(F = \) Inhalation and external dose scaling factor

The resultant calculated doses are presented in Section 6.

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

5.3.1 Reference organisms

Assessments of the potential impact of radiation on non-human biota are undertaken by identifying the nature of the ecosystem and a range of characteristic ‘reference organisms’ within it \([36]\). A range of such organisms is generally chosen to encompass different trophic levels\(^9\) and variations in lifestyle, geometry and uptake characteristics that determine exposure. The range of reference organisms for the terrestrial ecosystem that are relevant to an assessment of atmospheric releases are:

- amphibian
- bird
- bird egg
- detritivorous invertebrate
- flying insect
- gastropod
- grasses and herbs
- lichen and bryophyte
- mammal (deer)
- mammal (rat)
- mammal (fox)
- reptile
- shrub
- soil invertebrate
- tree

\(^9\) The term ‘tropic levels’ refers to a group of organisms that occupy the same position in the food chain.
Extensive databases of the occupancy, geometry, concentration factor and dose conversion coefficients necessary to assess dose to such organisms have been established over the last decade, as part of a series of EC-funded projects. Notably, the EC 6th Framework project ERICA [80] (Environmental Risk from Ionising Contaminants: Assessment and Management) was completed in February 2007 and provided an integrated approach to the assessment and management of environmental risks from ionising radiation. It included the development of the assessment tool which was used in this report.

It should be noted, however, that impacts on sensitive species, indicator species, endangered or locally important species may not be fully represented by the assessments undertaken for reference organisms. Additional assessments and alternative dose rate criteria may be necessary in such cases. However, we are not aware of any of these species at this time as we do not have a site for a GDF.

5.3.2 Routes of entry to the environment

Potential pathways from an initial gaseous discharge can be summarised as follows:

- Non-human biota (fauna) – inhalation, ingestion (including water uptake), aerosol skin contact.
- Non-human biota (flora) – metabolic absorption (respiration, photosynthesis), uptake from soil, surface absorption.

For scoping purposes in this assessment, non-human biota were assumed to be located at a distance of 300 metres from the point of release from a GDF. The releases were then scaled (as was done with the doses to member of the public) for a discharge stack height of 15 metres.

The transfer of radionuclides from the air to particular reference organisms is modelled on the basis of equilibrium concentration ratios; default factors for tritium and carbon-14 were used in this study, as taken from the ERICA assessment tool [80] to determine internal activity concentrations for each reference organism. Dose rates from these radionuclides were calculated on the basis of default dose conversion coefficients for internal exposure. For assessment of doses from radon, the approach developed by Vives et al [87], which details a method for deriving respiration rates to calculate the dose rates of daughter products of radon-222 to terrestrial animals, has been used to input data to the ERICA assessment tool.

To assess external exposure to reference organisms, it is necessary to make some assumptions about the time spent in different parts of the environment (occupancy above, on and within soil); default occupancy factors from the ERICA assessment tool were applied.
6 Safety analysis

In this section we:

(a) summarise qualitative supporting arguments for the operational environmental safety of a GDF (see Section 6.1); and

(b) report doses from off site gaseous discharges as calculated by applying the methodologies noted in Section 5 and the rates of off site operational discharges of GDF-derived radioactive gases noted in Section 4 (see Section 6.2 onwards). Two endpoint receptors - the public and non-human biota – are considered.

6.1 Qualitative safety arguments

Our high-level strategy for ensuring operational environmental safety is to eliminate hazards during normal operation of a GDF and, where this is not possible, to provide protection to control environmental impacts. The generic requirements on a GDF to meet this safety strategy are set out in the Disposal System Technical Specification [7]. 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 presents examples of supporting evidence to build confidence in our OESA:

- The wastes will be packaged before transport to a GDF in accordance with detailed waste package specifications. Strict quality assurance requirements are imposed to ensure these packaging specifications are met.

- All waste packages will be assessed through a waste acceptance process prior to dispatch and acceptance into a GDF.

- The packaged waste has a number of inherent safety features:
  - The wastes are solid, or have been solidified before transport to a GDF. Wastes other than SF will have been encapsulated in a stable matrix.\(^{10}\) The wastes do not contain liquids or pressurised gases. Any wastes that are expected to evolve significant gas (e.g. 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.
  - The wastes are packaged in a metallic or concrete container, which reduces the potential for release of radioactive materials during handling. The outer surface is essentially free of loose contamination.
  - The wastes are disposed of in robust containers, which provide radiation shielding as required, and are capable of normal handling during storage, transport and disposal operations.\(^{11}\)

- The most hazardous waste packages, including HLW/SF and some types of ILW, would be transported within reusable shielded transport containers that comply with the performance requirements for a “Type B” transport container, as defined in the

---

\(^{10}\) 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.

\(^{11}\) The containers also need to be able to withstand specific accident conditions (such as impact or fire) with little release of radioactivity or loss of shielding. This aspect does not form part of the ESC, but is considered in both the TSC [26] and OSC [27].
IAEA transport regulations. These regulations stipulate that a Type B transport container must remain intact even under conditions of severe impact, fire or immersion. We therefore expect no releases of radioactivity during their normal transport and handling. The unshielded waste packages would not be removed from their Type B transport containers until after they had been taken underground.

- 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; however, there will only ever be a relatively small number of such packages in the surface facilities at any one time, so we only expect minor environmental discharges at any time – these will be monitored.

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

- Waste would 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 would 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 would only ever be a relatively small number of such packages being transferred underground at any one time, so we only expect minor environmental discharges, which would be monitored. The air underground would be filtered to remove any particulate radioactivity that might escape from the packages.

- The unloading of unshielded waste packages from reusable transport containers would be carried out underground in cells equipped with high-integrity lifting equipment.

- There would be potential for public exposure from off site release of radioactive particulates as packages were lifted from their transport containers. This will be reduced by use of high-efficiency particulate in air (HEPA) filters at any discharge points of our underground ventilation system. There would also be potential for release of any radioactive gases that may have built up inside transport containers. These gases cannot be removed by filters, but would be monitored.

- The unloading and emplacement of other types of waste package would be undertaken using approaches specific to waste type and packaging. In particular, the emplacement of Type 2 Industrial Packages in disposal vaults may be done more simply and with less potential for release of radioactive particulates.

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

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

- Monitoring of the underground facilities during the operational period would detect any emissions or leakages of radioactivity. Any liquids would be collected, examined for radioactivity, and remediated if necessary in the surface facilities. Radioactive particulates in the air would be trapped by the use of HEPA filters at discharge points of the ventilation system. Our monitoring programmes would allow us to make informed decisions about management of discharges and the waste packages in the disposal areas. However, we may not easily be able to control 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, our 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.

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:

- In the case of SF, there are decades of experience of using Type B shielded transport containers for the safe transport of SF from power station sites to Sellafield for reprocessing. This involves transferring SF in and out of containers, and containers being loaded on and off rail vehicles using high-integrity equipment similar to that needed at a GDF.
- 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 a GDF. In Sweden there is significant operational experience of handling SF in an underground storage facility (CLAB).
- LLW is safely transported to and disposed of at the near-surface disposal facility, the LLWR 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 a GDF.
- Internationally there is currently one operating GDF – the WIPP facility near Carlsbad, New Mexico, which has been in operation since 1999. To date thousands of shipments of waste to this facility from across the US have been completed. GDFs in Finland and Sweden are nearing the licensing stage and a GDF for non-heat generating wastes has been licensed to start operating in 2013 in Germany.

### 6.2 Calculated doses to representative public: local resident family

Table 6 presents doses calculated using the average and peak gas release rates for the operational period to the local resident family receptor group (see Table 4), which is assumed:

- to be located 100 metres from a discharge point of a GDF; and
- to consume food grown at 500 metres from a GDF;

for discharges from a 15m stack height.
The total dose to members of the public for the operational period average gas generation rates is calculated to be 0.052 mSv per year. The majority of this dose arises from radon-222 (0.043 mSv), with minor contributions from carbon-14 bearing gas and tritium. This total dose is below the effective dose limit for members of the public of 1 mSv per year. It is also below the maximum dose to individuals which may result from the discharges from any single site, stated in the Environmental Permitting Regulations 2010, of 0.5 mSv per year, and below the maximum value of dose constraint to members of the public from a new facility of 0.15 mSv per year adopted by RWMD. However, although the dose from carbon-14 and tritium is below the design target of 0.01 mSv per year, the dose from radon-222 exceeds the design target.

