Nuclear Decommissioning Authority

NDA Report no. NDA/RWMD/021

Geological Disposal

Generic Environmental Safety Case main report

December 2010

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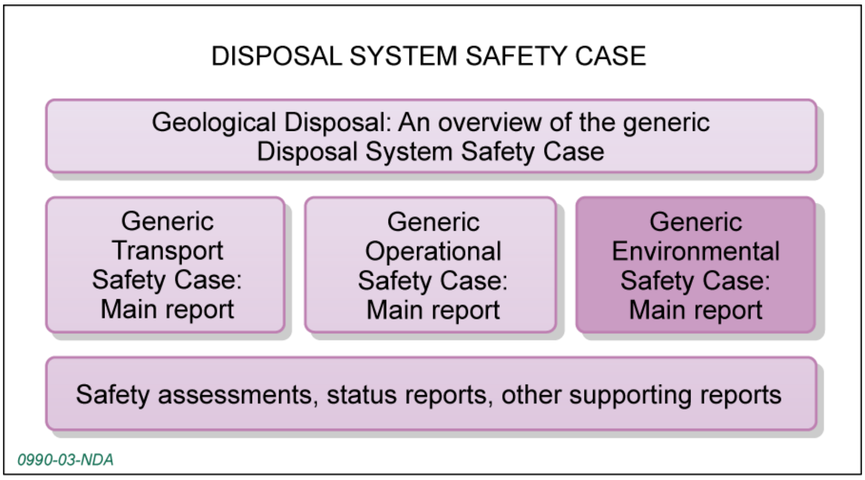
Preface

The Nuclear Decommissioning Authority (NDA) has been charged with implementing the UK Government's policy for the long-term management of higher activity radioactive waste by planning, building and operating a geological disposal facility (GDF). Within the NDA, we - the Radioactive Waste Management Directorate (RWMD) - are tasked with development of a GDF.

The UK has accumulated a legacy of radioactive waste from electricity generation and associated fuel cycle activities, defence activities and other industrial, medical, agricultural and research activities. Radioactive wastes continue to be produced from these activities. Some of these wastes will remain hazardous for thousands of years. The development of new nuclear power plants in the UK would lead to the generation of wastes similar to those already in existence.

As part of the Managing Radioactive Waste Safely (MRWS) programme, the UK Government undertook a wide-ranging consultation on the best means of dealing with higher activity radioactive wastes. This led, in October 2006, to the UK Government [1] deciding that geological disposal, preceded by safe and secure interim storage, is the way forward for the long-term management of these wastes.

The UK Government then published a White Paper in June 2008 that explained the framework for implementing a national GDF. The White Paper set out the stages in a site selection process that would lead to the identification of sites for desk-based studies, followed by surface investigations at candidate sites, and leading ultimately to the identification of a preferred site. The White Paper also invited communities to express an interest in opening up without commitment discussions with the UK Government on the possibility of hosting a GDF at some point in the future. Our first significant involvement in the UK Government's site selection process will be the conduct of desk-based studies in participating areas.



An important part of our preparatory work is for us to set out an approach for assessing the safety of the disposal system. We are setting out the approach we propose in an integrated set of documents under the collective title 'The Disposal System Safety Case' (DSSC). The DSSC presents methods, evidence and arguments concerning the safety of the transport of waste to a GDF, and the construction, operation and long-term safety of a GDF for UK higher activity radioactive wastes.

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[1] Throughout this document, the term UK Government includes all relevant departments and devolved administrations, but excludes the Scottish Government, which does not support geological disposal – the Scottish Government advocates interim near-surface, near-site storage for higher activity radioactive wastes.

This document is the generic Environmental Safety Case main report (ESC main report). The ESC considers the environmental safety of a GDF during the operational period and after closure of the facility. The ESC will be of particular interest to the environmental regulators – the Environment Agency – as it responds to the published regulatory guidance. Two companion reports address (i) the safety of transport of waste to a disposal facility (the generic Transport Safety Case main report) and (ii) the safety of the construction and operation of a GDF (the generic Operational Safety Case main report), and will be of particular interest to the regulators responsible for transport safety (the Department for Transport) and operational safety (the Health and Safety Executive) respectively.

As part of the DSSC, we are also publishing an overview report for a wider readership, and a series of safety assessment reports and research status reports on key topics relevant to the safety of a GDF, as well as other supporting reports.

The ESC – and the overall DSSC – will be developed in stages, alongside the development of a GDF itself, over a period lasting several decades. We call this document a “generic ESC” because no specific sites for a GDF have yet been identified.

This generic ESC explains in principle why we have confidence in the environmental safety of a GDF, and our approach to developing the necessary safety case to demonstrate that confidence. A range of possible geological environments and illustrative geological disposal concept examples is described, and we have included illustrative safety assessment calculations based on work documented in the underpinning safety assessment reports.

Progressive updating of the DSSC provides a management tool for use in our ongoing development of research, site characterisation and engineering design programmes that respond to the evolving information needs and outputs of the safety case. We also expect the DSSC to be a continuing focus for dialogue and consultation with the regulators and other stakeholders with an interest in safety. We have produced this document now mainly to provide information on our methodology for progressive development of the safety case, in order to obtain feedback on our approach prior to the conduct of site-specific analyses.

Wider dialogue on the ESC - and the DSSC overall - is supported by presentation of the ESC as much as possible in “plain English”, while recognising that informed environmental regulators are an important audience for this document.

We will update the ESC at regular intervals as appropriate and as required to meet regulatory expectations. Our intention is to use the framework we have developed for the generic ESC to prepare a site-specific ESC in due course, considering any comments we receive on the generic ESC, lessons learned in the preparation of the generic documentation, and accounting for the differences inherent in moving from a generic stage to a site-specific stage. This approach is consistent with a staged development and approval process, as advocated by the environmental regulators.

Executive summary

1 Introduction

The Nuclear Decommissioning Authority, through its Radioactive Waste Management Directorate, is responsible for implementing a geological disposal facility (GDF) for higher activity radioactive wastes. This is set out in the UK Government's Managing Radioactive Waste Safely White Paper (MRWS) along with the details of the radioactive waste inventory (Baseline Inventory) that requires disposal.

The wastes for disposal in a GDF include high-level waste (HLW), intermediate-level waste (ILW), and low-level waste (LLW) unsuitable for near-surface disposal. We also consider other nuclear materials that have not been declared as wastes by their owners, but which might be declared as wastes in the future if it were decided they had no further use, namely spent nuclear fuel (SF), separated plutonium and uranium. This inventory is associated with activities primarily related to the generation of nuclear power, and the Baseline Inventory includes "legacy" materials from existing nuclear facilities. We also consider the implications of disposing of an Upper Inventory, including additional wastes that might be generated in the future from a possible programme of new nuclear power stations.

The implementation of a GDF requires us to demonstrate our confidence that such a facility would be safe, including during both the operational period and in the long-term after the closure of such a facility. The Environmental Safety Case (ESC) is the vehicle we use to demonstrate our understanding of environmental safety.

The *Geological Disposal: Generic Environmental Safety Case main report* (and its supporting documents) demonstrates that we are confident that a GDF could be developed to meet the guidelines set down by the environmental regulators.

2 Context and objectives

The Environmental Safety Case needs to address the fundamental protection objectives laid down by the environmental regulators in their Guidance on Requirements for Authorisation (GRA). The GRA provides a set of criteria, both numerical and qualitative, against which the environmental safety of a GDF will be assessed during the operational and post-closure periods. We envisage that an ESC should be developed as part of the staged process before an environmental permit for underground operations would be granted, and that the ESC will continue to be updated and become increasingly more detailed as a GDF is developed. At this early stage in the MRWS site selection process, which is based on voluntarism and partnership with potential host communities, we cannot produce an ESC using site specific information and therefore our current ESC is generic.

Guidance on geological disposal has been developed over many years by various international bodies including the Nuclear Energy Agency (NEA), European Commission (EC) and the International Atomic Energy Agency (IAEA).

The scientific consensus worldwide is that geological disposal is technically feasible. The US has an operational GDF (the Waste Isolation Pilot Plant, or WIPP), and Finland, France, and Sweden are making good progress towards establishing new GDFs. Germany is now backfilling and closing a previously operational GDF (at Morsleben) and is scheduled to open a new GDF within a few years (at Konrad) for non-heat-producing wastes. We can learn from the experience of overseas implementation programmes and there is substantial international co-operation and consensus on the issues faced.

3 Our safety strategy

Our strategy for ensuring the environmental safety of a GDF consists of a design and siting strategy, an assessment strategy, and a management strategy, as follows:

- ***Design and siting*** – We will use the ESC to assist in the various planning stages of a GDF including; siting, layout, operation and closure. This will be within the site volunteered by a particular host community and for the preferred disposal concept. Disposal facility design will consider the inventory, will follow international good practice and the GRA requirements for passive safety, and use the safety functions of multiple barriers to provide that safety.
- ***Assessment*** – Our assessment strategy follows international good practice and the requirements of the GRA. Some components of this strategy will not be implemented until detailed information from specific candidate sites is available at future stages of a GDF implementation programme. However, we have implemented those components of the strategy needed to provide confidence in our ongoing assessment of waste packaging proposals by waste producers (through the disposability assessment process), to demonstrate GDF viability, and to inform our initial desk-based assessments of candidate sites once these are available.
- ***Management*** – The overall management strategy we need for the near future is already largely in place. This has been designed to show that we can deliver the disposal system specification, and the design and siting, and assessment strategies, in a coherent, integrated way over the long timescales required for the GDF programme. However, our management strategy will need to develop in the future to meet the needs of the programme as it evolves (e.g. to control site characterisation and eventual construction, operation and closure).

4 Assessment basis

Our disposal concept needs to meet two complementary high-level environmental safety objectives:

- **Isolation –** By isolation we mean removing the waste from people and the surface environment. Geological disposal provides isolation, therefore 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 from the impacts of climatic and other natural environmental events and shielding of the natural environment from direct radiation from the waste.
- **Containment –** By containment we mean retaining radionuclides within various parts of the multi-barrier system for as long as required. 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 decay products reduce 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. The geological barrier provides further containment by delaying the movement of any small amounts of long-lived radionuclides that are released from the engineered barrier system. Locating the GDF in a suitably deep and stable environment protects the engineered barriers, helping them to preserve their containment functions for longer times.

We have considered different examples of geological disposal concepts – drawing on UK and overseas experience – that are relevant to the UK context, inventory, and available geological environments. The illustrative examples used are all based on the principle of passive safety provided by a combination of engineered barriers designed to complement the natural barrier provided by the geological environment. This system of multiple barriers ensures that the radioactivity in the wastes is sufficiently contained so that regulatory requirements are met. The GRA requires exposures resulting from any releases to the surface to be as low as reasonably achievable and, much less than the amount everyone receives each year from naturally occurring sources of radioactivity.

Evaluation of impacts on the environment and people from a GDF during the operational phase is much the same process and is based on similar techniques to those used in assessing such impacts from other operating nuclear facilities (such as radioactive waste stores). For this reason, the generic ESC contains more detail on the long period following closure of a GDF – this is where the greatest challenges lie in evaluating compliance with regulatory guidance and, therefore, where we, the environmental regulators, and overseas radioactive waste management organisations have undertaken the most work.

5 Environmental safety analysis

We must show that such a facility would be safe, during both the operational period and in the long-term after the closure of such a facility.

Operational environmental safety assessment

Our high level strategy to ensure operational environmental safety is to eliminate hazards during the normal operation of a GDF and where this is not possible to provide protection to control any adverse environmental impacts.

During the operational period environmental safety is provided by the safety features inherent in waste packaging specifications, and the safety procedures and management in place during this period. The safety features of the waste packages include:

- the solid form of the wastes
- their packaging - mainly in steel or concrete containers designed to reduce the potential for radioactive releases during storage and handling
- their disposal in robust containers that provide the necessary degree of radiation shielding and containment, and are capable of normal handling during storage, transport and disposal operations.

The air underground will be filtered to remove any particles of radioactivity that might escape from the packages. Our illustrative quantitative assessment of possible discharges of radioactive gases from a GDF during the operational period indicates that the regulatory requirements could be met.

Post-closure safety assessment

Using both qualitative and quantitative reasoning, our post-closure safety assessment presents our understanding of how a GDF would evolve once it is closed. It shows how environmental safety could be provided by a system of multiple barriers working together to provide safety over long timescales, of hundreds of thousands of years.

Our understanding of post-closure performance and statements on environmental safety will come from various lines of reasoning including:

- description and analysis of the expected evolution of the geological disposal system based on understanding of the environmental safety functions provided by different disposal concepts and sites and by our research, design and site characterisation work programmes
- results of experiments in underground research laboratories in other countries under *in situ* conditions and long-term demonstration experiments
- studies of archaeological analogues, that is, materials that people have been using for hundreds or thousands of years and that have survived in the environment over long timescales and that are similar to the materials that could form part of the engineered barrier system of a GDF (e.g. glass, cement and iron)
- studies of natural systems that provide analogues for processes important in containing and retarding radionuclides in the multi-barrier system and which can

provide information over timescales comparable to or longer than those considered in our quantitative assessments (e.g. Cigar Lake in Canada)

- site-specific natural indicators of safety once we have candidate sites to consider (e.g. indicators of containment and retardation in the geological environment)
- demonstration that the geological disposal system is robust to unexpected events (e.g. climate change), uncertainties (e.g. concerning site-specific understanding), and decisions (e.g. the possible need to dispose of nuclear materials such as separated plutonium or uranium).

Quantitative studies of post-closure safety in the generic ESC focus on how safety is provided after a facility is closed. There are three ways by which radionuclides (and other contaminants) in the waste could return to the surface in the post-closure period, they might:

- dissolve in and be transported by groundwater
- be released as radioactive gases that migrate to the surface via rock fissures
- be returned directly to the surface as a result of human intrusion.

For the generic ESC we have selected some examples of disposal concepts to form the basis for modelling work and to understand the likely impacts. Almost any model by its nature is a simplification of reality; this is particularly true for models used in quantitative assessment studies over long timescales. For the illustrative example calculations undertaken as part of the generic ESC, we have represented the performance of the barriers in an appropriate simplified manner. We have varied the values of a few key model parameters in order to understand and illustrate the potential radiological impacts of disposing of the Baseline Inventory using different types of waste container in different kinds of geological environment. These parameters represent:

- rates of groundwater movement through the disposal areas
- groundwater travel time from the disposal areas back to the surface
- discharge area over which groundwater is released at the surface
- dilution of contaminated groundwater by uncontaminated groundwater in overlying rocks.

These calculations indicate that there are a wide range of parameter value combinations, representing different possible disposal concepts and geological environments, which would enable us to satisfy the radiological protection requirements in the GRA. This gives us confidence that a GDF could be designed to suit a wide variety of UK geological environments. The calculations also serve as one of the bases for our ongoing assessments of the disposability of proposed waste packages.

6 Conclusion and forward programme

The generic ESC illustrates how we could implement geological disposal safely in different geological environments for the UK's inventory of higher activity radioactive wastes. Our confidence is built on our understanding of how multiple barriers can work together to provide the required long-term safety. We therefore believe 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 of waste packaging proposals 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 when they are identified.

The staged GDF implementation process and progressive updating of the ESC will allow many opportunities for feedback from regulators and other stakeholders, and will provide opportunities for us to tailor our proposals with respect to new findings and comments we receive. The generic ESC summarises and addresses issues that have been identified by previous regulatory scrutiny. We expect ongoing dialogue with regulators and other stakeholders to inform the next update to the ESC.