Table 6 also presents doses calculated using the operational period peak gas generation rates to the local resident family receptor group (see Table 4). The total dose to members of the public is calculated to be 0.16 mSv per year. The majority of this dose (0.11 mSv per) arises from carbon-14, with contributions from radon-222 and tritium. This total dose is below the effective dose limit for members of the public of 1 mSv per year. It is also below the maximum dose to individuals which may result from the discharges from any single site, stated in the Environmental Permitting Regulations 2010, of 0.5 mSv per year. However, this assessment presents doses to members of the public from radon-222 and carbon-14 that exceed the design target of 0.01 mSv per year, and also the total dose is above the maximum value of dose constraint to members of the public from a new facility of 0.15 mSv per year adopted by RWMD.

Table 6  Sensitivity analysis: calculated doses from off site discharge to the local resident family receptor group using average and peak gas release rates for the operational period

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Discharge (Bq/yr)</th>
<th>Food DPUR from Table 5</th>
<th>External DPUR from Table 5 (µSv per year per Bq per year)</th>
<th>Inhalation DPUR from Table 5 (µSv per year per Bq per year)</th>
<th>Dose (mSv per year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tritium (operational average)</td>
<td>1 x 10^{12}</td>
<td>5.7 x 10^{-13}</td>
<td>-</td>
<td>1.4 x 10^{-12}</td>
<td>0.0004</td>
</tr>
<tr>
<td>Tritium (operational peak)</td>
<td>1 x 10^{13}</td>
<td>5.7 x 10^{-13}</td>
<td>-</td>
<td>1.4 x 10^{-12}</td>
<td>0.004</td>
</tr>
<tr>
<td>Carbon-14 (operational average)</td>
<td>5 x 10^{11}</td>
<td>3.3 x 10^{-11}</td>
<td>4.5 x 10^{-16}</td>
<td>2.4 x 10^{-10}</td>
<td>0.009</td>
</tr>
<tr>
<td>Carbon-14 (operational peak)</td>
<td>6 x 10^{12}</td>
<td>3.3 x 10^{-11}</td>
<td>6.4 x 10^{-17}</td>
<td>3.5 x 10^{-11}</td>
<td>0.11</td>
</tr>
<tr>
<td>Radon-222 (operational average and peak)</td>
<td>2 x 10^{12}</td>
<td>-</td>
<td>4.5 x 10^{-16}</td>
<td>2.4 x 10^{-10}</td>
<td>0.043</td>
</tr>
<tr>
<td>Total operational average</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0.052</td>
</tr>
<tr>
<td>Total operational peak</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0.16</td>
</tr>
</tbody>
</table>
Table 7 presents doses calculated using the bounding gas generation rates to the local resident family receptor group (noting that this does not imply such generation rates are expected to occur in the timeframe considered by the OESA).

### Table 7  Calculated doses from off site discharge to the local resident family receptor group using the bounding gas release rates

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Discharge (Bq/yr)</th>
<th>Food DPUR from Table 5</th>
<th>External DPUR from Table 5 (µSv per year per Bq per year)</th>
<th>Inhalation DPUR from Table 5 (µSv per year per Bq per year)</th>
<th>Dose (mSv per year)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tritium</td>
<td>$1 \times 10^{13}$</td>
<td>$5.7 \times 10^{-13}$</td>
<td>-</td>
<td>$1.4 \times 10^{-12}$</td>
<td>0.004</td>
</tr>
<tr>
<td>Radon-222</td>
<td>$2 \times 10^{12}$</td>
<td>-</td>
<td>$4.5 \times 10^{-16}$</td>
<td>$2.4 \times 10^{-10}$</td>
<td>0.043</td>
</tr>
<tr>
<td>Carbon-14</td>
<td>$1 \times 10^{13}$</td>
<td>$3.3 \times 10^{-11}$</td>
<td>$6.4 \times 10^{-17}$</td>
<td>$3.5 \times 10^{-11}$</td>
<td>0.180</td>
</tr>
<tr>
<td>Total</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0.23</td>
</tr>
</tbody>
</table>

The maximum resulting dose to members of the public is calculated in this scenario to be 0.23 mSv per year (see Table 7). The majority of this arises from discharges of carbon-14, with contributions from radon-222 and tritium. This dose is below the effective dose limit for members of the public of 1 mSv per year and may include some conservative assumptions in its derivation. It is also below the maximum dose to individuals which may result from the discharges from any single site, stated in the Environmental Permitting Regulations 2010, of 0.5 mSv per year. However, it is above the maximum value of dose constraint to members of the public from a new facility of 0.15 mSv per year adopted by RWMD, and presents a dose to members of the public from radon-222 and carbon-14 that exceeds the design target of 0.01 mSv per year.

Section 7 discusses these results further.

### 6.3 Calculated doses to terrestrial reference organisms

Table 8 presents calculated doses to non-human biota that are assumed be located at a distance of 300 metres from the point of release from a GDF; a stack height of 15m is assumed.

### Table 8  Calculated doses from off site discharge to the non-human biota receptor group using the bounding gas release rates

<table>
<thead>
<tr>
<th>Organism</th>
<th>15m stack height</th>
<th></th>
<th>radon-222</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Dose rate (µGy per hour)</td>
<td>Tritium</td>
<td>Carbon-14</td>
<td></td>
</tr>
<tr>
<td>Amphibian</td>
<td></td>
<td>$1.14 \times 10^{3}$</td>
<td>$3.52 \times 10^{-2}$</td>
<td>-</td>
</tr>
<tr>
<td>Bird</td>
<td></td>
<td>$1.14 \times 10^{3}$</td>
<td>$3.64 \times 10^{-2}$</td>
<td>$7.45 \times 10^{-4}$</td>
</tr>
<tr>
<td>Bird egg</td>
<td></td>
<td>$1.14 \times 10^{3}$</td>
<td>$2.34 \times 10^{-2}$</td>
<td>$1.95 \times 10^{-3}$</td>
</tr>
<tr>
<td>Detritivorous Invertebrate</td>
<td></td>
<td>$1.14 \times 10^{3}$</td>
<td>$1.13 \times 10^{-2}$</td>
<td>$5.34 \times 10^{-3}$</td>
</tr>
<tr>
<td>Flying insects</td>
<td></td>
<td>$1.07 \times 10^{3}$</td>
<td>$1.13 \times 10^{-2}$</td>
<td>$3.13 \times 10^{-3}$</td>
</tr>
</tbody>
</table>
### Table 8: Dose rate (μGy per hour) for Various Organisms

<table>
<thead>
<tr>
<th>Organism</th>
<th>15m stack height Dose rate (μGy per hour)</th>
<th>Tritium</th>
<th>Carbon-14</th>
<th>radon-222</th>
<th>Total</th>
</tr>
</thead>
<tbody>
<tr>
<td>Gastropods</td>
<td></td>
<td>1.14 x 10^-3</td>
<td>1.13 x 10^-2</td>
<td>-</td>
<td>1.24 x 10^-2</td>
</tr>
<tr>
<td>Grasses</td>
<td></td>
<td>1.14 x 10^-3</td>
<td>2.34 x 10^-2</td>
<td>1.45 x 10^-2</td>
<td>3.90 x 10^-2</td>
</tr>
<tr>
<td>Lichens and Bryophytes</td>
<td></td>
<td>1.14 x 10^-3</td>
<td>2.34 x 10^-2</td>
<td>1.45 x 10^-2</td>
<td>3.90 x 10^-2</td>
</tr>
<tr>
<td>Mammal (deer)</td>
<td></td>
<td>1.14 x 10^-3</td>
<td>3.64 x 10^-2</td>
<td>7.20 x 10^-4</td>
<td>3.83 x 10^-2</td>
</tr>
<tr>
<td>Mammal (rat)</td>
<td></td>
<td>1.14 x 10^-3</td>
<td>3.64 x 10^-2</td>
<td>1.73 x 10^-3</td>
<td>3.93 x 10^-2</td>
</tr>
<tr>
<td>Reptile</td>
<td></td>
<td>1.14 x 10^-3</td>
<td>3.64 x 10^-2</td>
<td>6.94 x 10^-4</td>
<td>3.83 x 10^-2</td>
</tr>
<tr>
<td>Shrub</td>
<td></td>
<td>1.14 x 10^-3</td>
<td>2.34 x 10^-2</td>
<td>1.45 x 10^-2</td>
<td>3.90 x 10^-2</td>
</tr>
<tr>
<td>Soil Invertebrates (worm)</td>
<td></td>
<td>1.14 x 10^-3</td>
<td>1.13 x 10^-2</td>
<td>2.65 x 10^-3</td>
<td>1.51 x 10^-2</td>
</tr>
<tr>
<td>Tree</td>
<td></td>
<td>1.14 x 10^-3</td>
<td>3.54 x 10^-2</td>
<td>1.45 x 10^-2</td>
<td>5.10 x 10^-2</td>
</tr>
<tr>
<td>Mammal (fox)</td>
<td></td>
<td>1.04 x 10^-3</td>
<td>1.86 x 10^-2</td>
<td>5.53 x 10^-4</td>
<td>2.02 x 10^-2</td>
</tr>
</tbody>
</table>

**Note:**

1. 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).

The dose rate associated with carbon-14 dominates for all species, followed by a smaller contribution from tritium and, to a much lesser extent, radon.

Reviews of the effects of ionising radiation on organisms [67, 68] give a broad conclusion that exposure of terrestrial organisms to dose rates of 40 μGy per hour, and exposure of aquatic organisms to dose rates of 400 μGy per hour, would be unlikely to lead to observable effects in these populations.

The total calculated dose rates for all the organisms considered in Table 8 are insignificant in comparison to these dose rates, and the assessment of dose to non-human biota from off site gaseous radioactive discharge from a GDF requires no further consideration in this report.
7 Discussion

Total doses to members of the public (local resident family receptor group) resulting from off site gaseous discharges relating to the operational period average and operational period peak gas generation rates are calculated respectively to be 0.052 mSv per year (dominated by radon-222) and 0.16 mSv per year (dominated by carbon-14) (see Table 6).

The following discussion identifies some of the conservatisms and uncertainties in 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 Uncertainties and potential conservatisms

7.1.1 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 a GDF, consuming food grown at 500 metres from a GDF. It is unclear whether or not these are conservative assumptions, but it should be noted that a receptor group living farther from a GDF would be expected to receive a lower off site dose during the operational period (as any gaseous discharges would 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 a gaseous discharge point has been demonstrated in previous assessments that have been undertaken [49]. Doses would also be affected by assumptions of residency time for the receptor group, and environmental factors such as average wind conditions; such factors are site-specific.

7.1.2 Carbon-14

The average release rate of carbon-14 bearing methane is estimated to be 0.5 TBq per year during most of the operational period. This release rate would give rise to a dose of 0.009 mSv per year using the approach noted and applied in Sections 5 and 6, which is at the design target. The peak release rate, occurring as a consequence of backfilling with a cementitious material, is estimated to be approximately 6 TBq per year. This would give rise to a dose of 0.11 mSv per year, which is below the maximum dose constraint.

The peak release rate calculation contains a number of pessimisms, as it assumes that all of the waste vaults are backfilled at the same time and that the temperature increases to 80°C uniformly across all waste vaults (affecting corrosion rates and hence gas generation rates). In reality, it is likely that backfilling would take place over a 10 year period and that the temperature would not reach 80°C in all areas of the waste vaults. If a lower temperature were to be assumed, then the peak discharge rate would be reduced; reference [88] infers that a maximum temperature of 40°C associated with backfill curing is appropriate.

Conversely, the study reported in [71] does not take account of the possible availability of water, related e.g. to in-drum grouting, within the waste packages during the operational period (prior to any backfilling operations undertaken as part of the GDF closure). This could lead to an increase in carbon-14 release in the timeframe of interest to the OESA, were the availability of this in-drum water to be explicitly considered as affecting gas generation and release.
There is also uncertainty in the inventory of waste that gives rise to discharges of carbon-14 bearing gases in the GDF operational period, in particular the ratio of corroded and un-corroded uranium metal (the corrosion of which leads to the peak discharge rate of carbon-14). If it is the case that more of the uranium metal has corroded before it is sent to a GDF for disposal than is currently assumed, then the release rates for carbon-14 bearing gases from the disposal vaults would be lower than assumed in this report. Furthermore, carbon-14 associated with small organic molecules is not considered as a gas source term herein, as it is assumed that the limited number of relevant wastes would be specifically conditioned and packaged such that significant quantities of carbon-14 bearing methane for this source would not be generated during the operational period.

7.1.3 Radon-222

The dose presented for radon in Section 6 (0.043 mSv per year) is below the maximum value of dose constraint to members of the public from a new facility (0.15 mSv per year) adopted by RWMD, but exceeds the design target (0.01 mSv per year); see Section 3.

This dose results from an assumed emanation coefficient of 2x10⁻³ (Section 4.5.2). However, since approximately 95% of the radium-226 in the Derived Inventory for ILW and LLW¹² is concentrated within a very small number of waste streams¹³, the opportunity exists for reducing this emanation coefficient (i.e. reducing the fraction of generated radon-222 that is released) by implementing bespoke packaging solutions. An example of such a solution is noted below.

Work undertaken to date as part of the LoC disposability assessment process for the development of packaging approaches for waste streams with a high radium-226 inventory has supported the use of emanation coefficients of less than 2x10⁻³ [89]. This work has shown that the use of a multi-barrier approach to waste packaging can result in emanation coefficients as low as 1.4x10⁻⁶. This is achieved by encapsulating the radium waste in small vessels with a polymeric grout and intimately grouting the encapsulated item using cement in a 500 litre drum waste package.

Assuming such an emanation coefficient (i.e. 1.4x10⁻⁶) could be applied to all of the radium-226 bearing inventory noted in Section 4.5.2, in place of the emanation coefficient selected therein (i.e. 2x10⁻³), would reduce the release rate of generated radon-222 and the associated dose to members of the public by up to three orders of magnitude - the design target (0.01 mSv per year) would then be met.

However, there remains uncertainty relating to the emanation coefficient that can be achieved in practice, and further work is required to confirm the use of values lower than 2x10⁻³. Further work is also required to better understand hold up of radon within the engineered barriers of the GDF and hence the rate of off-site discharge.

Note that only radon-222 associated with ILW is considered in this quantitative assessment. Qualitative arguments are used to claim that any additional radon-222 from DNLEU will be small in comparison. The robustness of this approach will be demonstrated though quantitative assessment to be included in future updates to the OESA.

7.1.4 Tritium

The calculated doses resulting from discharges of tritium are below the design target of 0.01 mSv per year. However, the uncertainties discussed with respect to carbon-14 (Section 7.1.2) and radon-222 (Section 7.1.3) also apply to tritium. Therefore in reality this dose could be reduced further. Waste packaging guidance [90] has been produced that if

¹² Based on the 2007 UKRWI.

¹³ ~90% of the total radium-226 in the ILW/LLW inventory is known to exist in a single waste stream.
followed would result in discharges of tritium remaining at a level that would not give rise to significant doses off site.

7.2 Mitigation measures

Additional steps could be taken to reduce the potential for gaseous discharges from a GDF that would, consequently, also reduce any off site doses.

- One of the key mitigation measures relates to the packaging of the wasteforms before they are dispatched to a GDF. Wastes that are accepted for disposal at a GDF must be treated, conditioned or packaged in such a way as to render them:
  - passively safe, such that it 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 it can be shown to be compliant with all of the relevant regulations and safety cases for transport to and disposal in the GDF [45].

This is undertaken through the RWMD disposability assessment process, which exists to support the waste packagers in their packaging of higher activity wastes to a standard compatible for disposal within a GDF [45]. Some potential specific packaging solutions for the gases of concern in this assessment were discussed in Section 7.1. Note that, pursuant to RWMD working practices, any issues identified with the release of gases from wastes in the development of the safety case (including the OESA component) can be discussed with the designers of a GDF and the waste packagers, to ensure that additional measures as required to mitigate the issues can be put in place at a stage in the waste packaging process prior to dispatching the waste to a GDF.

- In this assessment, the radioactive discharges are assumed to be dispersed from a 15 metre stack height. Increasing the height of the discharge stack would increase the dispersion of any gaseous discharges, and reduce the estimated doses to members of the public. We have undertaken some additional calculations to investigate the results of increasing the stack height to 30 metres. These results are shown in Table 9 and discussed below.

- A conservatism in this assessment is that the radon generation rate and discharge rate only account for an in-package hold-up factor; there is no accounting for the decay of radon as it travels through the ventilation system and out of the GDF through the discharge stack (this is relevant to radon-222 because of its 3.82 day half-life). A reduction in the discharge rate of radon-222 could potentially be achieved through a delay being introduced within the ventilation system, ensuring that off site dose associated with this radionuclide has been reduced as a result of radioactive decay.