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Main abbreviations and acronyms used in this report

National organisations

CoRWM	Committee on Radioactive Waste Management
HSE	Heath and Safety Executive
NDA	Nuclear Decommissioning Authority
Nirex	United Kingdom Nirex Limited
RDMB	Repository Development Management Board
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

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
OSA	Operational Safety Assessment
PCSA	Post-closure Safety Assessment
SEA	Strategic Environmental Assessment
TSC	Transport Safety Assessment

Radioactive material types

DNLEU	depleted, natural and low-enriched uranium
DU	DNLEU is sometimes referred to in our reports as DU (depleted uranium)
HEU	highly-enriched uranium
HLW	high-level waste
ILW	intermediate-level waste
LLW	low-level waste
Pu	plutonium
SF	spent nuclear fuel
U	uranium

Other acronyms

AGR	advanced gas-cooled reactor
ALARA	as low as reasonably achievable
DSS	Disposal System Specification
EBS	engineered barrier system
EDZ	excavation disturbed zone
FEPs	features, events and processes
GDF	geological disposal facility
LRQA	Lloyds' Register Quality Assurance
MRWS	Managing Radioactive Waste Safely
PWR	pressurised water reactor
R&D	research and development
WAC	waste acceptance criteria
WIPP	Waste Isolation Pilot Plant

1 Introduction

1.1 Rationale and use of the Environmental Safety Case

The concept of geological disposal of radioactive wastes dates back to the late 1950s, when it was first advocated in the US as the most appropriate way to deal permanently with higher activity solid radioactive wastes. Geological disposal means burial underground in a purpose-built facility at a depth in the range 200 – 1,000 metres, with no intention to retrieve the wastes. The overall aim of geological disposal is to remove a hazardous material from the immediate human, and rapidly changing, surface environment to a stable location where it will remain, protected from disturbance by disruptive natural or human processes.

An Environmental Safety Case (ESC) for a geological disposal facility (GDF) is a set of claims concerning the environmental safety of the disposal of radioactive waste in a GDF, substantiated by a structured collection of arguments and evidence. Such an ESC needs to address environmental safety at the time of disposal and in the long-term, after wastes have been emplaced and the facility has been closed. We know that the materials that we place underground will slowly degrade and that even the most stable geological environments will eventually change with the passage of geological time, but the hazard potential of the wastes also decreases by radioactive decay. Our safety assessment work looks at the balance of these processes so that we can evaluate the environmental safety of a GDF far into the future, as well as at the time of disposal.

An ESC needs to address the fundamental protection objective contained in the environmental regulators' Guidance on Requirements for Authorisation (GRA) [1, paragraph 4.21]:

> *"…to ensure that all disposals of solid radioactive waste to facilities on land are made in a way that protects the health and interests of people and the integrity of the environment, at the time of disposal and in the future, inspires public confidence and takes account of costs."*

The approach to achieving this in the ESC is by addressing the more detailed regulatory principles and requirements contained in the GRA, which encompass management, radiological and technical aspects of the safety case for a GDF. From April 2010, any application relating to the disposal of radioactive wastes at a GDF in England or Wales would be made under the Environmental Permitting Regulations (EPR 2010) [2], and would be supported by an ESC. EPR 2010 largely supersedes the Radioactive Substances Act 1993 (RSA 93) [3] in England and Wales, but much of what was contained in RSA 93 now appears in Schedule 23 of EPR 2010.

The Nuclear Decommissioning Authority (NDA) has been charged with implementing the UK Government's[2] policy for the long-term management of higher activity radioactive waste by planning, building and operating a GDF. Within the NDA, the Radioactive Waste Management Directorate (RWMD) is tasked with development of a GDF - the "delivery organisation" in the GRA [1]. In the future, it is envisaged that RWMD will be established as a subsidiary of the NDA, able to hold the necessary environmental permits and nuclear

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[2]Throughout this document, the term UK Government includes all relevant departments and devolved administrations, but excludes the Scottish Government. The Scottish Government does not support geological disposal – it advocates interim near-surface, near-site storage and disposal facilities so that the waste is monitorable and retrievable and the need for transporting it over long distances is minimal.

site licence, and contracted to deliver a GDF. We, RWMD, are currently operating under voluntary scrutiny by the regulators as a “prospective Site Licence Company” in order to demonstrate and develop the competences required of a future holder of an environmental permit and nuclear site licence.

The UK Government’s 2008 Managing Radioactive Waste Safely (MRWS) White Paper [4] defines the framework for implementing geological disposal. The White Paper sets out the stages in a GDF site selection process that would lead to the identification of sites for desk-based studies, followed by surface investigations at candidate sites, and leading ultimately to the identification of a preferred site. The White Paper also sets out the materials that may need to be managed through geological disposal; these are high-level waste (HLW), intermediate-level waste (ILW), some low-level waste unsuitable for near-surface disposal (LLW), spent nuclear fuel (SF), separated plutonium (Pu) and uranium (U), including both highly enriched uranium (HEU) and depleted, natural and low-enriched uranium (DNLEU). SF, Pu, HEU and DNLEU are not currently considered wastes by their owners (because they are still considered to be of potential use), but might be declared as wastes in the future. Additional ILW and SF, arising from any possible programme of new nuclear power stations, would also be managed through geological disposal.

The GRA [1] makes it clear that an ESC should be developed as part of the staged process before an environmental permit for underground operations would be granted, and that the ESC will continue to be updated and become increasingly more detailed as a GDF is developed. A fully developed ESC will be a substantial suite of documents that brings together the results of many years of desk, laboratory, design and site-based work. At this stage in the MRWS Site Selection Process, prior to the identification of sites for desk-based studies, we cannot produce an ESC using site-specific information; therefore, the ESC is currently generic as it does not relate to any specific site or disposal facility design.

This generic ESC is based on our understanding of the scientific and engineering principles underpinning geological disposal. We have used examples of different disposal concepts that have been developed around the world for a variety of geological environments and for a range of waste types similar to those we have to consider, to illustrate the types of engineered and natural barriers that could be used for a GDF in the UK. We have also used some of these concepts as the basis for undertaking illustrative calculations to understand the likely impacts of a GDF. These calculations help us quantify the *relative impacts* of disposing of different types of waste in a GDF. This information is of value at the present time in helping us advise waste producers on how different wastes should be packaged for disposal, and in assessing the implications of potential new wastes from any new nuclear power stations.

In the future, as we move forward with the MRWS Site Selection Process, we will replace these illustrative calculations with a detailed assessment of the selected site(s) and chosen GDF design(s).

Key aims of this generic ESC are to:

- Set out our understanding of the requirements of an ESC, consistent with the GRA, explaining how we will use the ESC at various hold points in the implementation process for a GDF (Section 2).
- Explain our safety strategy for a GDF and the way in which we will build confidence in environmental safety through a range of qualitative and quantitative lines of reasoning (Section 3).
- Provide arguments on the environmental safety of a GDF with reference to the principles and top-level requirements of the GRA; and, consistent with being at a generic stage, show that safety could be provided by a combination of engineered and natural barriers in different geological environments and illustrate how a GDF could be implemented in these environments (Sections 4 and 5, Appendices A, B

and C). The safety arguments are based on qualitative discussion and illustrative assessment calculations for a range of illustrative geological disposal concept examples and associated GDF designs as applied to the UK.

- Provide a continuing basis for our assessment of waste packaging proposals (“disposability assessments”) and the issue of Letters of Compliance (LoCs) to UK waste producers (Sections 3-5).
- Help provide an appropriate basis for undertaking assessments of candidate sites as part of the MRWS Site Selection Process (Sections 3-5).
- Identify the research and development (R&D) work needed to provide relevant evidence and develop confidence in the qualitative and quantitative environmental safety arguments presented in future updates of the ESC (Sections 5-6).
- Help demonstrate that we are developing the capability to perform the functions of a Site Licence Company in due course (Sections 2-6).

The ESC is just one part of the overall generic Disposal System Safety Case (DSSC) that we have produced as part of our preparatory studies in the first phase of the development programme for a GDF. The generic DSSC explains and assesses the safety and environmental implications of all aspects associated with the geological disposal of higher activity radioactive waste in the UK. The generic DSSC covers three main areas; for each we have prepared a separate safety report:

- transporting the waste to a GDF – the safety arguments and assessment of this are presented in the **generic Transport Safety Case (TSC)** [5]
- construction of and emplacement of waste within a GDF, subsequent storage and eventual backfilling, decommissioning and closure – presented in the **generic Operational Safety Case (OSC)** [6]
- the environmental safety of a GDF during the operational period and after its closure – presented in this **generic Environmental Safety Case (ESC)**

These programmes of work are coordinated to ensure that, for example, the design, construction and operation of a GDF meet the requirements for long-term safety as set out in the disposal system specification. When integrated together, the TSC, OSC and ESC and their supporting documents comprise the DSSC, an integrated statement of the safety of the complete disposal system.

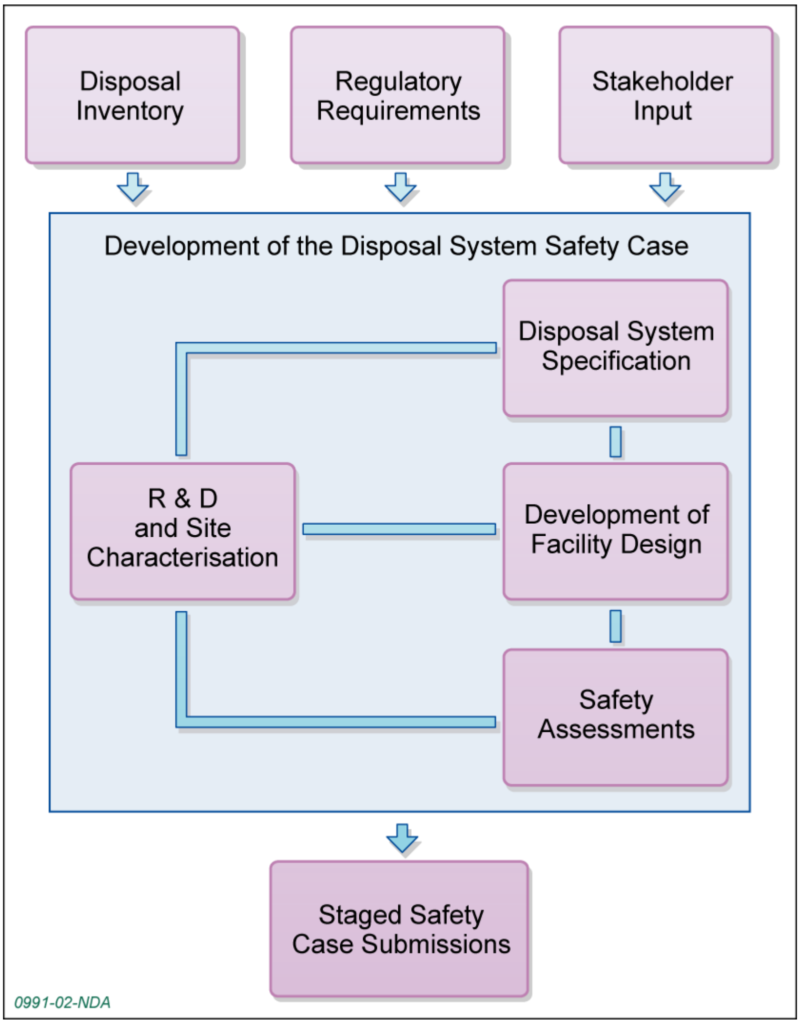
The DSSC serves as an integrating tool within the development programme for a GDF. It brings together our work in topic areas as diverse as disposal system specification, design, R&D, site characterisation, safety assessment, inventory specification, and stakeholder and regulatory dialogue. For each successive update of the DSSC, we need to integrate activities in these topic areas as shown in Figure 1.1.

The generic DSSC provides a platform that can be updated with time, as discussed in Sections 2.1 and 2.2. In particular, the ultimate objectives of the ESC will only be met after several iterations, as progressively more information on candidate sites and disposal facility designs becomes available. Updates of the ESC are foreseen at key stages in the implementation programme for a GDF, e.g. prior to the start of surface-based intrusive site investigations (e.g. boreholes), prior to the start of underground operations, prior to construction of disposal areas, prior to waste emplacement, prior to sealing and closure of a GDF, and after closure, prior to the termination of regulatory control. We envisage that this progressive development of the ESC will provide a focus for consultation and dialogue with those involved in the MRWS Site Selection Process, including providing a basis for examining key issues relevant to the major decisions at each stage. This report is therefore the first in a series that will be developed throughout the MRWS Site Selection Process.

Our intention is to use the framework we have developed for the generic ESC to prepare a site-specific ESC in due course, considering any comments we receive on the generic ESC, lessons learned in the preparation of the generic documentation, and accounting for the differences inherent in moving from a generic stage to a site-specific stage.

Figure 1.1 Interaction between different topic areas that are addressed and integrated in developing the ESC

Note that the Disposal System Specification covers environmental safety, transport safety, and operational safety.



As the programme evolves, we will use the ESC to:

- continue to provide disposability assessments to waste producers;
- inform choices between disposal concepts and assist in the optimisation of preferred concepts;
- identify and focus research and site characterisation that inform technology development, provide relevant evidence, and develop confidence in the ESC;
- further develop our technical and management systems as needed to support confidence in GDF construction, operation and closure;
- continue to promote dialogue internally and externally about the developing ESC; and
- prepare an application for disposal of wastes that meets the regulatory guidance (the GRA) set out by the environment agencies.

In structuring this document in the form of an ESC main report, we are seeking feedback on structure, content and safety strategy. In particular, we are keen to gain an early view from the Environment Agency and other interested parties on our understanding and implementation of the requirement to produce an ESC, and our proposed approach to developing an acceptable basis for operating and ultimately closing a GDF.

For the purpose of the DSSC, the **disposal system** includes transport of waste to a GDF and its handling and management in the surface and underground facilities at the disposal site. Therefore, for the sake of clarity, we use the term **geological disposal system** when we are referring only to a GDF (surface and underground facilities) and the geological environment in which the underground facilities are located.

1.2 Approach to developing the generic Environmental Safety Case

The development of the ESC is led by RWMD staff (see Section 3.3.1.1 for a description of staff competence), with support from key contractors having experience of environmental safety case development for GDFs in other countries and for UK and overseas near-surface radioactive waste disposal facilities.

Many countries are investigating the geological disposal of radioactive waste in order to provide long-term protection for people and the environment. The MRWS White Paper [4] notes that 25 countries have opted for a policy of geological disposal. The US has an operational GDF for some long-lived “transuranic” wastes (the Waste Isolation Pilot Plant, or WIPP), and Finland, France and Sweden are making good progress towards establishing new GDFs for long-lived wastes, including SF. Germany is now backfilling and closing a previously operational GDF (at Morsleben) and is scheduled to open a new GDF within a few years (at Konrad) for non-heat-producing wastes. The work of the organisations responsible for developing these disposal facilities is publicly available.

We can learn from the experience of GDF implementation overseas - and we are actively involved with work in other countries through co-operation agreements with overseas national waste management organisations, and through international organisations such as the European Commission (EC), the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (NEA/OECD), and the International Atomic Energy Agency (IAEA). There is substantial international co-operation and consensus on the issues faced. Guidance on different aspects of geological disposal has been developed over many years through such organisations as the EC, the NEA and the IAEA. The work of these organisations is also available to the public.

When we started the generic DSSC development work, we were able to draw on results from both the previous programme of work in the UK by United Kingdom Nirex Limited (Nirex) and the wide range of international experience of safety assessment and safety case development over the last 40 years. The previous work illustrates how geological disposal can be implemented safely in many different kinds of geological environment and for many different types of radioactive waste. We are using relevant parts of this information to develop our own approach based on the best experience and practice worldwide. Specifically, in developing a safety case we have taken account of international guidance [e.g. 7, 8] as well as the additional requirements for an ESC provided in the GRA [1].

The GRA was updated in 2009 to take account of the latest international guidance. Throughout this document we provide annotations (using boxed text at the start of appropriate sections, starting in Section 2) to refer to specific high-level regulatory principles and requirements that are being addressed by the various elements of our approach.