- The bounding gas generation rates used in this report is derived from work reported in [71] that conservatively assumes that the backfilling of the GDF vaults happens simultaneously, once all the waste is emplaced and just prior to closing the facility. The rates of gas generation associated with backfilling could be managed by an alternative backfilling strategy, e.g. backfilling vaults in sequence and over a period of time once they are all filled with waste rather than instantly and simultaneously as assumed herein, to manage the temperature increase and hence the rates of gas generation.
Table 9  Dose calculations to members of the public from a varying discharge stack height

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Doses calculated using average release rates for operational period from Table 6 (mSv per year) for a stack height of 30m</th>
<th>Doses calculated using bounding release rates from Table 6 (mSv per year) for a stack height of 30m</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tritium</td>
<td>0.0001</td>
<td>0.001</td>
</tr>
<tr>
<td>Carbon-14</td>
<td>0.003</td>
<td>0.07</td>
</tr>
<tr>
<td>Radon-222</td>
<td>0.007</td>
<td>0.007</td>
</tr>
<tr>
<td>Total</td>
<td>0.01</td>
<td>0.078</td>
</tr>
</tbody>
</table>

Table 9 presents the dose calculations to members of the public when the discharge stack height is increased to 30 metres for both the average release rates for the operational period (see Table 7) and the bounding release rates. This change in stack height changes the food dose scaling factor and the inhalation and external dose scaling factors used in calculating the radiological dose to the receptor. Using the EA methodology report [77], these scaling factors are, for a 30m stack height, recalculated as 0.18 for the food dose scaling factor and 0.015 for the inhalation and external dose scaling factor. The total dose calculated using the average release rates for the operational period is 0.01 mSv per year, which is equivalent to the design target – by increasing the height of the discharge stack to 30 metres, the doses to members of the public would be reduced by 81%.

7.3 Monitoring

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.

At this stage in the MRWS site selection process, a specific site has not been identified for a GDF. In the absence of a site, the strategy for monitoring can be developed. This is discussed in more detail in Section 3.1.6 of the generic ESC. It is anticipated that a large amount of monitoring will be required prior to construction of a GDF to inform the development of a site-specific DSSC, including components such as the OESA.

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

7.4 Summary

This report focuses on the potential radioactive discharges from a GDF during the operational period that may affect members of the public and non-human biota. Non-radioactive discharges from the wastes are also considered in this report. We consider qualitative supporting arguments for the operational environmental safety of a GDF, and also report a quantitative assessment of operational environmental safety.

The operational period is considered to begin when construction of the GDF is started, and ends with the closure of the facility after the completion of backfilling and final sealing. At this early stage of development we focus on aerial discharges, with liquid and solid discharges being qualitatively considered only (noting that such discharges will be managed on a site-specific basis).
Aerial discharges associated with gases generated in packaged wastes have been estimated. Discharges are dominated by gases generated in low level waste and intermediate level waste packages. 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 canisters. Aerial discharges as a result of gas generation within depleted, natural and low-enriched uranium (DNLEU) are expected to be low and not significant in comparison to releases from low level and intermediate level wastes.

The impact on aerial discharges from an Upper Inventory (which has been defined to account for potential increases in volumes that may arise through, for instance, the introduction of new-build nuclear power stations) is discussed in a qualitative fashion.

The total dose to members of the public for the operational period average gas generation rates is calculated to be 0.052 mSv per year. The majority of this dose arises from radon-222 (0.043 mSv), with minor contributions from carbon-14 bearing gas and tritium. This total dose is below the effective dose limit for members of the public of 1 mSv per year. It is also below the maximum dose to individuals which may result from the discharges from any single site, stated in the Environmental Permitting Regulations 2010, of 0.5 mSv per year, and below the maximum value of dose constraint to members of the public from a new facility of 0.15 mSv per year adopted by RWMD. However, although the dose from carbon-14 and tritium is below the design target of 0.01 mSv per year, the dose from radon-222 exceeds the design target.

Backfilling of the GDF with a cementitious material is expected to take place prior to sealing and closure. Such backfilling has the effect of heating the waste packages as it cures (cement curing is exothermic) and backfilling may make available free water that can be used in e.g. corrosion reactions. Emplacement of a cementitious backfill could therefore contribute to a short-lived enhancement of corrosion rates and hence gas generation rates during the operational period. The rate of gas generation is calculated to peak during the backfilling process, and to be lower at earlier times (which will account for the majority of the operational period).

Studies have also been undertaken using the gas generation rate associated with the backfilling period. The total dose to members of the public for the operational period peak gas generation rates is calculated to be 0.16 mSv per year. The majority of this dose (0.11 mSv per) arises from carbon-14, with contributions from radon-222 and tritium. This total dose is below the effective dose limit for members of the public of 1 mSv per year. It is also below the maximum dose to individuals which may result from the discharges from any single site, stated in the Environmental Permitting Regulations 2010, of 0.5 mSv per year. However, this assessment presents doses to members of the public from radon-222 and carbon-14 that exceed the design target of 0.01 mSv per year, and also the total dose is above the maximum value of dose constraint to members of the public from a new facility of 0.15 mSv per year adopted by RWMD.

The retention of radon-222 (half-life 3.82 days) within a waste package is expressed in terms of an 'emanation coefficient', which corresponds to the fraction of radon-222 that is released from a waste package in comparison with the in-package radon-222 generation rate. Radon-222 doses presented in this report assume an emanation coefficient of 2x10^{-3}. This is considered to be a reasonable value to use at this generic stage, although it is known that significant improvements can be made on this through adoption of multiple barriers in the wasteform and waste package. As part of the RWMD Letter of Compliance disposability assessment process, we will ensure that the emanation coefficient for radon-222 is reduced to a level consistent with the needs of the GDF. Existing practice has established that a significant improvement in emanation coefficient is possible to the extent that the associated dose to members of the public can be reduced by up to three orders of
magnitude. We have confidence therefore that the design target of 0.01 mSv per year can readily be achieved by application of appropriate packaging measures.

Studies have also been undertaken to investigate the effect on calculated radiological dose of varying the discharge stack height – a significant reduction in radiological doses can be achieved by increasing the discharge stack height. Other such potential mitigation measures are also discussed.

A generic illustrative assessment of the potential doses to non-human biota has been undertaken. The total calculated dose rates for all the organisms considered are insignificant in consideration of reviews of the effects of ionising radiation on organisms reported by IAEA and UNSCEAR, which give a broad conclusion that exposure of terrestrial organisms to dose rates of 40 μGy per hour, and exposure of aquatic organisms to dose rates of 400 μGy per hour, would be unlikely to lead to observable effects in these populations. The assessment of dose to non-human biota from off site gaseous radioactive discharge from a GDF is therefore concluded to require no further consideration.

Any actual radiological dose from off site discharges from a 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. We are aware of the need to ensure active management of gas in the operational phase of a GDF, to ensure regulatory dose constraints are met. The issues and uncertainties associated with estimates of gas generation are set out in the gas status report. Mitigation measures could be taken to reduce the potential for gaseous discharges from a GDF that would, consequently, also reduce any off site doses.

We will continue to advance our understanding of discharges (both radioactive and non-radioactive) from a facility that may affect members of the public and the environment including non-human biota, by commissioning research and development to address areas of uncertainty in a manner that takes account of their potential significance, and the status and progress of our programme to implement a GDF.

We will update this report, in line with updates to the DSSC, as part of each major stage of a GDF development programme. Over time, the design options under consideration and the choices we have to make will change from an emphasis on strategy to one on implementation. This approach is consistent with a staged development and approval process.
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. We will continue to improve our understanding and confirm our assumptions about gas generation from all of the packaged waste and materials in the Derived Inventory. Once a site for a GDF is selected, a site-specific OESA will be produced that is based on the then-current UK radioactive waste inventory.

8.2 Future research

To date, much of the gas related work undertaken by NDA RWMD has focused on the post-closure phase for a GDF. We have recognised the need to determine in more detail the shorter term dynamics of gas evolution from the waste packages, noting especially the effect of assumptions regarding the presence of water, temperature, the maintenance of aerobic conditions and the impact of progressive backfilling of completed disposal areas during the operational period on gas generation rates. Some of the work in progress or planned is noted as follows:

- Work is currently being undertaken to study gas generation in low and intermediate level wastes as the GDF transitions to the post-closure state. This study aims to further develop the understanding of gas generation as currently published [33, 71] particularly during GDF operations (including backfilling and closure) and the early post-closure period. An appropriate methodology for assessment of gas generation during this transition period will be developed. This work will also utilise the Derived Inventory dataset as specified in [7] to update the quantitative assessment reported in this generic OESA.