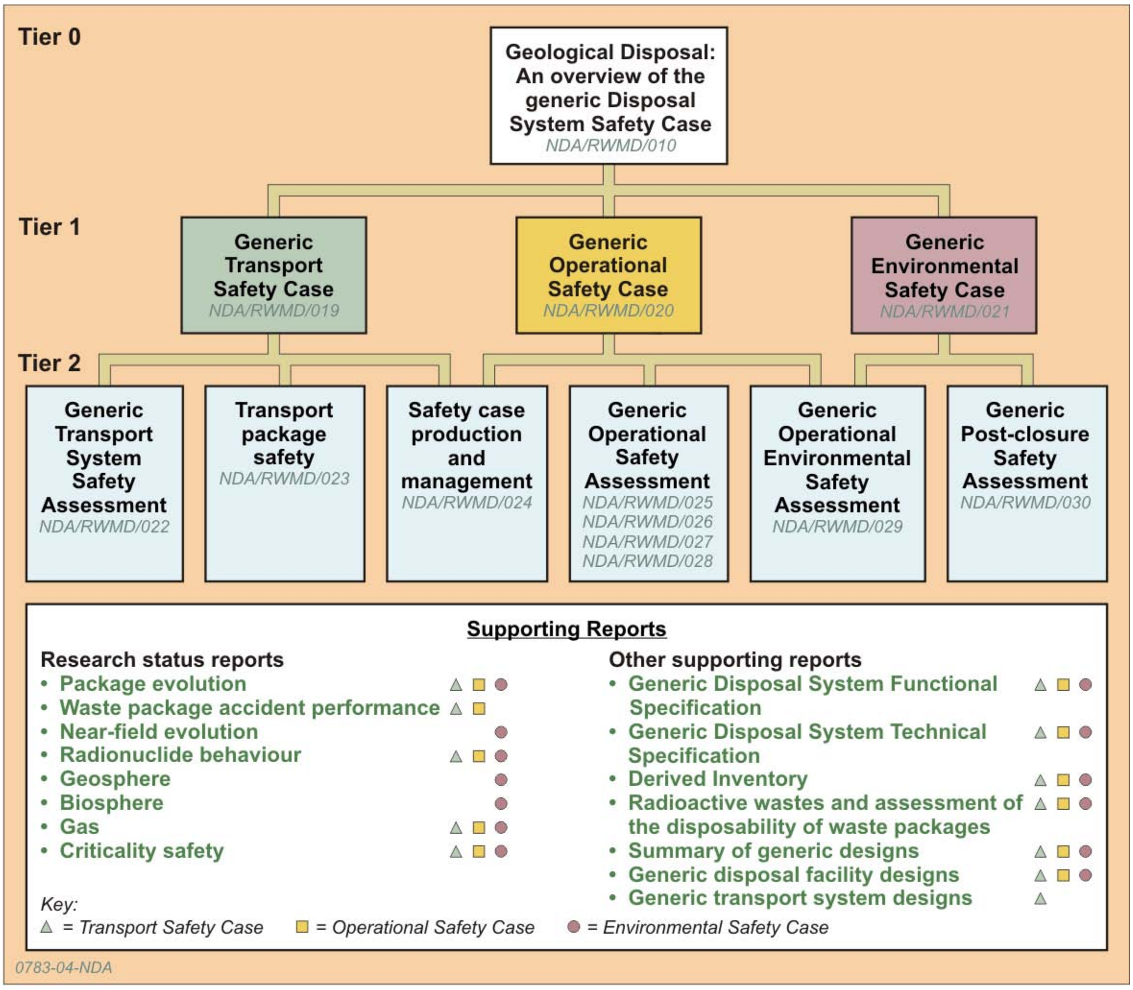
1.3 Structuring the Environmental Safety Case documentation

This Generic ESC main report forms part of a tiered suite of documents being produced by us to support formal regulatory submissions and dialogue during the development of a GDF (Figure 1.2).

At the top of the hierarchy is the DSSC overview report (Tier 0). Below the overview report are three documents ("main reports") written to support submissions for the three main regulatory disciplines of transport safety, operational safety, and environmental safety (Tier 1). The three Tier 1 safety case documents build on a set of "safety assessments" and a set of "supporting reports", all at Tier 2.

Figure 1.2 Disposal System Safety Case document hierarchy

The safety cases consist of a main report at Tier 1, a set of assessment reports at Tier 2, and a set of research status reports on key topics relevant to the initial conditions and evolution of the disposal system and other supporting reports also at Tier 2. The full set of Tier 2 reports that form part of this ESC is set out in Table 1.1.



The Tier 2 supporting reports are of three types:

(i) reports that summarise our understanding of the initial conditions of the main components of a GDF (i.e., the inventory, disposal system specifications, and design reports);

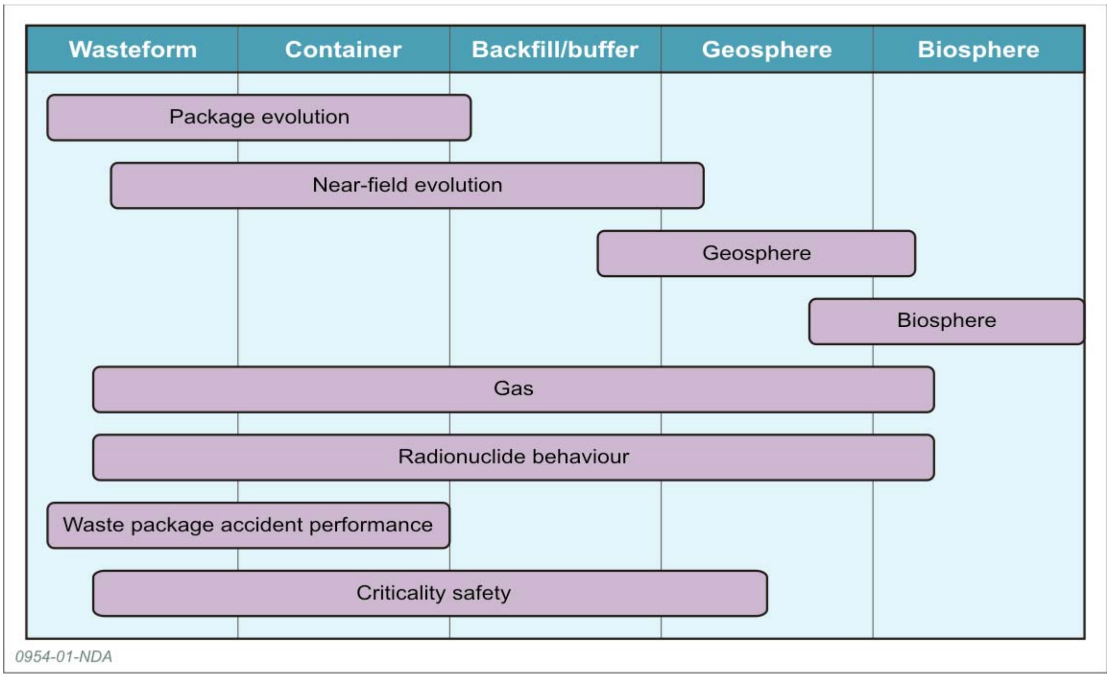
(ii) reports that discuss the behaviour and evolution of key physical components of the engineered and natural systems (i.e., the package evolution, near-field evolution, geosphere, and biosphere research status reports); and

(iii) reports dealing with specific cross-cutting issues (i.e., the criticality safety, gas, and radionuclide behaviour research status reports).

We refer to the second two types of report as “research status reports”. The research status reports summarise current understanding of key research topics important to safety, and act as a route map into more detailed supporting literature. The relationship between the Tier 2 research status reports and the main components of the geological disposal system is shown in Figure 1.3.

Figure 1.3 Relationship between the Tier 2 research status reports and geological disposal system components we have to consider in developing safety arguments

All of these research status reports form part of the ESC, with the exception of the Waste package accident performance status report.



The ESC comprises the Tier 1 document (this report), plus those Tier 2 documents that are relevant to the discussion of environmental safety in this report. The full set of Tier 2 reports that form part of this ESC is set out in Table 1.1.

Table 1.1 List of ESC Tier 2 reports

Area	Title and Reference
Inventory	• Summary of the Derived Inventory based on the 2007 UK Radioactive Waste Inventory [9] • Development of the Derived Inventory for ILW and LLW based on the 2007 UK Radioactive Waste Inventory [10] • Development of the Derived Inventory for HLW and spent fuels based on the 2007 UK Radioactive Waste Inventory [11] • Production of the Derived Inventory for uranium and plutonium [12] • Production of the Derived Inventory for new build reactor wastes [13] • Geological Disposal: Radioactive wastes and assessment of the disposability of waste packages [14]
Research Status Reports – Conceptual Understanding of Processes	• Geological Disposal: Package evolution status report [15] • Geological Disposal: Near-field evolution status report [16] • Geological Disposal: Geosphere status report [17] • Geological Disposal: Biosphere status report [18] • Geological Disposal: Gas status report [19] • Geological Disposal: Radionuclide behaviour status report [20] • Geological Disposal: Criticality safety status report [21]
Disposal System Specification, and Generic Design Reports	• Geological Disposal: Generic Disposal System Functional Specification [22] • Geological Disposal: Generic Disposal System Technical Specification [23] • Geological Disposal: Generic disposal facility designs [24] • Geological Disposal: Generic disposal facility designs summary report [25]
Assessment Reports	• Geological Disposal: Generic Operational Environmental Safety Assessment 2010 [26] • Geological Disposal: Generic Post-closure Safety Assessment 2010 [27]

We note that the overall social and economic impacts of constructing and operating a GDF, and the associated environmental impacts such as noise and resource use, will be covered under the land-use planning process and are outside the scope of the ESC, as discussed in Section 2.5 and Appendix D.[3]

The intent of this Tier 1 ESC main report is to cover the environmental safety regulations and requirements associated with the disposal of radioactive waste in a GDF in a single summary document, with referencing to the Tier 2 reports as appropriate. Overall, the Tier 2 reports provide more detail on the key issues that are of significance to the safety of a GDF. Several of them support more than one part of the DSSC (such as the Derived Inventory reports, which are an important part of all three safety cases), and some of them are specific only to other parts of the DSSC (e.g. the transport reports).

At this stage, it is not appropriate, or possible, to produce a full set of the Tier 2 reports that would be required to support a full ESC, and we have focused on developing the reports

[3] Note that our consideration of social and economic impacts with respect to optimisation of a GDF is discussed in Section 3.1.2.

that are relevant to this generic stage. We expect the Tier 2 supporting reports to evolve significantly at the next update of the DSSC, when we will have site-specific information and designs. In particular, for updates of the ESC, Tier 2 supporting reports on site characteristics and site characterisation plans, on monitoring plans, and on option assessments and optimisation studies will be introduced. We will also need to develop other supporting reports in due course as appropriate.

Below the Tier 2 reports, there is a large body of documentation, including our R&D reports, quality assurance records, contractor and third-party reports, and scientific literature.

1.4 Report outline

This document's structure is outlined in Figure 1.4, which is based on headings suggested by the NEA [8]. We have adapted the NEA's suggested headings to serve the purpose of an ESC in the UK context.

Section 2 provides more detail on the context and objectives of an ESC and this generic ESC in particular, with respect to the MRWS Site Selection Process, the environmental permitting process, and wider international and national legislation.

Section 3 sets out our safety strategy. We outline our safety concept for a GDF and describe our approach to optimisation and optioneering, design, and site investigation and R&D. Our safety strategy also describes our approach for assessing environmental safety, including our approach to managing uncertainty and dealing with open questions. Much of the strategic thinking that forms our design and siting strategy and our assessment strategy is aspirational, because we are still at a generic stage and do not yet have specific sites and designs.

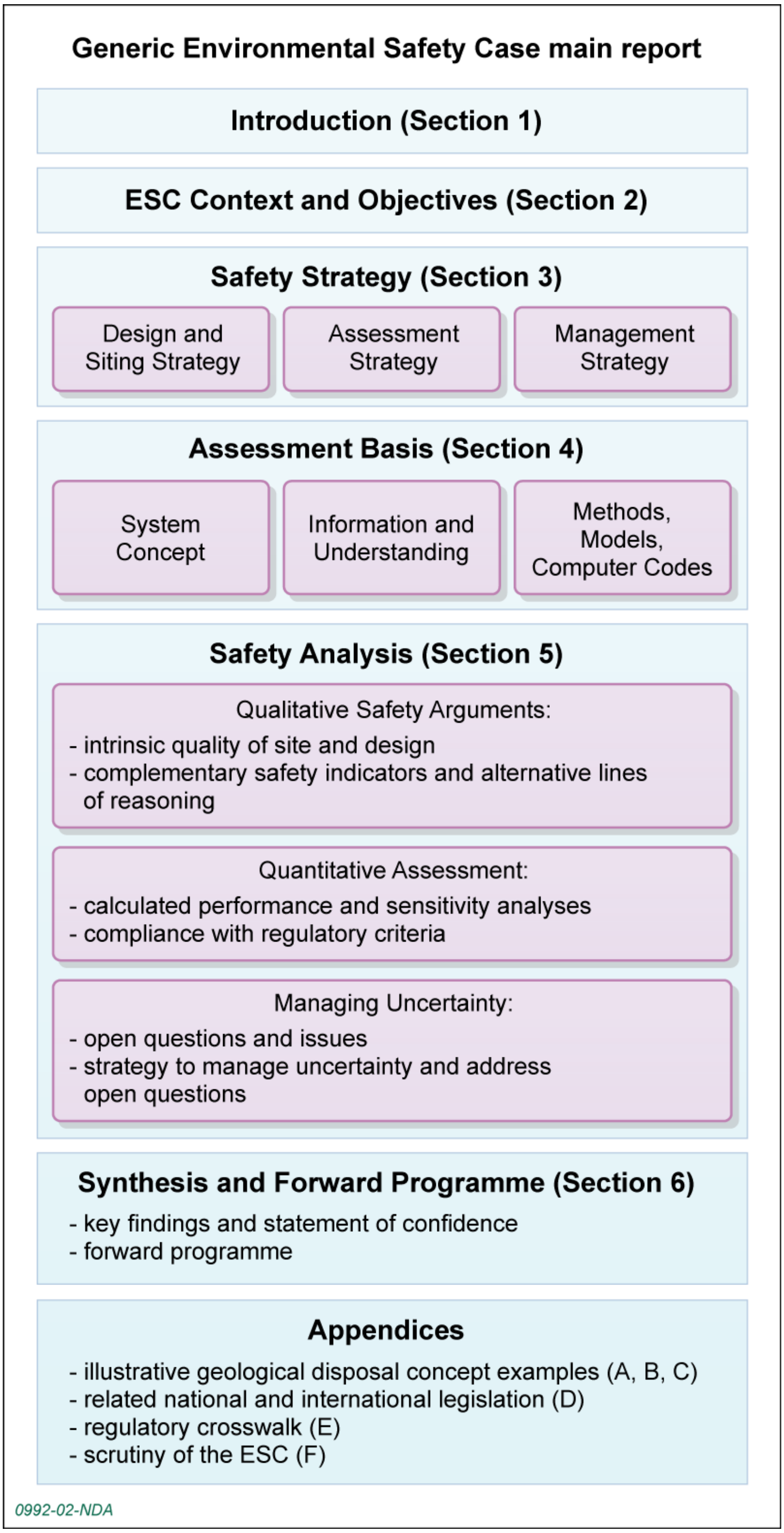
The various management systems we have in place for planning and controlling our work to the high standards of quality required are also described in Section 3. We consider it particularly important that the systems we need for the near future are in place and documented prior to starting significant work at specific sites. The relevant parts of our management strategy are already being exercised, for example in influencing the content and controlling the production of the DSSC documentation. However, significant parts remain to be developed and exercised, e.g. relating to the management of site characterisation and eventual GDF construction and operation.

Section 4 describes the assessment basis – for the generic ESC this focuses on components of the assessment basis that are valid for a range of possible geological environments. This section is relatively brief at this early stage of the MRWS Site Selection Process. In future site-specific updates to the ESC, this section will summarise and provide pointers to significant information on GDF design, the scientific and technical information that underpins the safety case, and methods, models and computer codes and databases that support our understanding of the disposal system and its evolution.

Section 5 includes a range of qualitative and quantitative safety arguments. For the generic ESC this focuses on generic qualitative arguments that are valid for a GDF in different geological environments, and illustrative examples of calculations of potential operational environmental impacts and of potential post-closure impacts, drawn from our safety assessment reports. We also summarise the key uncertainties that currently exist. In future site-specific updates to the ESC, this section will summarise and provide reference to safety information relevant to the identified site(s).

Figure 1.4 Structure of the Generic ESC main report

Table 2.1 sets our where in this report structure each of the GRA requirements is considered.



The key findings of the generic ESC are brought together in Section 6, and issues requiring further work are summarised. In future site-specific updates to the ESC, this section will provide a description of our forward work programme tied into findings and specific gaps in the ESC.

Appendices A, B and C contain summary information specific to the UK application of a set of illustrative geological disposal concept examples developed for three different host rocks – higher strength rocks, lower strength sedimentary rocks, and evaporites. These appendices include conceptual design information, scientific and technical information and safety arguments relevant to each illustrative example. These appendices also provide a high-level summary of – and introduction to – many of the supporting reports (research status reports, generic design reports) that form part of this generic ESC.

Our decision on suitable illustrative geological disposal concept examples to consider at this generic stage of the programme was informed by a suite of disposal concept option studies [28, 29, 30, 31].

At this generic stage, the illustrative geological disposal concept examples are needed for the following reasons [32]:

- to further develop our understanding of the functional and technical requirements of the geological disposal system;
- to further develop our understanding of design requirements;
- to support the scoping and assessment of the safety, environmental, social and economic impacts of a GDF;
- to support development and prioritisation of our R&D programme;
- to provide a basis for our analysis of the potential costs of geological disposal; and
- to support our assessment of the disposability of waste packages proposed by waste holders.

However, this does not necessarily mean that any of the illustrative disposal concepts discussed at this generic stage represent the concept intended to be used in a particular geological environment. At this stage, no geological disposal concept has been ruled out.

Appendix D describes international and national legislation and documents relevant to the ESC and the evaluation of environmental safety.

Appendix E presents a table documenting how we intend to show where GRA requirements are considered in ESC documentation and identifying forward work programmes to address requirements where there are gaps.

Appendix F summarises previous regulatory scrutiny of work that is particularly relevant to the development of the ESC. Each update of the ESC will have an appendix that identifies reviews of the preceding version of the ESC and our responses to those reviews and/or where in the ESC the comment has been considered. In addition, we will identify the main changes between successive updates of the ESC.

2 Environmental Safety Case context and objectives

The context for the ESC relates to the current stage in the decision-making process and explains in more detail why the safety case is being produced, when, why and how it will be updated with time, and the framework against which it will be evaluated. This section is structured as follows:

- Section 2.1 describes the stages of the MRWS Site Selection Process, and outlines points at which the ESC will be updated with reference to this process and the environmental permitting process.
- Section 2.2 describes the objectives of the ESC as it evolves with time to meet the requirements of both the MRWS Site Selection Process and the environmental regulators.
- Section 2.3 describes the regulatory context and process within which the ESC work is conducted, and the manner of our engagement with the environmental regulators.
- Section 2.4 describes the context of wider dialogue on the ESC and what we have done to satisfy the requirement for wider dialogue at the generic stage of the MRWS Site Selection Process, prior to specific sites being known, and how we intend to move forward with wider dialogue on the ESC.
- Section 2.5 provides a pointer to related international and national environmental safety legislation and documents, which are discussed in more detail in Appendix D.

2.1 The MRWS Site Selection Process and progressive updating of the ESC

The UK Government's 2008 MRWS White Paper [4] sets out a series of stages in the selection of a site for a GDF and the development and operation of a GDF, and identifies the decisions to be made and stakeholders involved at each stage.

The guidance provided in the GRA [1] on the timing of development of the ESC has been written to be compatible with the MRWS Site Selection Process. However, whereas the UK Government's 2008 White Paper [4] just considers GDF implementation, the GRA considers the entire life cycle of GDF implementation, operation and closure, and the regulatory approvals required throughout this process.

We discuss first the timing of ESC updates with regard to the MRWS Site Selection Process (Section 2.1.1), and then with regard to the staged environmental permitting process under EPR 2010 [2] and the GRA [1] (Section 2.1.2).

2.1.1 Timing of ESC updates with regard to the MRWS Site Selection Process

The stages of the MRWS Site Selection Process are illustrated in the left-hand side of Figure 2.1, and are described below, along with the ESC work we have to do during each stage:

- ***Stage 1: Invitation issued and Expression of Interest from communities.***
  Stage 1 was launched in June 2008 with publication of the UK Government's MRWS White Paper [4]. Stage 1 is being undertaken independently of ourselves, led by the UK Government. Communities can express an interest in participating in the MRWS Site Selection Process without commitment in order to open discussions with the Government.

- ***Stage 2: Consistently applied 'sub-surface unsuitability' test.*** Once a community has made an Expression of Interest, another independent organisation, the British Geological Survey, makes an assessment on behalf of the UK Government of whether sub-surface conditions mean that areas covered by the community are obviously unsuitable for development of a GDF. This assessment is against the sub-surface screening criteria listed in the MRWS White Paper [4]. The aim of this stage is to eliminate any obviously unsuitable sites.
- ***Stage 3: Community consideration leading to Decision to Participate.*** If the sub-surface screening criteria do not rule out the whole area covered by the community, the community will need to decide whether to participate further in the siting process, still without commitment. Following a Decision to Participate, the UK Government expects that a formal Community Siting Partnership will be set up, such that the host community, decision-making bodies, and wider local interests will work with us and other relevant interested parties for the remaining stages.

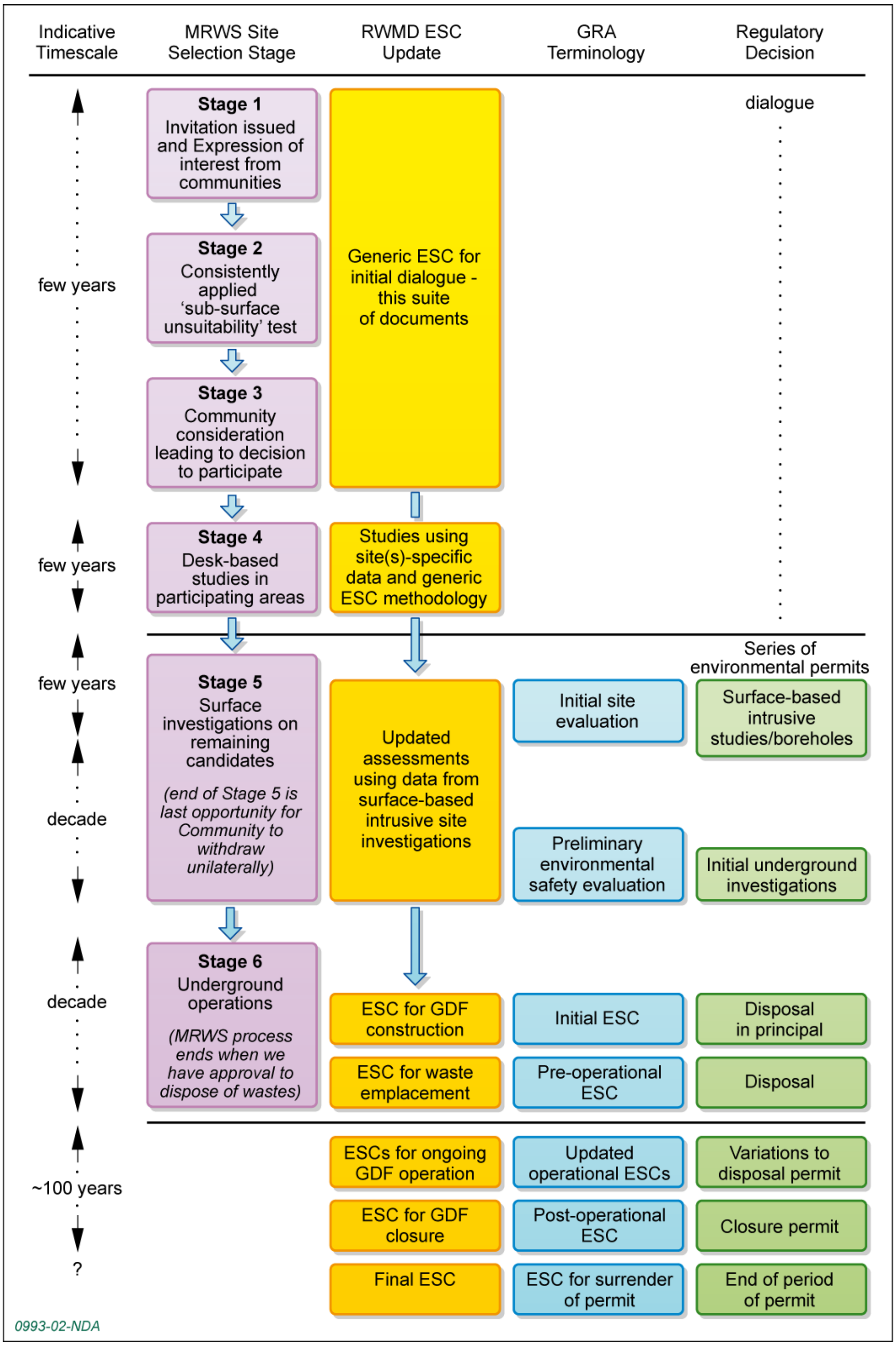
  The generic ESC has been assembled during Stages 1-3 of the MRWS Site Selection Process, and will be used by us in Stage 4.
- ***Stage 4: Desk-based studies in participating areas.*** Once the Community Siting Partnership has identified one or more potential sites, we will undertake desk-based assessments and evaluations of the sites carried forward to this stage. These studies will focus on the suitability of a specific site or sites within each potential host community area. The assessments will involve gathering information about the candidate communities and sites and evaluating them against criteria that will have been previously agreed by the UK Government. Following a request from the Government, we have published proposals for a site assessment methodology [33] for consideration and comment. **In particular, we propose that for any candidate site to remain in the MRWS Site Selection Process and be carried through to the next stage, there must be an ability to develop a disposal concept for a GDF appropriate to the geological environment that is likely to meet operational and long-term safety requirements, and security and safeguards requirements.** During Stage 4 we expect that we will be able to be more specific about relevant disposal concepts for the geological environments under consideration, depending on the extent and nature of pre-existing information about the geological environment for the candidate site(s).

  Several of the evaluation objectives we have proposed for the desk-based assessments in Stage 4 of the MRWS Site Selection Process are relevant to the ESC. Consistent with the MRWS White Paper [4], these objectives include:

  1. finding a suitable geological setting for a GDF;
  2. minimising the potential impact of a GDF on people during GDF site investigations, construction, operation and closure;
  3. minimising the potential impact of a GDF on the natural environment and landscape;
  4. maximising the beneficial effect of a GDF on local socio-economic conditions and minimising any adverse effect;
  5. minimising the requirement for transport to a GDF and the provision of transport infrastructure; and
  6. minimising cost and timing and maximising ease of implementation of a GDF.

Figure 2.1 The stages in the MRWS Site Selection Process, illustrating when we intend to provide updates to the ESC, and links to documents requested by the environmental regulators to support dialogue and environmental permitting decisions

Indicative timescales are shown on the left side of the figure.



Each of these objectives covers between two and four specific criteria (see [33] for a full list of our proposed evaluation criteria). We will work with Community Siting Partnerships to ensure that local issues are addressed in our assessments. Discussions will also take place on how to ensure that a GDF is acceptable to the potential host community and contributes to its social and economic well-being. The Stage 4 assessments will be reviewed by the regulators and the UK Government's advisory body, the Committee on Radioactive Waste Management (CoRWM). The Community Siting Partnerships will make recommendations to their local decision-making bodies, who will then decide whether to proceed to the next stage of the Site Selection Process. On the basis of the assessments, reviews, recommendations and decisions of all parties, the UK Government will then select one or more candidate sites to take forward to Stage 5 of the MRWS Site Selection Process.

Where appropriate to conducting the Stage 4 siting evaluations, we will apply the assessment methods used in this generic ESC to provide arguments regarding our ability to produce a safety case at the candidate site(s). Our focus will be on providing sufficient understanding of the properties of the site(s) to identify site-specific safety strategies and disposal concepts, key safety arguments, and the site-specific evidence that supports them or the type of evidence that is expected to support them. The Stage 4 site assessments will therefore start the process of translating the generic ESC into one or more site-specific ESCs when the appropriate information is available.

- ***Stage 5: Surface investigations on remaining candidates*.** We will need to obtain a great deal of site-specific information to develop and build confidence in GDF designs tailored to individual sites. These investigations are likely to include non-intrusive geophysical surveys and, later, the drilling of boreholes to various depths to investigate the local geology in more detail. Surface-based investigations could last a number of years, and will support more detailed assessments of the sites. We will need to obtain planning permission to undertake some of the investigations (e.g. boreholes). The ESC will support our application to the environmental regulators under EPR 2010 [2] for an environmental permit to commence drilling work at the candidate site(s). Thereafter, our work will be subject to ongoing formal regulatory control, through the issue of a series of environmental permits, as described in the GRA [1] and discussed in Section 2.1.2. **We will update our site assessments during this period, and will discuss with the environmental regulators any need for periodic updates of our ESC during Stage 5.**

  Once detailed site-specific data have been obtained, and site assessments have been developed and reviewed, the Community Siting Partnership(s) will make a recommendation to their local decision-making bodies about whether to proceed to the next stage of the Site Selection Process. The end of Stage 5 is the last opportunity for a Community Siting Partnership to withdraw from the MRWS Site Selection Process. The local decision-making bodies will decide whether they wish to proceed further, and the UK Government will then make an informed decision on a preferred site. Assuming the local decision-making bodies have decided to proceed further, we would make an application to the environmental regulators to revise our environment permit to commence initial underground investigations at the preferred site, and would also need to apply for planning permission for such work.

- ***Stage 6: Underground operations*.** We will undertake underground construction work and investigations at the preferred site. The aims of the initial phase of underground work will be to confirm the site's suitability to host a GDF that complies with safety and environmental regulatory requirements, and to provide additional information for the final stages of detailed design. We will introduce a new set of investigation techniques suitable for use in the underground facilities, although some surface-based work will continue. **After gathering sufficient additional information, we will update the ESC to feed into a regulatory decision on GDF construction**. If regulatory requirements are met, the regulators will permit further underground operations, including construction of waste disposal areas and all required waste handling facilities (at the surface and underground). We will continue to develop our safety case, in consultation with the regulators and Community Siting Partnership, throughout Stage 6, and **we will update the ESC when we are ready to begin accepting waste at a GDF**.

If at any stage in the development process for a GDF, an issue arises that may significantly impact the ESC, we will discuss the means of resolving the issue with regulators and the Community Siting Partnership.

Although we have been discussing the sequence of updates to the ESC, the ESC forms part of an integrated DSSC, and the TSC and OSC will be updated at the same time, if appropriate.