- Additional work is also in progress to address the effects that delayed closure of a GDF could have on its evolution, and what the related impacts could be for the OESA and a PCSA (noting that delayed closure would create a longer operational period for a GDF). Delayed closure could also result in alternative GDF operation, GDF engineering and design features and a range of appropriate GDF environmental conditions and management choices affecting the emplaced wastes prior to closure. The effects of these alternative scenarios on issues covered in the OESA will be assessed as part of this work.

- The assumptions discussed in this report relating to gas generation and discharge rates will be considered further, including assumptions relating to waste packaging. Work is currently being undertaken to study waste packages that are in storage, monitoring the types and amounts of gases that are being produced. Such studies will help with assessments of gas generation and release rates from the wastes emplaced in a GDF.

- Work has been undertaken to develop the methodology for assessing non-radioactive discharges from a GDF [91]. This work will need to be drawn upon and developed further for future assessments.

- We will continue to investigate the release rates of gaseous carbon-14 from wastes and the extent to which radon-222 decays as it migrates within and from waste packages, and as it is transported through the engineered systems in a GDF during the operational period.
• There remains uncertainty relating to the emanation coefficient that should be applied in quantified assessments. It is considered that values lower than that utilised in this assessment ($2 \times 10^{-3}$) can be justified. It is expected that assessments undertaken through the Letter of Compliance disposability assessment process will enable us to consider what can be achieved in practice for specific waste streams. Further work is also required to better understand hold up of radon within the engineered barriers of the GDF and hence the rate of off-site discharge.

• As discussed in Section 7.1.2, the current gas generation calculations assume a temperature increase of 80°C on backfilling of the vaults. Following a recent study [88], which infers that 40°C may be a more likely temperature within the vaults, any future gas generation calculations should be able to include this change in temperature.

8.3 Further work on non-radioactive discharges

During a GDF operational period, there would 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, would generate waste lubricants and solids, and combustion engine vehicles would generate exhaust fumes. These would be assessed to determine the local consequences and would be subject to optimisation with respect to the ‘carbon footprint’ of the facility. Such considerations form part of both the Operational Safety Case [27] and environmental assessments.

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 are based on the illustrative disposal concept examples for the design of a GDF described in [8]. 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 we move through the MRWS process towards site selection, these assumptions will become site specific and our assessments can then be refined accordingly.

Note that, pursuant to RWMD working practices, any issues identified with the release of gases from wastes in the development of the safety case (including the OESA component) can be discussed with the designers of a GDF and the waste packagers, to ensure that additional measures as required to mitigate the issues can be put in place at a stage in the waste packaging process prior to dispatching the waste to a GDF. Proposed changes that may have safety or environmental impacts are categorised for significance and where appropriate these changes are considered by our Nuclear Safety and Environment Committee.

The assessment of the impact of gaseous discharges off site 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 ALARP. 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 ALARP, they do not compromise the off site doses, and vice versa. At this stage in the process it is too early to make any judgements on this issue.
Optioneering work is being undertaken to consider alternative options for implementation; radiological protection is a consideration in these optioneering studies. Detailed optimisation studies would be carried out at the appropriate time (when more details are known about the actual systems and processes once a site is identified) to consider the need to reduce off site doses and any consequent impact on worker (on-site) doses. As described above the results of options studies inform the development of the design and are considered through our management arrangements.

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 republished.

8.6 Statement of confidence

In the generic OESA we have explained our assessment approach and described our methodology for:

- assessing the non-radioactive discharges from a GDF;
- assessing the discharges from a GDF both for surface facilities and those underground;
- calculating doses to members of the public located off site from a GDF; and
- calculating doses to non-human biota located off site from a GDF.

We have illustrated parts of our modelling approach by example at this generic stage. We have confidence in our ability to model the doses to members of the public and the environment from a GDF and have the necessary experience to assimilate information about a site, and incorporate this into our modelling capability, when it becomes available.

The generic ESC, and underpinning reports such as this generic OESA and the PCSA, illustrate how geological disposal could be implemented safely in different geological environments for the UK inventory of higher activity radioactive wastes. Our confidence that we can develop a GDF for the UK’s inventory of higher activity radioactive wastes is built on our understanding of how to build and operate a GDF, and how in the longer term multiple barriers can work together to provide long-term safety for a wide range of geological environments. We therefore have confidence that once we have a preferred site and disposal concept, we will be able to develop an optimised design that meets all environmental safety requirements. The safety and environmental assessments we have undertaken are sufficient to underpin future disposability assessments and decisions taken as part of issuing a Letter of Compliance to waste producers. Overall, our knowledge base is sufficient to progress from the generic stage to studies of candidate sites.
Glossary

**activity**
The quantity \( A \) for an amount of *radionuclide* in a given energy state at a given time, defined as:

\[
A(t) = \frac{dN}{dt}
\]

where \( dN \) is the expectation value of the number of spontaneous nuclear transformations from the given energy state in the time interval \( dt \).

The rate at which nuclear transformations occur in a *radioactive material*. The equation is sometimes given as:

\[
A(t) = -\frac{dN}{dt}
\]

where \( N \) is the number of nuclei of the *radionuclide*, and hence the rate of change of \( N \) with time is negative. Numerically, the two forms are identical.

The SI unit of activity is the reciprocal second (s\(^{-1}\)), termed the *Becquerel* (Bq).

Formerly expressed in curies (Ci); activity values may be given in Ci (with the equivalent in Bq in parentheses) if they are being quoted from a reference that uses Ci as the unit.

**advanced gas-cooled reactor (AGR)**
The reactor type used in the UK’s second generation nuclear power plants.

**backfill**
A material used to fill voids in a *geological disposal facility*. Three types of backfill are recognised:

- *local backfill*, which is emplaced to fill the free space between and around *waste* packages;
- *peripheral backfill*, which is emplaced in *disposal* modules between waste and local backfill, and the near-field rock or access ways; and
- *mass backfill*, which is the bulk material used to backfill the excavated volume apart from the *disposal* areas.

**backfilling**
The refilling of the excavated portions of a *disposal facility* after *emplacement* of the waste.

**barrier**
A physical obstruction that prevents or inhibits the movement of people, *radionuclides* or some other phenomenon—(e.g. fire), or provides shielding against radiation.

See also *containment*.

**multiple barriers.** Two or more natural or engineered barriers used to isolate *radioactive* waste in, and prevent migration of *radionuclides* from a *repository*.

The term ‘chemical barrier’ is sometimes used in the context of waste disposal to describe the chemical effect of a material that enhances the extent to which *radionuclides* react chemically with the material or with the host rock, thus inhibiting the migration of the *radionuclides*. As defined above, this is not strictly a barrier (unless the material also constitutes a physical barrier), but the effect may be equivalent to that of a barrier, and it may therefore be convenient to regard it as such.
baseline inventory
An estimate of the higher activity radioactive waste and other materials that could, possibly, come to be regarded as wastes that might need to be managed in the future through geological disposal drawn from the UK Radioactive Waste Inventory.

becquerel (Bq)
The SI unit of activity, equal to one transformation per second.
Supersedes the non-SI unit curie (Ci). $1 \text{ Bq} = 27 \text{ pCi} \left(2.7 \times 10^{-11} \text{ Ci}\right)$ approximately.
$1 \text{ Ci} = 3.7 \times 10^{10} \text{ Bq}.$

bentonite
A highly sorbing clay material used as a backfill in certain disposal concepts.

biosphere
That part of the environment normally inhabited by living organisms.

British Geological Survey (BGS)
The BGS provides expert services and impartial advice in all areas of geoscience.

buffer
Any substance placed around a waste package in a repository to serve as a barrier to restrict the access of groundwater to the waste package and to reduce by sorption and precipitation the rate of eventual migration of radionuclides from the waste.

canister
A term used in specific concepts to describe the empty vessel into which a wasteform is placed.

closure
1. Administrative and technical actions directed at a repository at the end of its operating lifetime e.g. covering of the disposed waste (for a near surface repository) or backfilling and/or sealing (for a geological repository and the passages leading to it) and the termination and completion of activities in any associated structures.

For other facilities, the term decommissioning is used.

2. The completion of all operations at some time after the emplacement of spent fuel or radioactive waste in a disposal facility. This includes the final engineering or other work required to bring the facility to a condition that will be safe in the long term.