2.1.2 Timing of ESC updates with regard to the environmental permitting process

The GRA [1] sets out the expectations of the environment agencies in terms of the submission of an updated ESC at key points in the lifecycle of the development of a GDF. These are also illustrated in Figure 2.1 and are discussed below and in Section 2.2. Consistent with the GRA, we anticipate that an initial environmental permit will be required before proceeding with the Stage 5 “intrusive” investigations (e.g. boreholes), and an “initial site evaluation” would be expected at this time. We will discuss the timing, nature and content of any application with the relevant environment agencies. Therefore, our applications for an environmental permit for surface-based intrusive site investigations and for initial underground operations may be supported by an update to our ESC. We are likely to seek approval from the environment agencies, via the issue of an environment permit, prior to committing a significant amount of money and time to evaluating one or more candidate sites.

The initial site evaluation will be followed by a “preliminary environmental safety evaluation” before the start of Stage 6 work. The GRA identifies that the first or “initial environmental safety case” would be produced during Stage 6, after the first phase of underground investigations. It is only at that time that the environmental regulators would be expecting us to have fully met all of the main requirements of the GRA, and hence would be prepared to grant an environmental permit for “disposal in principle” [1].

Although the GRA does not formally require an ESC prior to Stage 6, there are other reasons for developing one earlier, even if the ESC cannot yet fully address the GRA. In particular, progressive development of the ESC from the beginning of the process provides us with a management tool to help us develop work programmes focused on those areas most important to building confidence in the safety of a GDF and those issues raised by the regulators and Community Siting Partnerships. An ESC is also important in helping to demonstrate our capability to become a Site Licence Company in due course, and in our ongoing role of providing disposability assessments to waste producers.

Assuming we receive approval for disposal in principle, we expect to begin construction of the disposal areas and complete construction of all necessary waste handling facilities. Then, when we are ready to begin waste emplacement operations at the site, we expect to prepare a “pre-operational ESC” to seek final approval for disposal.

Once we commence waste emplacement operations, we expect to produce updates of the ESC at a frequency to be agreed with the environmental regulators (10 yearly is normal practice for periodic safety reviews for operating nuclear plant), to take account of operational experience and to consider the potential impacts of any new proposals for waste packaging or GDF design modifications. We expect to produce a closure ESC when we have completed disposal operations and a decision has been taken to seal and close the facility. We anticipate a final ESC would be produced after closure of the facility, when we are ready to request that our environmental permit is surrendered.

We have discussed the timing of updates to the ESC here. In the next section, we discuss the wider purposes of the ESC both generally and with reference to the main development stages.

2.2 Objectives of the ESC

GRA Requirement R3: Environmental safety case[4]

An application under RSA 93[5] relating to a proposed disposal of solid radioactive waste should be supported by an environmental safety case.

We will prepare an ESC in support of any application for an environmental permit to dispose of radioactive wastes under EPR 2010 [2]. The GRA defines an ESC as:

> *"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."*

This definition is consistent with international guidance on the preparation and content of a safety case for radioactive waste disposal facilities, including guidance from the IAEA [7] and the NEA [8].

We note that the ESC needs to consider environmental safety both during the period when an environmental permit is held, and in the long term, after a GDF is closed. This generic ESC contains more detail on the post-closure period because this is where the greatest challenges lie in demonstrating compliance with regulatory guidance and, therefore, where the most work is required. Evaluation of impacts on people and the environment from a GDF during the operational phase is based on similar techniques as those used in assessing such impacts from other operating nuclear plant (e.g. waste stores). Clearly there cannot be equivalent experience of a purpose-built underground disposal facility to guide safety assessments over the million-year timescale typically considered for the post-closure period of a GDF – although there is significant experience available from the previous work of Nirex and from GDF implementation programmes in other countries.

Having said that, we note that the waste would be at its most hazardous during the first few hundreds of years – during operation of a GDF and in the immediate post-closure period – because that is when the activity of the waste is at its highest (see Figure 4.1). Environmental safety during this period is a priority of our design strategy (see Section 3.1.1, Figure 3.2).

In this Section, we describe the objectives of the ESC as it evolves with time to meet the requirements of the environmental regulators as set out in the GRA [1], first from a general viewpoint and specifically with regard to the generic ESC (Section 2.2.1), and then at later development stages (Section 2.2.2).

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[4] Boxed text throughout this document reproduces the top-level requirements for authorisation of a GDF from the GRA [1] at the start of the sections that are most relevant to demonstrating how they are met by the ESC.

[5] RSA 93 is now superseded by EPR 2010 in England and Wales.

2.2.1 Using the ESC to demonstrate compliance with the GRA

The “required standard of environmental safety” is set out in the GRA [1]. The GRA sets out the fundamental protection objective, as reproduced in Section 1 of this document, and a set of five principles for solid radioactive waste disposal (Figure 2.2). It then provides a series of top-level requirements: Chapter 5 of the GRA sets out requirements on process and Chapter 6 sets out management, radiological and technical requirements. Each of the top-level requirements in the GRA is underlain by other more detailed requirements set out in supporting paragraphs. Although the GRA represents regulatory guidance and is not mandatory, the term “requirement” is used in the GRA to emphasise items that are particularly important from the regulatory perspective and where there is a strong expectation that they will be met. By meeting the requirements in the GRA, a GDF will comply with the GRA principles and, therefore, national and international waste management policy and principles (see Section 2.5, Appendix D).

As discussed in Section 2.1 and illustrated in Figure 2.1, we are currently a long way from making an application to the environmental regulators to dispose of wastes. In this generic ESC, we explain why we consider we would be able to make an environmental safety case for a GDF, based on our understanding of the scientific and engineering principles underpinning geological disposal. We do not yet have a site or disposal concept for a UK GDF. For this generic ESC, our discussion of environmental safety is therefore based on consideration of selected examples of different disposal concepts that have been developed around the world for a variety of geological environments and for a range of waste types similar to those we have to consider in the UK. The application to the UK of these illustrative geological disposal concept examples is discussed in Appendices A, B and C.

The main objectives of this generic ESC, set out in Section 1.1, are consistent with the general guidance on an ESC in Chapter 7 of the GRA, relevant to the early stages of the MRWS Site Selection Process. Our focus in this generic ESC is on the strategy for meeting the top-level GRA requirements and, with the exception of GRA Requirements R1, R2 and R3, these are shown in a text box at the top of the relevant part of Section 3. Requirements R1, R2 and R3 are considered in Section 2. In future site-specific updates of the ESC, we expect the focus will move to demonstration of how the GRA requirements are met, rather than mainly a strategic discussion of methodology. Table 2.1 sets out where in this document we discuss each of the GRA requirements. In future updates of the ESC, we also expect the detailed requirements in supporting paragraphs of the GRA would also be explicitly considered (see Appendix E).

Figure 2.2 Relationship between the fundamental protection objective, principles and top-level requirements for solid radioactive waste disposal in the GRA

Figure based on the GRA [1, Figure 3.1]. Table 2.1 sets out where in the generic ESC we address each of these requirements. Note that references in this document to the "period of authorisation" should be considered to be the same as the period for which an environmental permit is held.

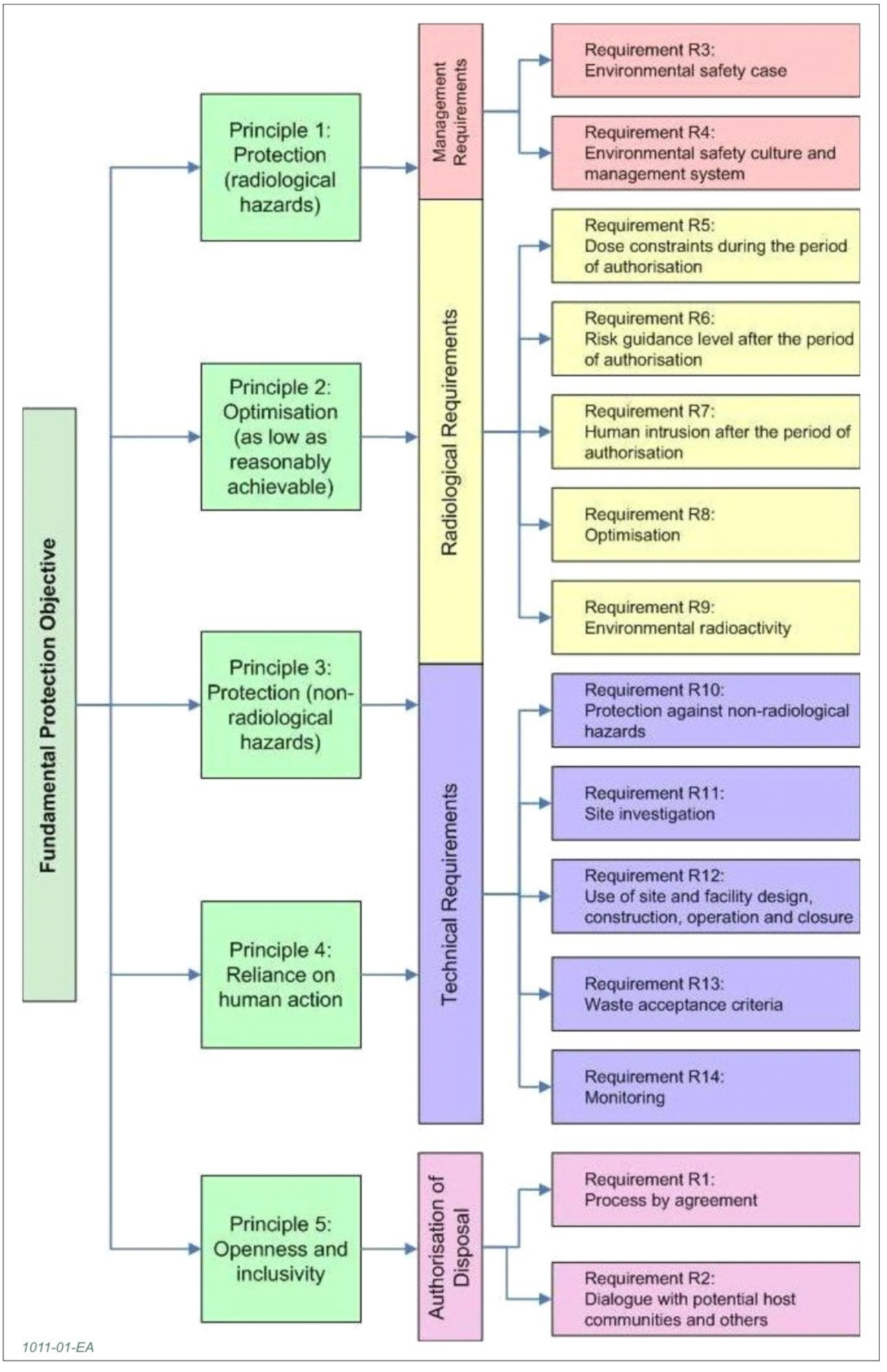

Table 2.1 Section(s) in this report where each high-level requirement in the GRA is discussed

As this report is following the international guidance on structuring of an ESC, the GRA requirements are discussed within the logical order of presentation in this report, not the order in which they are set out in the GRA. Bold type indicates those requirements we consider to be met in this generic ESC. For the majority of the requirements, full demonstration that they have been met will only be possible at later stages in the development of the ESC.

No.	GRA requirement title	Approach to meeting requirement	Demonstration of meeting requirement
R1	**Process by agreement**	**Section 2.3**	**Section 2.3**
R2	Dialogue with potential host communities and others	Section 2.4	Section 2.4 Section 3.3.3
R3	Environmental Safety Case	Section 2.1 Section 2.2	Entire document
R4	**Environmental safety culture and management system**	**Section 3.3**	**Section 3.3**
R5	Dose constraint during the period of authorisation	Section 3.2.1	Section 5.1
R6	Risk guidance level after the period of authorisation	Section 3.2.2	Section 5.2
R7	Human intrusion after the period of authorisation	Section 3.2.3	Section 5.2.2.3
R8	Optimisation	Section 3.1.2	Section 4.1.4
R9	Environmental radioactivity	Section 3.2.4	Section 3.2.4
R10	Protection against non-radiological hazards	Section 3.2.5	Section 3.2.5
R11	Site investigation	Section 3.1.5	Section 4.1.3
R12	Use of site and facility design, construction, operation and closure	Section 3.1.3	Section 4.1.5
R13	Waste acceptance criteria	Section 3.1.4	Section 4.1.2
R14	Monitoring	Section 3.1.6	Section 4.2.2

2.2.2 Objectives of the ESC at later development stages

The UK Government will decide the candidate sites where we will begin surface-based investigations in Stage 5, based in part on the desk-based site assessment studies we conduct during Stage 4. We will update these assessments to feed into a decision to initiate surface-based intrusive investigations (e.g. boreholes) at one or more sites. This is termed the “initial site evaluation” in the GRA (Figure 2.1). The updated assessments for surface-based intrusive site investigations would contain site-specific disposal concepts and assessments based on the results of desk-based information gathering and any initial surface-based non-intrusive site investigations (e.g. surface geophysics). The forward programme leading to these assessments will become more detailed and will link clearly to site-specific R&D, site characterisation and design requirements.

By early in Stage 5 there would be several new objectives for any work related to developing the ESC. In particular, we would wish to:

- provide largely qualitative views on the feasibility of constructing a GDF at the candidate site(s);
- demonstrate how a GDF at the candidate site(s) could meet the principles and requirements of the GRA;

- explain how we intend to continue characterising the candidate site(s); and
- indicate how we would go about continuing to develop the ESC for a GDF.

We would also undertake limited quantitative assessment based on available site knowledge and data. An important aim would be to demonstrate that any proposed surface-based intrusive site investigations would not compromise the integrity of a candidate site to the unacceptable detriment of the environmental safety of a GDF.

At the conclusion of Stage 5 and before continuing to Stage 6 (underground operations), another update of the siting assessments would be prepared to justify a decision on underground operations at a single site. This is termed the “preliminary environmental safety evaluation” in the GRA. This update would include increasingly more detailed quantitative assessment based on available site knowledge and data and GDF designs. This update to the assessments would also need to demonstrate that underground operations would not compromise the integrity of a candidate site to the unacceptable detriment of the environmental safety of a possible GDF.

As underground investigations/operations progress and increase in scale during Stage 6, an “initial environmental safety case” would be expected by the environment agencies, developed to the degree necessary to inform a regulatory decision on whether an environmental permit for disposal in principle could be granted.

At a final hold point before waste is placed in a GDF, a “pre-operational environmental safety case” is expected by the regulators. This update to the ESC would be based on a single site, design and intended inventory, taking account of knowledge and understanding gained during underground investigation and the initial phase of construction[6], and demonstrating that a GDF meets the requirements of the GRA. The pre-operational environmental safety case would provide a basis for an environmental permit to allow waste disposal to start.

As the ESC develops beyond the generic stage towards supporting an application for an environmental permit for disposal of radioactive waste at a specific site, it would also need to:

- set out a site-specific and design-specific environmental safety strategy, i.e. the top-level description of the fundamental approach to be taken to demonstrate environmental safety of the system;
- demonstrate a clear understanding of the disposal facility in its geological setting, how the geological disposal system will evolve, and how its various components contribute to meeting the requirement of providing a safe long-term solution for the UK’s higher activity radioactive wastes; and
- describe the key environmental safety arguments and the underpinning lines of reasoning and detailed analysis, assessments and supporting evidence.

A site-specific ESC in support of such an application would be a substantial submission and would need to include detailed information of the following types:

- the geology, hydrogeology, geochemistry, geotechnical characteristics and surface environment of the chosen site and its setting;
- the characteristics of the waste including its radionuclide and materials content, treatment and packaging;

[6] Note that after initial construction of the GDF sufficient to allow disposals to commence, further construction of new disposal areas would proceed in parallel to disposal in existing vaults and tunnels.

- the design and layout of the disposal facility, including the design of the engineered barriers, and how it will be constructed, operated and closed;
- the basis for, and output from, computer-based models of the performance of the disposal system and its components; and
- semi-quantitative and qualitative supporting evidence that builds confidence in our claims for environmental safety.