Committee on Radioactive Waste Management (CoRWM)
CoRWM was set up in 2003 to provide independent advice to Government on the long-term management of the UK’s solid higher activity radioactive waste. In October 2007, CoRWM was reconstituted with revised Terms of Reference and new membership. The Committee will provide independent scrutiny and advice to UK Government and devolved administration Ministers on the long-term radioactive waste management programme, including storage and disposal. Further information available at http://www.corwm.org.uk

conditioning
Treatment of a radioactive waste material to create, or assist in the creation of, a wasteform that has passive safety.

conditioned waste volume
The conditioned waste volume is the volume of the wasteform (waste plus immobilising medium) within the container.
**containment**

Methods or physical structures designed to prevent or control the release and the dispersion of radioactive substances.

Although related to confinement, containment is normally used to refer to methods or structures that perform a confinement function, namely preventing or controlling the release of radioactive substances and their dispersion in the environment.

**criticality**

The state of a nuclear chain reacting medium when the chain reaction is just self-sustaining (or critical), i.e. when the reactivity is zero.

Often used, slightly more loosely, to refer to states in which the reactivity is greater than zero.

**decision-making body**

Local Government will have decision-making authority for their host community in respect of continued participation at key stages in the siting process, or exercising a Right of Withdrawal; the local acceptability of proposals for Community Benefits Packages; the local acceptability of sites proposed for surface investigations; and whether potential retrievability of wastes has been adequately considered. There are different local authority structures in different parts of the UK. For example, in England local authorities include district councils, county councils, metropolitan district councils and London Boroughs whereas in Wales, local authorities are unitary. Such a body is termed a ‘Decision-Making Body’.

**decommissioning**

1. Administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility (except for a repository or for certain nuclear facilities used for the disposal of residues from the mining and processing of radioactive material, which are ‘closed’ and not ‘decommissioned’). Decommissioning typically includes dismantling of the facility (or part thereof), but in the IAEA’s usage this need not be the case. A facility could, for example, be decommissioned without dismantling and the existing structures subsequently put to another use (after decontamination).

The use of the term decommissioning implies that no further use of the facility (or part thereof) for its existing purpose is foreseen.

Decommissioning actions are taken at the end of the operating lifetime of a facility to retire it from service with due regard for the health and safety of workers and members of the public and the protection of the environment. Subject to national legal and regulatory requirements, a facility (or its remaining parts) may also be considered decommissioned if it is incorporated into a new or existing facility, or even if the site on which it is located is still under regulatory control or institutional control.

The actions will need to be such as to ensure the long term protection of the public and the environment, and typically include reducing the levels of residual radionuclides in the materials and on the site of the facility so that the materials can be safely recycled, reused or disposed of as exempt waste or as radioactive waste and the site can be released for unrestricted use or otherwise reused.

For a repository, the corresponding term is closure.

2. All steps leading to the release of a nuclear facility, other than a disposal facility, from regulatory control. These steps include the processes of decontamination and dismantling.

**depleted uranium (DU)**

Uranium in which the proportion of U-235 is less than ~0.7%.
deposition hole
Vertical hole within a deposition tunnel in which a HLW, SF, Pu or HEU canister is placed for disposal.

disposability
The degree to which conditioned waste meets the requirements for final disposal.

disposal
Emplacement of waste in an appropriate facility without the intention of retrieval.
In some States, the term disposal is used to include discharges of effluents to the environment.
In some States, the term disposal is used administratively in such a way as to include, for example, incineration of waste or the transfer of waste between operators. In IAEA publications, disposal should be used only in accordance with the more restrictive definition given above. In many cases, the only element of this definition that is important is the distinction between disposal (with no intent to retrieve) and storage (with intent to retrieve).
In such cases, a definition is not necessary; the distinction can be made in the form of a footnote at the first use of the term disposal or storage (e.g. “The use of the term disposal indicates that there is no intention to retrieve the waste. If retrieval of the waste at any time in the future is intended, the term storage is used.”).
The term disposal implies that retrieval is not intended; it does not mean that retrieval is not possible.
For storage in a combined storage and disposal facility, for which a decision may be made at the time of its closure whether to remove the waste stored during the operation of the storage facility or to dispose of it by encasing it in concrete, the question of intention of retrieval may be left open until the time of closure of the facility.

disposal canister
A term used to describe the assembly of certain waste types (e.g. HLW, spent fuel, plutonium, HEU) within a metal container, as prepared for disposal.

disposal facility (for solid radioactive waste)
An engineered facility for the disposal of solid radioactive wastes.

disposal module
Collective term for a group of deposition tunnels. The number of deposition tunnels in a module could vary for different rock types.

disposal tunnel
Tunnel in which HLW, spent fuel, Pu and HEU canisters are placed for disposal.

disposal unit
A waste package, or group of waste packages, which is handled as a single unit for the purposes of transport and/or disposal.

disposal vault
Underground opening where ILW or LLW waste packages are emplaced.

dose
1. A measure of the energy deposited by radiation in a target.
2. Absorbed dose, committed equivalent dose, committed effective dose, effective dose, equivalent dose or organ dose, as indicated by the context.
**dose equivalent**
Dose equivalent takes into account not only the energy deposited in body tissue by radioactivity (either external or internal) but also the different biological effectiveness of the various forms of radiation in causing harm to body tissues. The SI unit of dose equivalent is the sievert (Sv).

The product of the absorbed dose at a point in the tissue or organ and the appropriate quality factor for the type of radiation giving rise to the dose.

A measure of the dose to a tissue or organ designed to reflect the amount of harm caused.

A quantity used by the International Commission on Radiation Units and Measurements in defining the operational quantities ambient dose equivalent, directional dose equivalent and personal dose equivalent (see dose equivalent quantities). The quantity dose equivalent has been superseded for radiation protection purposes by equivalent dose.

**dose rate**
The effective dose equivalent per unit time. Typical units of effective dose are sievert/hour (Sv h⁻¹), millisieverts/hour (mSv h⁻¹) and sievert/year (Sv y⁻¹).

**drift**
A sloping underground tunnel.

**effective dose equivalent**
A measure of dose designed to reflect the risk associated with the dose, calculated as the weighted sum of the dose equivalents in the different tissues of the body.

Superseded by effective dose.

**emplacement (of waste in a disposal facility)**
The placement of a waste package in a designated location for disposal, with no intent to reposition or retrieve it subsequently.

**engineered barrier system**
The combination of the man-made engineered components of a disposal facility, including the waste packages/disposal canisters, buffer, backfills and seals.

**enrichment (uranium)**
The proportion (usually expressed as a % of the total mass) of U-235 in uranium.

**Environment Agency**
The environmental regulator for England and Wales. The Agency’s role is the enforcement of specified laws and regulations aimed at protecting the environment, in the context of sustainable development, predominantly by authorising and controlling radioactive discharges and waste disposal to air, water (surface water, groundwater) and land. The Environment Agency also regulates nuclear sites under the Environmental Permitting Regulations and issues consents for non-radioactive discharges.

**environmental safety case**
The collection of arguments, provided by the developer or operator of a disposal facility, that seeks to demonstrate that the required standard of environmental safety is achieved.

**European Commission**
The executive body of the European Union. Its primary roles are to propose and implement legislation, and to act as guardian of the treaties which provide the legal basis for the European Union.
**evaporite**

The generic term for a geological environment created by the evaporation of water from a salt bearing solution to form a solid structure.

**excavation disturbed zone**

A region of the geosphere surrounding the engineered barrier system which has been affected (i.e. physically damaged) as a result of construction of a GDF.

**external irradiation**

The exposure of a body to radiation arising from sources located outside of the body.

**far field**

The geosphere outside a repository, comprising the surrounding geological strata, at a distance from the repository such that, for modelling purposes, the repository may be considered a single entity and the effects of individual waste packages are not distinguished.

For practical purposes, this is often interpreted simply as the geosphere beyond the near field.

**fissile material**

Uranium-233, uranium-235, plutonium-239, plutonium-241 or any combination of these radionuclides. Excepted from this definition are:

(a) Natural uranium or depleted uranium which is unirradiated;

(b) Natural uranium or depleted uranium which has been irradiated in thermal reactors only.

As with radioactive material, this is not a scientific definition, but one designed to serve a specific regulatory purpose.

**geological disposal**

A long term management option involving the emplacement of radioactive waste in an engineered underground geological disposal facility or repository, where the geology (rock structure) provides a barrier against the escape of radioactivity and there is no intention to retrieve the waste once the facility is closed.