As the ESC is progressively updated, it will also need to include further information on how we manage safety and how key decisions have been made, including:

- a description of our technical and management system to ensure that a GDF would be constructed, operated and closed as required;
- explanation of how the supporting work programme, including site characterisation and R&D activities, has been prioritised;
- how the disposal facility design has been developed and optimised - for example, how choices between design options for a specific site were made; and
- how uncertainties in our planning assumptions and uncertainties specific to our understanding of candidate site(s) have been and are being managed.

At this generic stage of the MRWS Site Selection Process, it is not possible to include detailed information of this type.

2.3 Regulatory context

GRA Requirement R1: Process by agreement

The developer should follow a process by agreement for developing a disposal facility for solid radioactive waste.

In this section we set out our understanding of GRA Requirement R1 and explain why we consider this requirement to be met (Section 2.3.1). We also provide selected examples of regulatory scrutiny work that we have considered in developing the generic ESC (Section 2.3.2).

2.3.1 Compliance with GRA Requirement R1

The developer of a radioactive waste disposal facility must hold an environmental permit under EPR 2010 [2] or an authorisation under RSA 93 [3] from the relevant environment agency before any disposal of radioactive waste can take place. However, a significant amount of resources in both time and money will be expended by a developer of a GDF before the facility is ready to accept waste. This is recognised in the GRA, and the environment agencies expect a developer to enter into an agreement by which the agencies can charge the developer to provide advice and assistance after a decision has been made to start a process to select a site for a GDF. This arrangement is termed a “process by agreement” in the GRA, and is a top-level requirement for the regulatory approval process set out in the GRA.

It is highly likely that the relevant environmental regulator for a GDF would be the Environment Agency[7], and we already have an ongoing formal arrangement with the Environment Agency to facilitate scrutiny of our work. Therefore, we are already in a process by agreement and we believe that we fulfil GRA Requirement R1. This generic ESC will be scrutinised by the Environment Agency as part of that agreement. A summary

[7] The Environment Agency is the environmental regulator in England and Wales.

of the Environment Agency's ongoing scrutiny work leading up to publication of this generic ESC is contained in [34], and selected examples are discussed in Section 2.3.2.

Ongoing early dialogue between ourselves and the environmental regulator will help build mutual confidence in the regulatory process. It will help both us and the environmental regulator to identify any potential environmental safety issues that require resolution before there is a significant investment of time and money. It will also allow us to develop a strategy for addressing possible environmental safety concerns before proceeding with an application under EPR 2010 [2]. Early dialogue also increases understanding of our proposals so that the environmental regulator can make informed comments to the planning authority during applications under the land-use planning process. Publishing the record of the dialogue will facilitate open discussion with stakeholders of regulatory views and our views.

The UK Government has a general responsibility to ensure that the regulatory framework for a GDF is adapted to the MRWS Site Selection Process. The UK Government has recently amended the legislative powers to enable a staged environmental permitting process for a GDF by taking radioactive substances regulation into the Environmental Permitting Regulations [2][8]. This is illustrated in the right-hand side of Figure 2.1, which shows that staged environmental permitting would involve regulatory control of the development from the start of "intrusive" site investigation onwards (i.e. at some point during Stage 5 of the MRWS Site Selection Process), with the process by agreement covering dialogue until this hold point. Submissions to the environmental regulator and corresponding submissions under the nuclear site licensing process (see Section 2.5) would then be formally required to obtain a series of regulatory permits throughout intrusive investigation, construction, operation, and closure of a GDF.

The regulatory context of the ESC will change over time, consistent with a staged development and approval process. Currently, the emphasis is on strategy and how we intend to meet requirements and demonstrate safety. This will gradually change to an emphasis on implementation of a GDF, and how we have met requirements. The detailed objectives of the generic ESC and future ESCs are set out in Section 1.1 and Section 2.2.

2.3.2 Selected examples of regulatory scrutiny

The Environment Agency has been involved in regulatory scrutiny of our work since the formation of RWMD and, previously, it scrutinised the work of Nirex. A wide range of regulatory scrutiny interactions and reports has been produced, as summarised most recently in [34]. Selected examples include reviews of the following reports or topic areas:

- The GDF viability assessment conducted by Nirex [35] – see Appendix F.
- Context for post-closure generic performance assessment (GPA) [36] – see Appendix F and our response to the review in [37].
- Near-field processes [38]. The recommendations in this review back up those provided in the Environment Agency's review of the viability assessment and are therefore also essentially covered in Appendix F of this report. Although the Environment Agency's recommendations are not specifically highlighted in the Near-field evolution status report [16], they were considered in developing that report.
- The longevity of ILW packages for geological disposal [39]. The Environment Agency's recommendations have been considered in developing the Package evolution status report [15] and our response to the review is contained in [40].

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[8] Note that the Environmental Permitting Regulations only apply to England and Wales.

- Gas generation and migration from a GDF [41]. The Environment Agency's recommendations have been considered in developing the Gas status report [19], and our response to the review is contained in [42].

The first two reviews are the most relevant to this generic ESC. In the interest of demonstrating a connection between past and current work on UK GDF development and regulatory dialogue, we set out in Appendix F in some detail the issues identified in the viability assessment work and the Environment Agency's review of that work, and the issues identified in the Environment Agency's review of the GPA and our responses to them. The regulatory review of previous work by Nirex forms part of the regulatory context for the generic ESC.

The Environment Agency also reviewed a storyboard we developed in 2009 to set out our intentions for this Generic ESC main report [43]. The storyboard was a ~40-page document that outlined in some detail the proposed structure of this report, and contained brief statements about the likely content of each section. We also summarise in Appendix F the Environment Agency's comments on that storyboard and our responses to those comments and/or where they have been considered in this report. The regulatory review of this storyboard also forms part of the regulatory context for this report.

2.4 Wider dialogue on the ESC

> ***GRA Requirement R2: Dialogue with potential host communities and others***
>
> **The developer should engage in dialogue with the planning authority, potential host community, other interested parties and the general public on its developing environmental safety case**.

As the developer of a GDF, we will engage in dialogue regarding the ESC with the identified potential host communities, their Community Siting Partnerships, the relevant local planning authority, other interested parties, and the general public. We regard such dialogue as an essential element of the step-wise process for the successful development of a GDF.

As discussed in Section 1.1, a key reason we are producing this generic ESC at an early stage, prior to the identification of any sites, is to provide information to all stakeholders on our proposals for the structure of an ESC and to seek their feedback on our strategy so that highlighted issues can be addressed in future updates of the ESC.

Wider dialogue on the ESC - and the DSSC overall - is supported by the production of a DSSC overview report [44] written in non-technical language. We have also prepared a more publicly accessible version of our generic design report [25], and note that the Disposal System Functional Specification [22] has been prepared to be accessible to wider audiences that the more detailed Disposal System Technical Specification [22]. These reports are supported by presentation of all the ESC and DSSC documents as much as possible in 'plain English', while recognising that informed regulators are an important audience for these reports, and that some of the arguments we need to make are of a specialist technical nature.

Future updates of the ESC will be supported by more extensive programmes of dialogue with potential host communities, once they are participating in the UK Government's MRWS Site Selection Process. We expect to take an active role in face-to-face dialogue with the Community Siting Partnership(s), once these have formed.

However, we recognise that there is a wide range of other interested parties with differing requirements (e.g. nuclear industry, academia, non-governmental organisations, politicians, media, schools, etc.), and we will prepare other materials and engage in dialogue as necessary to meet the needs of these other stakeholders. For dialogue on the ESC to be effective, we will need to ensure that we understand the needs and interests of all our stakeholders. We have conducted research to understand the needs and interests of wider audiences and how best to communicate safety issues (e.g. [45]). We can use the generic ESC as a means of furthering such research.

The dialogue processes we will use are still under consideration; however, a variety of processes will be used to encourage a broad level of involvement [46]. These could involve our staff undertaking the following kinds of activity:

- preparing further written documents, presentations and posters on specific safety issues;
- using electronic communication tools (e.g. via the internet) to present safety information; and
- participating in various forms of public meetings (e.g. lectures, workshops, drop-in sessions), depending on the issue(s) to be considered and the stakeholder group(s) involved.

Dialogue will be important not only prior to the key decisions that need to be taken during the MRWS Site Selection Process (Section 2.1), but between these points to provide confidence that we are listening to what others have to say and responding appropriately, and to build confidence that our safety evaluations are being undertaken in an unbiased manner.

It will not be possible to engage in dialogue on every decision we have to make and, even where dialogue occurs, we realise that not all stakeholders will necessarily have the same views. Where there are disagreements between stakeholders, our aim will be to satisfy the needs of regulators and the Community Siting Partnership(s), while carefully explaining to others the basis behind our decisions.

We will also respond to comments we receive from stakeholders on the ESC. Finally, if at any stage in the development process for a GDF, an issue arises that may significantly impact the ESC, we will discuss the means of resolving the issue with the Community Siting Partnership and others.

Further information on our public and stakeholder engagement and communications strategy and how it was developed is contained in [46].

2.5 Related international and national environmental legislation, guidance and obligations

International treaty obligations affecting UK radioactive waste management policy include obligations stemming from membership of the European Union [47], obligations under the 1993 Oslo and Paris Convention on the Protection of the Marine Environment of the North East Atlantic (OSPAR Convention) [48], and obligations under the IAEA-sponsored Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management [49]. Policy and regulation in the UK also takes account of guidance from the IAEA [50] and the International Commission on Radiological Protection (ICRP) [51]. The UK Government’s 2008 MRWS White Paper [4] and the GRA [1] are consistent with these international treaties and guidance. Therefore, compliance with the GRA ensures compliance with existing international expectations on the environmental safety of a GDF. A summary of international obligations and guidance relevant to the environmental safety of a GDF is provided in Appendix D.

In addition to EPR 2010 [2] and the GRA [1], development of a GDF in England or Wales must comply with a range of other legislation relevant to environmental safety and the ESC [52], and in particular:

- Strategic Environmental Assessment (SEA) under European Directive 2001/42/EC [53];
- land-use planning and Environmental Impact Assessment (EIA) under the Town and Country Planning Act 1990 in England and Wales [54];
- the 2000 European Water Framework Directive [55] and 2006 Groundwater Directive [56];
- the Health and Safety at Work Act 1974 [57] and the Nuclear Installations Act 1965 [58] – see below;
- the Nuclear Industries Security Regulations 2003 [59];
- Carriage of Dangerous Goods and Use of Transportable Pressure Equipment Regulations 2009 [60]; and
- Regulations for the Safe Transport of Radioactive Material 2005 [61] – see below.

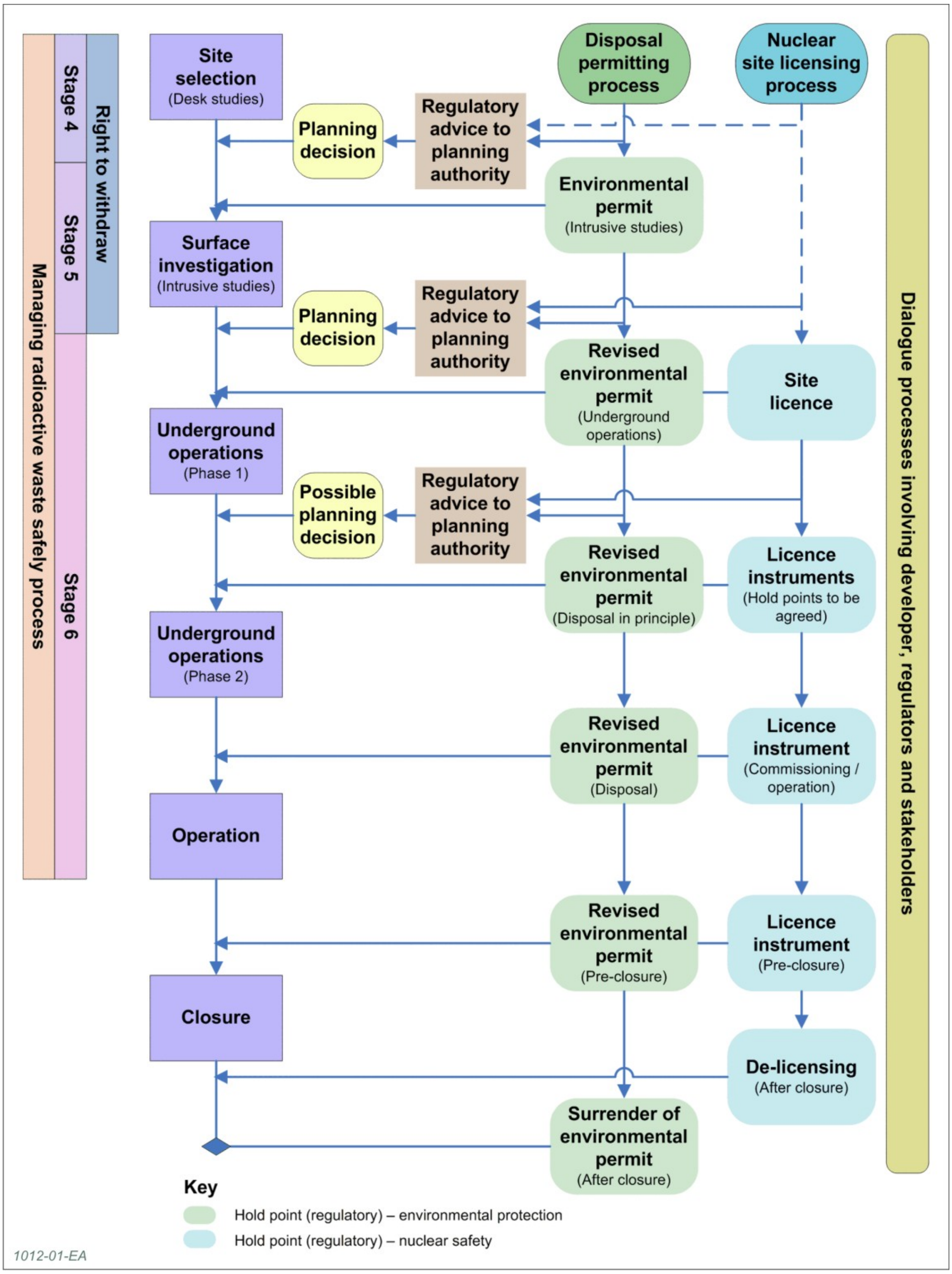
A summary of extant legislation and how this is currently expected to impact the ESC is also provided in Appendix D.

A GDF would require a site licence under the Nuclear Installations Act 1965, issued by the Nuclear Directorate of the Health and Safety Executive (HSE). The Nuclear Directorate has published “Safety Assessment Principles for Nuclear Facilities” [62] that applies to its assessment of safety cases for nuclear facilities and covers nuclear safety and radioactive waste management. We would prepare the regulatory submission required for a nuclear site licence in a parallel stream to the preparation of the ESC. The Operational Safety Case (OSC) represents the second key component of the DSSC. The relationships between staged environmental permitting under EPR 2010, as shown in Figure 2.1, the submissions to HSE for a site licence, and submissions for the planning process are illustrated in Figure 2.3 [1].

Regulation of transport of waste to a GDF (from outside the boundaries of the nuclear licensed site) is undertaken by the Department for Transport. We would prepare the regulatory submission required for waste transport in a parallel stream to preparation of the ESC. The Transport Safety Case (TSC) represents the third key component of the DSSC.

Figure 2.3 Indicative links between the staged environmental permitting process under EPR 2010, nuclear site licensing, and the planning process

Figure from the GRA [1, Figure 5.3].