**geological disposal facility**

An engineered underground facility for the disposal of solid radioactive wastes.

**geosphere**

Those parts of the lithosphere not considered to be part of the biosphere.

In safety assessments usually used to distinguish the subsoil and rock (below the depth affected by normal human activities, in particular agriculture) from the soil that is part of the biosphere.

**groundwater**

Water located beneath the earth’s surface in rock pores and fractures.

**half-life**

1. For a radionuclide, the time required for the activity to decrease, by a radioactive decay process, by half.

Where it is necessary to distinguish this from other half-lives (see (2)), the term radioactive half-life should be used.
The half-life $T_i$ for process $i$ is related to the decay constant $\lambda$, by the expression:

$$T_i = \ln2/\lambda.$$  

2. The time taken for the quantity of a specified material (e.g. a radionuclide) in a specified place to decrease by half as a result of any specified process or processes that follow similar exponential patterns to radioactive decay.

**biological half-life.** The time taken for the quantity of a material in a specified tissue, organ or region of the body (or any other specified biota) to halve as a result of biological processes.

**radioactive half-life.** For a radionuclide, the time required for the activity to decrease, by a radioactive decay process, by half.

The term **physical half-life** is also used for this concept.

**hazardous materials**

Materials that can endanger human health if improperly handled. As defined by the Control of Substances Hazardous to Health Regulations, 2002.

**higher activity radioactive waste**

Generally used to include the following categories of radioactive waste: high level waste, intermediate level waste, low level waste with a concentration of specific radionuclides.

**higher strength rock**

Typically crystalline igneous and metamorphic rocks or geologically older sedimentary rocks where any fluid movement is predominantly through discontinuities.

**high enriched uranium (HEU)**

See uranium.

**high level waste (HLW)**

Radioactive wastes in which the temperature may rise significantly as a result of their radioactivity, so this factor has to be taken into account in the design of storage or disposal facilities.

**host community**

The community in which any facility will be built. The host community will be a small geographically defined area, and include the population of that area and the owners of the land. For example, it could be a town or village.

**immobilisation**

Conversion of waste into a waste form by solidification, embedding or encapsulation. Immobilisation reduces the potential for migration or dispersion of radionuclides during handling, transport, storage and/or disposal.

**industrial package**

See packaging.

**intermediate level waste (ILW)**

Radioactive wastes exceeding the upper activity boundaries for LLW but which do not need heat to be taken into account in the design of storage or disposal facilities.

**Internal irradiation**

The exposure of a body to radiation arising from sources located inside the body.
**International Atomic Energy Agency (IAEA)**

The IAEA is the world’s centre of cooperation in the nuclear field. It was set up as the world’s "Atoms for Peace" organisation in 1957 within the United Nations family. The Agency works with its Member States and multiple partners worldwide to promote safe, secure and peaceful nuclear technologies.

**International Commission on Radiological Protection (ICRP)**

An international advisory body founded in 1928 providing recommendations and guidance on radiation protection. ICRP recommendations normally form the basis for EU and UK radiation protection standards.

**isolation**

A feature of a geological disposal facility component that contributes to safety. The depth of disposal and the characteristics of the host rock isolate the waste from the biosphere, thus reducing the likelihood of inadvertent and unauthorised human interference.

The depth of disposal also provides isolation of the disposal facility from the impacts of climatic and other natural environmental events, and shielding from direct radiation.

**legacy waste**

Radiactive waste which already exists or whose arising is committed in future by the operation of an existing nuclear power plant.

**letter of compliance**

A document, prepared by RWMD, that indicates to a waste packager that a proposed waste package is compliant with the relevant packaging criteria and disposal safety assessments, and is therefore deemed to be compatible with disposal in a GDF.

**low enriched uranium**

Uranium in which the proportion of U-235 is greater than ~0.7% but less than ~20%.

**lower strength rock**

Typically geologically ‘young’ sedimentary rocks where any fluid movement is predominantly through the rock matrix.

**low level waste**

LLW is defined as “radioactive waste having a radioactive content not exceeding 4 gigabequerels per tonne (GBq/te) of alpha or 12 GBq/te of beta/gamma activity”.

**managing radioactive waste safely**

A phrase covering the whole process of public consultation, work by CoRWM, and subsequent actions by Government, to identify and implement the option, or combination of options, for the long term management of the UK’s higher activity radioactive waste.

**natural uranium**

See uranium.

**near field**

The excavated area of a GDF which is near or in contact with the waste packages, backfill or sealing materials and those parts of the geosphere whose characteristics have been or could have been altered by the presence of the GDF or its contents.

**new build**

New build of a nuclear power station.
**Nirex (United Kingdom Nirex Limited)**

An organisation previously owned jointly by Department for Environment, Food and Rural Affairs and the Department of Trade and Industry. Its objectives were, in support of Government policy, to develop and advise on safe, environmentally sound and publicly acceptable options for the long-term management of radioactive materials in the United Kingdom. The Government’s response to Committee on Radioactive Waste Management in October 2006 initiated the incorporation of Nirex functions into the NDA, a process which was completed in March 2007.

**Nuclear Decommissioning Authority (NDA)**

The NDA is the implementing organisation, responsible for planning and delivering the GDF. The NDA was set up on 1 April 2005, under the Energy Act 2004. It is a non-departmental public body with designated responsibility for managing the liabilities at specific sites. These sites are operated under contract by site licensee companies (initially British Nuclear Group Sellafield Limited, Magnox Electric Limited, Springfields Fuels Limited and UK Atomic Energy Authority). The NDA has a statutory requirement under the Energy Act 2004, to publish and consult on its Strategy and Annual Plans, which have to be agreed by the Secretary of State (currently the Secretary of State for Trade and Industry) and Scottish Ministers.

**nuclear waste**

A general term for the radioactive waste produced by those industries involved with nuclear energy and nuclear weapons’ production.

**overpack**

A secondary or additional outer container used for the handling, transport, storage or disposal of waste packages.

**operational period (of a disposal facility)**

The period during which a disposal facility is used for its intended purpose, up until closure.

**packaging**

The assembly of components necessary to enclose the radioactive contents completely. It may, in particular, consist of one or more receptacles, absorbent materials, spacing structures, radiation shielding and service equipment for filling, emptying, venting and pressure relief; devices for cooling, absorbing mechanical shocks, handling and tie-down, and thermal insulation; and service devices integral to the package. The packaging may be a box, drum or similar receptacle, or may also be a freight container, tank or intermediate bulk container.

**packaged waste volume**

The packaged waste volume is the displacement volume of a container used to package a wasteform.

**passive safety**

The need to provide and maintain a safety function by minimising the need for active safety systems, monitoring or prompt human intervention. Requires radioactive wastes to be immobilised and packaged in a form that is physically and chemically stable. The package should be stored in a manner that is resistant to degradation and hazards, and which minimises the need for control and safety systems, maintenance, monitoring and human intervention.

**performance assessment**

Assessment of the performance of a system or subsystem and its implications for protection and safety at an authorised facility.
This differs from safety assessment in that it can be applied to parts of an authorised facility (and its environment), and does not necessarily require the assessment of radiological impacts.

**period of authorisation**
The period of time while *disposals* are taking place and any period afterwards while the site is under *active institutional control*.

**permeability**
A measure of the rate at which a gas or a liquid moves under a pressure gradient through a porous material.

**plutonium (Pu)**
A radioactive element occurring in very small quantities in *uranium* ores but mainly produced artificially, including for use in nuclear fuel, by neutron bombardment of *uranium*.

**porewater**
*Groundwater* held within a space or pore in rock.

**porosity**
The ratio of the aggregate volume of interstices or porous media to total volume of a body.

**pressurised water reactor (PWR)**
Reactor type using ordinary water under high pressure as coolant and neutron moderator. PWRs are widely used throughout the world for electricity generation. The Sizewell B reactor in Suffolk is of this design.

**radioactive decay**
The process by which radioactive material loses activity, e.g. *alpha* activity naturally. The rate at which atoms disintegrate is measured in *becquerels*.

**radioactive material**
Material designated in national law or by a regulatory body as being subject to regulatory control because of its radioactivity.

**radioactive waste**
Any material contaminated by or incorporating radioactivity above certain thresholds defined in legislation, and for which no further use is envisaged, is known as radioactive waste.