3 Our safety strategy

As stated in the GRA [1, paragraph 4.21] and set out in Section 1, the fundamental protection objective is "…*to ensure that all disposals of solid radioactive waste to facilities on land are made in a way that protects the health and interests of people and the integrity of the environment, at the time of disposal and in the future, inspires public confidence and takes account of costs*." Therefore, in the context of the GRA, a safety strategy is an approach or course of action designed to achieve and demonstrate the safety of people and the environment both at the time of disposal and in the future. This section sets out our safety strategy and our approach within the strategy to meeting each of the specific technical requirements in the GRA.

Our safety strategy is developed from national and overseas experience of developing safety assessments, our knowledge of safety cases for GDFs in other countries, working with international (EC, NEA, IAEA) safety case groups, and lessons learned from previous ESCs developed in the UK, e.g. for the national low-level waste repository (LLWR) near the village of Drigg in West Cumbria [63], and for the proposed Dounreay low-level waste disposal facility [64].

Two constraints on our safety strategy are the geological environments of the potential host communities that come forward as part of the MRWS Site Selection Process, and the inventory for disposal. The latter is defined in the UK Government's 2008 MRWS White Paper [4] as higher activity radioactive wastes that cannot be managed under the "Policy for the Long-term Management of Solid Low Level Radioactive Waste in the United Kingdom" [65] or are not managed under the Scottish Executive's policy for higher activity radioactive wastes. This 'Baseline Inventory' is associated with activities primarily related to the generation of nuclear power and includes 'legacy' materials from existing nuclear facilities. We also consider the implications of disposing of an 'upper inventory', which includes additional wastes that might be generated in the future from a possible programme of new nuclear power stations. The different types of material that make up the potential inventory of higher activity radioactive waste are discussed in Section 4.1.

The UK Government sees no case for having separate disposal facilities for different types of higher activity radioactive wastes if one facility can be developed to provide suitable, safe containment for the Baseline Inventory. This is because the sharing of surface facilities, access tunnels, construction support and security provision could lead to significant benefits, including major cost savings and lower environmental impacts [4]. There is no reason why a single GDF should not be technically possible, in theory, although the final decision would be made in the light of the latest technical and scientific information, international best practice and site-specific environmental, safety and security assessments at candidate sites.

Our safety strategy consists of a design and siting strategy, an assessment strategy, and a management strategy:

- Within the site offered by a particular host community and for our preferred disposal concept, we would use the ESC to assist in the siting, layout, operation and closure planning of a GDF. Disposal facility design would consider the inventory we are required to manage in a GDF, and would follow international good practice and the GRA in providing for passive safety and using the safety functions of multiple barriers to provide safety. Our design and siting strategy is presented in Section 3.1. We present only a high-level summary of our strategy here in mainly generic terms; it will be developed further once we have specific candidate sites to consider.

- Our assessment strategy follows international good practice and the requirements of the GRA. Our assessment strategy is presented in Section 3.2. Some components of it are still under development, and will benefit from dialogue with regulators to better understand their expectations. Many components of our assessment strategy have not been implemented in this generic ESC because we consider it premature to do so until we have sufficiently detailed information from specific candidate sites and have developed site-specific disposal concepts. However, we have implemented those parts needed to provide confidence in our ongoing assessment of waste packaging proposals by waste producers, to demonstrate GDF viability, and to inform our initial desk-based assessments of candidate sites once these sites are available.
- An overall management strategy is needed to provide confidence that we can deliver the disposal system specification and the design and assessment strategies in a coherent, integrated way and with appropriate quality and management accountability over the long timescales of GDF planning and delivery. We have developed a Safety and Environmental Management Prospectus [66] that sets out our management strategy and safety procedures for delivering a GDF. A summary is included in Section 3.3. The key elements of our management strategy needed for the near future are already in place, and have, for example, influenced the content of this generic ESC and controlled its production. However, our management strategy will need to develop in the future to meet the needs of the programme as it evolves (e.g. to control site characterisation and eventual GDF construction, operation and closure).

3.1 Design and siting strategy

3.1.1 Safety concept

Concentrating and containing solid radioactive waste, and isolating it from the biosphere, is the internationally accepted strategy for the safe long-term management of such materials. In geological disposal, long-term containment and isolation of solid radioactive waste is provided by its emplacement in a facility located underground in a stable geological formation – **the underground facilities and the geological environment** comprise the **geological disposal system**. A GDF includes **surface facilities** during the period in which an environmental permit is held, to support activities such as waste receipt and operations in the underground facilities (construction, waste emplacement, engineered barrier emplacement) and monitoring.

A distinctive feature of geological disposal is the depth of emplacement, 200 -1,000 metres below ground. The depth chosen for disposal in a particular facility, as well as specific elements of its design, will depend on a number of factors including, but not limited to, groundwater conditions, rock stability, host rock composition and the nature of the waste.

This section explains the timeframes we use to set out our considerations of safety in this generic ESC (Section 3.1.1.1); the importance of passive safety as a design principle (Section 3.1.1.2); and the high-level safety functions that can be provided by different components of the geological disposal system (Section 3.1.1.3). A **safety function** is a physical or chemical property or process that contributes to safety, by isolating or containing the disposed waste. The barriers that make up the geological disposal system may contribute to satisfying a number of different safety functions and these contributions may be independent of one another. Our **safety concept** is the set of specific safety functions provided by the natural and engineered barriers in a geological disposal system for a specific site and design.

3.1.1.1 Safety timeframes

In the context of demonstrating environmental safety, we identify three periods associated with the development of a GDF:

- The **pre-operational period** includes concept definition, site investigation and construction of initial parts of a GDF, essentially the period defined by the MRWS Site Selection Process. During the pre-operational period, site assessment, design studies and environmental impact assessments will be carried out, together with the development of those aspects of the ESC for operational environmental safety and post-closure safety required to obtain the environmental permit to proceed with the construction of a GDF. Sufficient construction will then be undertaken during the pre-operational period to bring a GDF to a point at which disposal operations could safely commence.
- The **operational period** begins when waste is first received at the facility. From this time, radiation exposures may occur as a result of waste management activities and these must be controlled according to radiological protection and safety requirements. In this period, monitoring and testing programmes will continue, and the post-closure aspects of the ESC will be refined based on any additional data collected. During the operational period, further construction activities will take place at the same time as waste emplacement and, possibly, closure of parts of the underground facilities that have been filled with waste. This period could include a phase when disposal operations have been completed, but the underground facilities are left open for monitoring of the performance of a GDF. The operational period ends with the sealing and final closure of the underground facilities and the decommissioning of the surface facilities.
- The **post-closure period** begins at the time when waste acceptance and handling operations are concluded, and the disposal areas and all shafts, access routes from the surface, as well as access to the waste packages have been backfilled and sealed. We discuss the post-closure period in this generic ESC in terms of an early post-closure period and a late post-closure period (see Section 4.2.1 and Appendices A, B and C). After closure, the safety of a GDF will be provided by passive means inherent in the site and facility design and waste package characteristics. However, active institutional controls, including some monitoring, may continue for some time after closure, e.g. for the purpose of meeting nuclear safeguards agreements or for the purpose of providing public reassurance.

When we use the term GDF, we generally mean both the surface and underground facilities during the pre-operational and operational periods, but only the underground facilities during the post-closure period.

3.1.1.2 Designing for passive safety

The design of a GDF would incorporate the principle of **passive safety** to the extent possible. Passive safety can be defined slightly differently in different contexts. In the context of environmental safety regulation and the GRA [1], passive safety means that there is no reliance on active safety systems or human intervention to provide safety.

During the operational period, passive safety design measures (e.g. the integrity of waste containers and the stability of wasteforms) would be relied on as much as possible, but there would be a need to rely on some active measures, e.g. forced ventilation and filtration systems at the discharge point. In addition, the surface facilities would require systems for control of potentially contaminated gaseous and liquid discharges. There are international precedents for operational safety being readily achievable in a GDF, e.g. the US Waste Isolation Pilot Plant (WIPP), providing confidence that we can design and deliver the required level of operational safety.

A GDF would be designed to rely only on passive safety once the facility is closed and sealed. Closure brings to an end the need for active management of the underground facilities. This means no ongoing inspection and maintenance would be required in order to assure continued safety performance. Thus, it is confidence in the passive operation of the natural and engineered barriers that provides assurance of safety in the post-closure period, although active controls may be maintained for some time at the surface to prevent access and inadvertent disturbance of the underground facilities.

To build confidence in the implementation of our safety concept, our design process currently takes account of the designs of GDFs in other countries. Building confidence in the long-term behaviour of materials under physical, chemical and thermal conditions similar to those that are expected in a GDF is a key focus of our R&D and engineering programmes.

3.1.1.3 Barrier components and safety functions

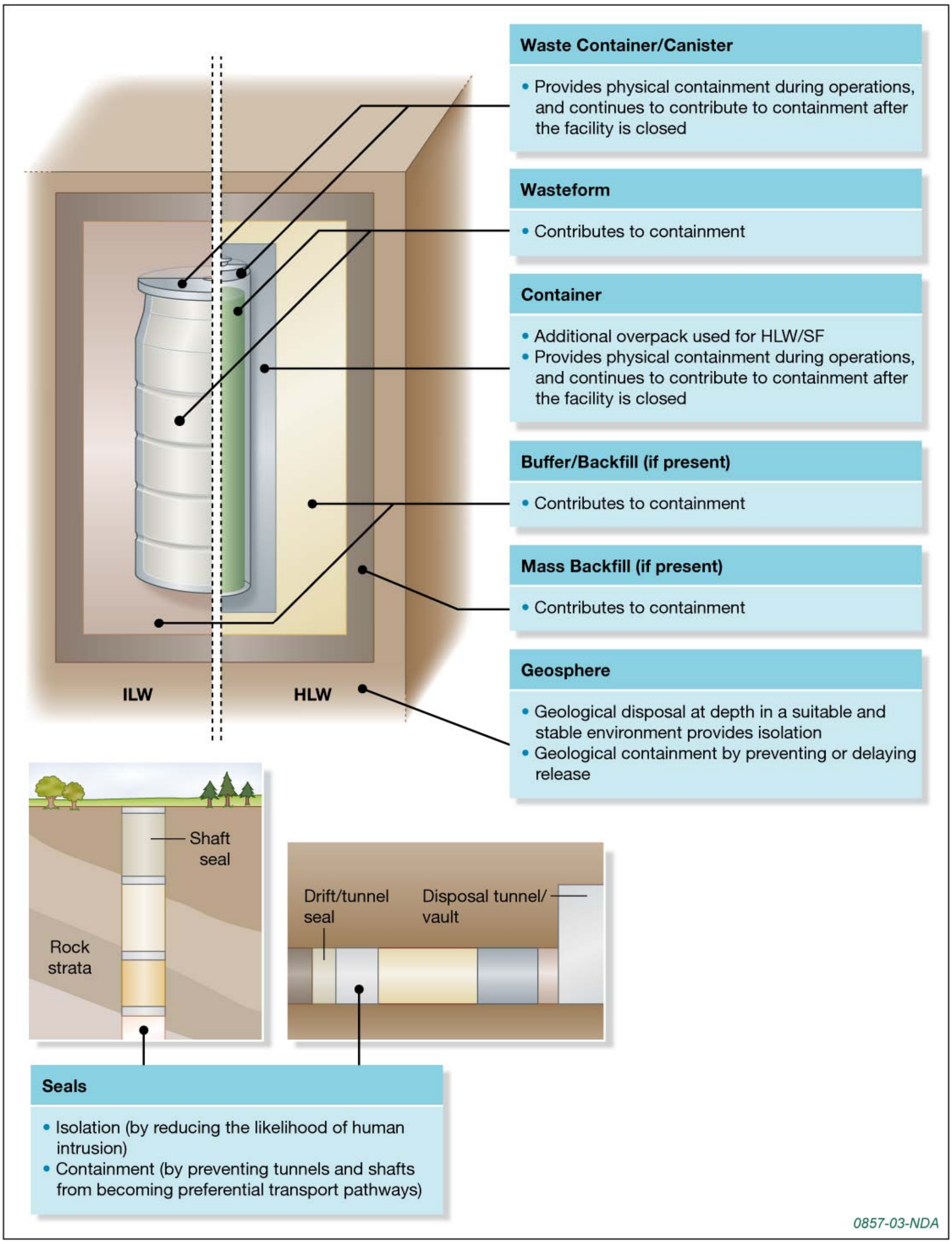
Radioactive decay progressively reduces the quantities of radionuclides present in a geological disposal system. The wastes are most hazardous in the operational and early post-closure periods and it is at these times that the waste requires the greatest degree of isolation and containment. As illustrated in Section 4.2.1, the characteristics of the Baseline Inventory are such that significant decay occurs on a timescale of several hundred years, although protection continues to be required for very much longer than that.

The conceptual basis of geological disposal has been firmly established internationally for the last 40 years as being based upon the **multi-barrier system**, whereby a series of engineered and natural barriers act in concert to isolate the wastes from the surface environment, and to contain the radionuclides associated with the wastes. The relative importance of the various barriers at different times after closure of a GDF and the way that they interact with each other depend upon the design of a GDF, which depends on the geological environment in which the facility is to be constructed, as well as the type of waste under consideration. Consequently, the multi-barrier system can work in different ways for different disposal concepts.

The barriers found in a typical geological disposal system are illustrated in Figure 3.1. These are the wasteform, the waste container and overpack (where used), buffer/backfill material, mass backfill, seals, and the host rock. The ensemble of wasteform, waste container and overpack is referred to as a waste package. An overpack is an additional container used for disposal of wastes that are already packaged; for example, conditioned HLW is currently stored in stainless steel containers, and an overpack made of another material could be used to support the longevity of the waste package and the long-term safety of a GDF. The buffer/backfill generally refers to material placed adjacent to and around the waste package in a GDF. Mass backfill generally refers to material used to fill some or all of the empty space remaining in the excavated areas of a GDF, once other engineered barriers have been emplaced.

The **engineered barriers** and the **natural barrier** provided by the geological environment will work together to provide the necessary level of safety and ensure that undue reliance is not placed on any one barrier. Each of the barriers will provide specific safety functions contributing to **isolation** or **containment** over different timescales.

Figure 3.1 Barriers found in a typical geological disposal system



Isolation means removing the waste from people and the surface environment. Geological disposal at depth in a suitable environment provides isolation, therefore 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 from direct radiation.

The surface environment is often referred to as the 'biosphere'. The biosphere is generally taken to include the atmosphere and the Earth's surface, including the soil and surface water bodies, seas and oceans. The depth of disposal and the characteristics of the geological environment provide isolation of the waste from the biosphere. Long-term isolation requires adequate sealing and closure of the underground facilities at the end of the operational period.

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.

Isolation differs from containment in that isolation is about situating the underground disposal facility away from people and reducing the likelihood of inadvertent future human intrusion, and containment is about the ability of a barrier to hold the radionuclides within it. However, isolation also contributes to containment in that locating a GDF in a suitably deep and stable environment protects the engineered barriers, helping them to preserve their containment functions for longer times. In that sense, the isolation and containment functions can be regarded as complementary.

When applied to the geological environment of a GDF, the isolation function will remain intact as long as the waste in the underground facilities remains out of people's way – isolation does not represent a particular property of the geological environment. The only naturally occurring processes that could affect isolation are large-scale tectonic processes (e.g. rapid uplift and erosion)[9], processes that are not relevant to the UK.

Isolation is an inherent feature of geological disposal. The MRWS White Paper [4] indicates the depth at which the underground vaults and disposal tunnels will be located is likely to be somewhere between 200 and 1,000 metres, depending on the geological and hydrogeological characteristics at the site in question.

Location of a GDF away from known areas of underground mineral, geothermal and groundwater resources, as considered as part of the site unsuitability screening during Stage 2 of the MRWS Site Selection Process [4], reduces the likelihood of inadvertent disturbance of a GDF in the future, when the location and/or purpose of the facility may have been forgotten by society.