**Radioactive Waste Management Directorate (RWMD)**
The NDA Directorate established to design and build an effective delivery organisation to implement a safe, sustainable, publicly acceptable geological *disposal* programme. It is envisaged that this directorate will become a wholly owned subsidiary company of the NDA. Ultimately, it will evolve under the NDA into the organisation responsible for the delivery of the GDF. Ownership of this organisation can then be opened up to competition, in due course, in line with other NDA sites.

**radioactivity**
Atoms undergoing spontaneous random disintegration, usually accompanied by the emission of radiation.

**radiolysis**
The degradation of a chemical as a result of exposure to radiation.
**radionuclide**
A radioactive form of an element, for example carbon-14 or caesium-137.

**repository**
A nuclear facility where waste is *emplaced* for *disposal*.

**resaturation**
The process of returning the concentration of water in a system to its maximum holding capacity.

**retardation**
A feature of a *geological disposal facility* component that contributes to safety. The engineered *barriers* and host rock provide retardation of *radionuclides* through physical and chemical processes that reduce the concentration of contaminants or their rate of release from the barrier. Retardation processes may result in effective *containment* of the *radionuclides* if they would only be released through the barriers after the time at which they and their daughters have decayed to negligible levels.

**right of withdrawal**
An important part of the voluntarism approach intended to contribute to the development and maintenance of community confidence. Up until a late stage, when underground operations and construction are due to begin, if a community wished to withdraw then its involvement in the process would stop.

**safety case**
A collection of arguments and evidence in support of the safety of a facility or activity.

This will normally include the findings of a safety assessment and a statement of confidence in these findings.

For a *repository*, the safety case may relate to a given stage of development. In such cases, the safety case should acknowledge the existence of any unresolved issues and should provide guidance for work to resolve these issues in future development stages.

**Safety function**
A specific purpose that must be accomplished for safety.

In early IAEA publications, the terms ‘basic safety function’ and ‘fundamental safety function’ were also used.

**shaft**
A vertical or near-vertical tunnel extending underground from the surface.

**shielded waste package**
A shielded waste package is one that either has in-built shielding or contains low activity materials, and thus may be handled by conventional techniques. In most cases, shielded waste packages are also designed to qualify as *transport packages* in their own right.

**shielding**
Shielding is the protective use of materials to reduce the *dose rate* outside of the shielding material. The amount of shielding required to ensure that the *dose rate* is as low as reasonably practicable (ALARP) will therefore depend on the type of radiation, the activity of the source, and on the dose rate that is acceptable outside the shielding material.
site characterisation

Detailed surface and subsurface investigations and activities at a site to determine the radiological conditions at the site or to evaluate candidate disposal sites to obtain information to determine the suitability of the site for a repository and to evaluate the long term performance of a repository at the site.

Site characterisation is a stage in the siting of a repository; it follows area survey and precedes site confirmation.

Site characterisation may also refer to the siting process for any other authorised facility.

Spent fuel (SF)

1. Nuclear fuel removed from a reactor following irradiation that is no longer usable in its present form because of depletion of fissile material, poison buildup or radiation damage.

2. Nuclear fuel that has been irradiated in and permanently removed from a reactor core.

The adjective ‘spent’ suggests that spent fuel cannot be used as fuel in its present form (e.g. as in spent source). In practice, however (as in (2) above), spent fuel is commonly used to refer to fuel which has been used as fuel but will no longer be used, whether or not it could be used (and which might more accurately be termed ‘disused fuel’).

standard waste package

A waste package designed to be consistent with a specific waste package specification.

storage

The emplacement of waste in a suitable facility with the intent to retrieve it at a later date.

transport container

A reusable container into which waste packages are placed for transport, the whole then qualifying as a transport package under the Transport Regulations.

transport package

As defined in the IAEA Transport Regulations: ‘the complete assembly of the radioactive material and its outer packaging, as presented for transport.’

Transport Regulations

The IAEA Regulations for the Safe Transport of Radioactive Material and/or those regulations as transposed into an EU Directive, and in turn into regulations that apply within the UK. The generic term ‘Transport Regulations’ can refer to any or all of these, since the essential wording is identical in all cases.

UK Radioactive Waste Inventory (UKRWI)

A compilation of data on UK radioactive waste holdings, produced about every three years. The latest version, for a holding date of 1 April 2007, was published in June 2008. It is produced by the Department for Environment, Food and Rural Affairs and the NDA. It is the latest public record of information on the sources, quantities and properties of LLW, ILW and HLW in the UK. It comprises of a number of reports and additional detailed information on the quantities and properties of radioactive wastes in the UK that existed at 1 April 2007 and those that were projected to arise after that date.

uncertainty

A state of limited knowledge that precludes an exact or complete description of past, present or future.
**unshielded waste package**

An unshielded *waste package* is one that, owing either to radiation levels or *containment* requirements, requires remote handling and must be transported in a reusable *transport container* (the container and contents then forming a Type B transport package).

**uranium (U)**

A heavy, naturally occurring and weakly radioactive element, commercially extracted from uranium ores. By nuclear fission (the nucleus splitting into two or more nuclei and releasing energy) it is used as a fuel in nuclear reactors to generate heat.

Uranium is often categorised by way of the proportion of the *radionuclide* uranium U-235 it contains (see *natural uranium, depleted uranium, low enriched uranium and highly enriched uranium*).

*depleted uranium*. Uranium containing a lesser mass percentage of uranium-235 than in *natural uranium*.

*enriched uranium*. Uranium containing a greater mass percentage of uranium-235 than 0.72%.

*high enriched uranium (HEU)*. Uranium containing 20% or more of the isotope uranium-235. *HEU* is considered a special *fissile material* and a direct use material.

*low enriched uranium (LEU)*. *Enriched uranium* containing less than 20% of the isotope uranium-235. *LEU* is considered a special *fissile material* and an indirect use material.

*natural uranium*. Uranium (which may be chemically separated) containing the naturally occurring distribution of uranium isotopes (approximately 99.28% uranium-238 and 0.72% uranium-235 by mass).

In all cases, a very small mass percentage of uranium-234 is present.

The naturally occurring distribution of uranium isotopes including uranium-234 (approximately 99.285% uranium-238, 0.710% uranium-235 and 0.005% uranium-234 by mass) corresponds to approximately 48.9% uranium-234, 2.2% uranium-235 and 48.9% uranium-238 by activity.

**waste container**

The vessel into which the *waste form* is placed for handling, transport, *storage* and/or eventual *disposal*; also the outer barrier protecting the waste from external intrusions. The waste container is a component of the *waste package*. For example, molten *high level waste* glass would be poured into a specially designed container (*canister*), where it would cool and solidify.

Note that the term waste *canister* is considered to be a specific term for a container for *spent fuel* or vitrified *high level waste*.

**wasteform**

The waste in the physical and chemical form in which it will be disposed of, including any conditioning media and container furniture (i.e. in-drum mixing devices, dewatering tubes etc) but not including the waste container itself or any added inactive capping material.

**waste form**

The physical and chemical characteristics of an unconditioned waste.

**waste hierarchy**

A hierarchical approach to minimise the amounts of waste requiring *disposal*. The hierarchy consists of non-creation where practicable; minimisation of arisings where the creation of waste is unavoidable; recycling and reuse; and, only then, disposal.
**waste package**
The complete assembly of waste container and wasteform as prepared in accordance with requirements for handling, transport, storage and disposal.

**waste packager**
An organisation responsible for the packaging of radioactive waste in a form suitable for transport and disposal.

**waste producer**
An organisation responsible for the creation and/or storage of radioactive waste in an unconditioned form.

**wider local interests**
Outside the host community, there are likely to be other communities that have an interest in the development of a facility in the host community, and there needs to be a mechanism that allows them to become involved in the process. Such a community might be the next village, a neighbouring district or a community on the local transport routes to the host community. Such communities are termed ‘wider local interests’.

**WVP canister**
A stainless steel vessel containing vitrified HLW, as manufactured in the Waste Vitrification Plant (WVP) at Sellafield.
References


48. Radioactive Waste Management Directorate Radiological Protection Policy Manual RWM02 Revision 1 August 2010


59. HSE, Safety Assessment Principles (SAPs) for Nuclear Facilities, 2006 Edition, Revision 1

60. Health and Safety at Work etc Act 1974.

61. Nuclear Installations Act 1965 (as amended) (NIA65).


80. [http://project.facilia.se/erica/](http://project.facilia.se/erica/)


89. A Fenton and A W Harris, *Derivation of Radium-226 Inventory Limits for Packaged Wastes*, AEA Technology Report, RWMD (99)P70.