Containment is provided by both engineered and natural barriers. Containment within the engineered barrier system (EBS) can be provided by means of a durable wasteform, waste container, overpack (where used) and buffer/backfill that is compatible with the waste package and the host rock. The system of seals that will be emplaced in disposal areas and ventilation access points, underground tunnels for waste transport, and, finally, surface access tunnels and/or shafts also provides containment. The geosphere (the rock and groundwater between a GDF and the biosphere) supports containment in two ways: it can

[9] Such processes would obviously also affect containment.

provide physical and chemical protection to the EBS and thereby prolong containment within the EBS; and it can provide for an extremely slow mobility of radionuclides that are released through the EBS in the very long term.

Note that the containment functions and properties of a barrier differ depending on the form of the contaminant (e.g. gaseous, solid, liquid).

Once a barrier ceases to completely contain the waste, retardation processes often continue to play an important role in limiting the release of contaminants from the barrier. Retardation processes occur in both the engineered and natural barriers. They include chemical and physical processes:

- Chemical processes include solubility limitation within a GDF, and sorption and precipitation in both the engineered and natural barriers; such processes slow the rates of release and migration of contaminants. Such processes may be promoted in the EBS by incorporating barriers that chemically condition the environment within a GDF.
- Physical processes include slow flow rates in both the engineered and natural barriers, dispersion and dilution in groundwater, and matrix diffusion (movement of dissolved species into dead-end pores in the rock) in the geosphere. For some materials and environments, the permeability may be so low that all solute transport is by chemical diffusion. In some geological environments, processes of dispersion, dilution and matrix diffusion play an important role in reducing the concentrations of GDF-derived contaminants in the biosphere to very low levels.

Most of these retardation processes can contribute significantly to the initial period of containment provided by a barrier. Finally, we note that **radioactive decay** will progressively reduce the quantities of radionuclides present in the geological disposal system (see Section 4, Figure 4.1). There will be essentially complete containment for any radionuclides that are not released through the barriers before the time at which they and their daughters have decayed to negligible levels.

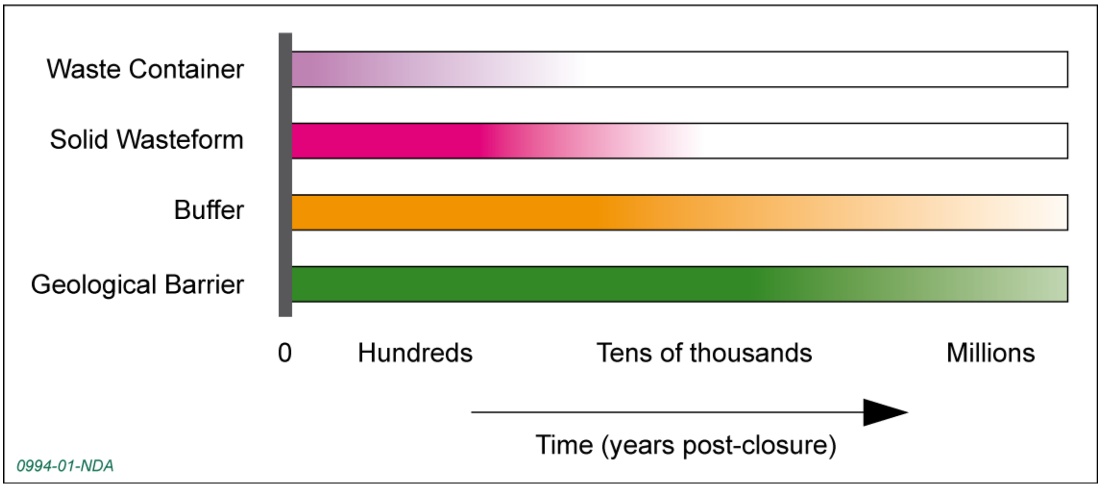
Both isolation and containment are especially important during the operational period and during the early centuries after closure of the GDF when the hazard potential of the wastes is highest. The isolation and containment capabilities of the geological disposal system and the different barriers of which it is comprised are demonstrated through safety analysis relevant to the waste type, the design of a GDF, and the site. An important objective of GDF design development and safety analysis is to provide assurance that the majority of shorter-lived radionuclides will decay *in situ* and that any releases of longer-lived species will be spread over later times so as not to give rise to significant concentrations of radionuclides in the biosphere. Much of our post-closure safety analysis work involves evaluating the fate and impact of the relatively small amounts of radionuclides that might eventually reach people and the surface environment, even though this may not happen until many thousands of years into the future.

A review of multi-barrier systems for geological disposal is provided in [28, 29], and the UK application of a set of illustrative geological disposal concept examples is discussed in Appendices A, B and C for ILW/LLW and for HLW/SF. Further discussion of the safety functions relevant to the illustrative geological disposal concept examples for ILW/LLW and for HLW/SF is provided in Section 5.2.1.1 and in Appendices A, B and C and in the Disposal System Technical Specification [23]. Our research status reports discuss in more detail the nature of particular safety functions and their role in providing safety with regard to different waste types and the illustrative geological disposal concept examples.

As noted above, the presence of multiple barriers and corresponding safety functions working in concert enhances both safety and confidence in safety by ensuring that the overall performance of the geological disposal system is not unduly dependent on a single barrier or safety function. As a hypothetical example, a long-term safety concept for geological disposal is illustrated in Figure 3.2. The greatest degree of containment needs to be provided in the first few hundreds to thousands of years, when the wastes are most hazardous. Note that we expect to develop at least two different disposal concepts, one for ILW/LLW and one for HLW, because of the different characteristics of these waste types. Disposal concepts for other materials not yet declared as wastes have also been considered, initially based on similar concepts to those for existing wastes – see Section 3.1.3.2 and Section 4.

Figure 3.2 The extent to which safety relies on the performance of different barriers for a hypothetical disposal concept

The engineered barriers degrade with time and gradually lose their containment function. In this hypothetical disposal concept, the waste container is expected to remain intact for thousands of years, the internal waste packaging materials for tens of thousands of years, and the surrounding buffer material for hundreds of thousands of years. The geological barrier isolates the disposal facility from the human environment and provides a substantial degree of containment and retardation over this entire period. The greatest degree of containment is provided in the first few hundreds to thousands of years, when the wastes are most hazardous.



Each barrier in a GDF design may perform a number of safety functions. These safety functions will depend on the properties of each barrier. Examples include:

- the function of a waste package in providing containment of radionuclides will depend on a number of factors, including the corrosion rate of the material of which the container is constructed;
- the functions of the geological environment in limiting groundwater inflow to a GDF and providing long groundwater travel times to the biosphere (i.e. providing containment of radionuclides) will depend on the permeability of the disposal horizon and the hydraulic gradient in the geologic units between a GDF and the biosphere; and
- the function of the geosphere in retarding the transport of radionuclides (i.e. contributing to containment) will depend on various geological, geochemical and hydrogeological properties of the system.

In the long term, progressive degradation of the EBS is expected and, consequently, radionuclides and other materials disposed of in a GDF may be released into the geological environment where they may eventually migrate to the biosphere. The geological disposal system will provide a combination of engineered and natural characteristics to support efficient isolation and containment by ensuring sufficient depth of disposal, by maintaining waste package integrity for as long as possible, by limiting the solubility of radionuclides and the wasteform, by minimising groundwater inflow, by providing a long travel time for groundwater from a GDF to the biosphere, and by providing for additional retardation of any radionuclides released from the waste package.

3.1.2 Optimisation

GRA Requirement R8: Optimisation

The choice of waste acceptance criteria, how the selected site is used and the design, construction, operation, closure and post-closure management of the disposal facility should ensure that radiological risks to members of the public, both during the period of authorisation and afterwards, are as low as reasonably achievable (ALARA), taking into account economic and societal factors.

3.1.2.1 Optimisation and optioneering

Consistent with UK radiation protection legislation, the GRA [1], and the Environment Agency's recently published principles of optimisation in the management and disposal of radioactive waste [67], we define **optimisation** as the principle of ensuring that radiation exposures are as low as reasonably achievable (ALARA) in the given circumstances. This is a narrower definition of the word optimisation, compared to its more general sense of a process of finding the best decision, result or way forward in given circumstances. The GRA recognises this distinction, and requires the developer to carry out "*…options studies, where there are choices to be made among significantly different alternatives*" ([1], GRA paragraph 6.3.54). We refer to such options studies as **optioneering**.

As discussed in Section 2.1, the UK Government's policy is that the siting of a GDF in the UK should be based on the principle of voluntary participation by potential host communities. The UK Government, not us, will choose the site(s), in partnership with the local community, although the choice of site(s) will consider site evaluation work conducted by ourselves and reviewed by others, including the environmental regulators. While radiological safety analyses will play a role in our evaluation of candidate sites, our site evaluation work will also consider a wide range of other aspects (e.g. transport, engineering feasibility, social impacts). We expect it to be some time before we have sufficient understanding of candidate sites and preferred disposal concepts for radiological assessments to be useful for underpinning optimisation decisions.

We will be conducting *optioneering* analyses throughout the entire lifecycle of GDF implementation, operation and closure, including analyses to help inform UK Government decisions on site selection. Our approach to optioneering is set out in [68]. Our ability to inform decisions based on *optimisation* considerations will increase as we move through the MRWS Site Selection Process and obtain the necessary information specific to particular candidate sites and disposal concepts.

At the end of Stage 5, we expect to have a substantial understanding of the investigated site(s) from information gathered during the surface-based investigations. This understanding will support the UK Government's selection of a preferred site, and will enable us to select preferred disposal concepts and designs for that site. However, we will obtain more information from the underground operations that constitute Stage 6 of the MRWS Site Selection Process, which could have implications for the precise layout and detailed design of a GDF. Therefore, the Stage 5 design work will recognise – for the preferred disposal concepts – the potential for variations in depth of the underground facilities, layout of the disposal rooms, excavation dimensions, and rock support that would be addressed in Stage 6.

This distinction between optioneering and optimisation also reflects the fact that selection of the most appropriate method for implementing geological disposal requires assessments and decisions at different levels of detail. In planning how we do this, it is important to distinguish between the assessments and decisions on geological disposal concepts to be carried forward at a given point in the programme, and the more detailed assessments and decisions to identify preferred design solutions to implement such concepts.

In Stage 6 of the MRWS Site Selection Process, we expect to focus on *optimisation* of the design, construction, operation, closure and post-closure management of the preferred disposal concept for the given site and inventory in terms of overall radiological safety. Radiological safety analyses (with appropriate uncertainty analysis) will be an important input into refinement of the design of a GDF, based on site-specific geological, hydrogeological and geochemical features and properties that emerge from surface-based and underground site characterisation.

The application of the ALARA principle for the operational period of a GDF will be carried out in conjunction with the application of the ALARP principle (As Low As Reasonably Practicable) in the OSC. We will apply the concept of Best Available Technique, as set out in Environment Agency guidance [67], to demonstrate that potential discharges / releases of radioactivity to the environment are ALARA in the given circumstances.

3.1.2.2 Optimisation and optioneering considerations and constraints

In balancing relevant considerations as part of optimisation studies, the reduction of radiological risk "*…should not be given a weight out of proportion to other considerations. In other words, the best way forward is not necessarily the one that offers the lowest radiological risk*." ([1], GRA paragraph 6.3.52). Some relevant considerations are listed in the GRA [1] and include:

- the number of people (workers and the public) and other environmental targets that may be exposed to radiological risk;
- the chance they could be exposed to radiation, where exposure is not certain to happen;
- the magnitude and distribution in time and space of radiation doses that they will or could receive;
- nuclear security and safeguards requirements;
- issues similar to those above, but relating to non-radiological hazards;
- economic, societal and environmental factors; and
- uncertainties in any of the above.

Similar criteria may be considered in optioneering analyses, but there is greater scope to tailor the criteria to the issue under consideration (e.g. engineering feasibility could be an important consideration) and the weight given to particular criteria may differ.

Any decision is necessarily constrained by the information available and the circumstances prevailing at the time it is taken. In the case of decisions based on optimisation and optioneering analyses for a GDF, additional constraints that will need to be considered include:

- There are radiological dose constraints during the period in which an environmental permit is held, when a measure of active control over release and exposure is possible.
- The risk guidance level that is applicable after the period in which an environmental permit is held provides an indication of, although not an absolute constraint on, the expected standard of protection that should be afforded over the very long term by an optimised disposal facility.
- The choice of disposal concept is determined in part by the inventory and wasteform, but will also inevitably be constrained by the candidate geological environments and what is known about them.

- Optimisation of the preferred disposal concept will be influenced by consultation with regulators and local stakeholders regarding issues such as whether and for how long to keep open the option of retrievability (see Section 3.1.3.3).
- The main economic factor to be considered is cost; we have an obligation to ensure that public funds are spent wisely.

In the case of the step-wise decision process we will undertake for implementing a GDF, the information base will be incomplete at early stages in the process. This emphasises the importance of documenting decisions and the reasons behind them at all stages of the process, and keeping an open mind about revisiting decisions if new information comes to light. We recognise that even when a decision has apparently been made, it continues to represent an uncertainty until it has been implemented, because the decision still might not be implemented or might be implemented in a different way from that originally envisaged. For example, until a GDF has actually been closed at a given time, it is not certain that it will be closed at that time, and optimisation decisions concerning closure can be periodically reviewed up until the point that closure is implemented.

However, once a decision has been implemented, it forms part of the framework within which further decisions, and the considerations that go with them, must be made. We will not seek to revisit decisions that have already been implemented.

3.1.2.3 Optimisation and optioneering and the ESC

To succeed, optimisation and optioneering analyses require transparency and good communication regarding the processes and procedures used and the judgements made. The staged updates to the ESC discussed in Section 2 will describe how optimisation and optioneering have been applied in decision-making; the ESC will therefore provide a record/audit trail of key decisions in a GDF implementation programme, with evidence for why particular choices were made. It will also set out how the current round of safety analyses has supported, or is designed to support, the particular decisions that need to be made at that particular stage of the programme.

For the reasons noted in Section 3.1.2.1, it is premature to be considering optimisation until much later in a GDF implementation programme. However, we have undertaken several disposal concept options studies [28, 29, 30, 31], as outlined in Section 3.1.3.2. It is these disposal concept options studies that have allowed us to identify a set of relevant illustrative geological disposal concept examples, which run as a thread through our research status reports and whose possible application to the UK is summarised in Appendices A, B and C. The ESC also considers the impacts of alternative inventory scenarios, as documented in the Derived Inventory reports [10, 11, 12, 13], and as outlined in Section 4.1.1 and Section 5.2.2. Work is underway to develop methods for the selection of preferred geological disposal concepts once we have specific candidate sites to consider.

3.1.3 Design

GRA Requirement R12: Use of site and facility design, construction, operation and closure

The developer/operator of a disposal facility for solid radioactive waste should make sure that the site is used and the facility is designed, constructed, operated and capable of closure so as to avoid unacceptable effects on the performance of the disposal system.

Our design strategy includes adoption of passive safety and the use of established technology to achieve the objectives of isolation and containment, as discussed in Section 3.1.1. The design process itself is a key input to optioneering studies and the demonstration of optimisation, as discussed in Section 3.1.2. We discuss here the issues of how we establish and manage design requirements (Section 3.1.3.1) and the relationship to our design process (Section 3.1.3.2). We also give particular attention to the issue of waste retrievability from a GDF, which is an important design consideration that will need to be informed by consultation with candidate communities and regulators (Section 3.1.3.3).

3.1.3.1 Design requirements

The requirements for the disposal system design are expressed through the Disposal System Specification (DSS). As indicated in Figure 1.1, the DSS forms the fundamental basis for the disposal system and covers a large number of interacting variables relating to the waste, its packaging, transport and disposal. The DSS includes constraints imposed by regulators' and stakeholders' requirements, as well as defining the nature and characteristics of the waste to be considered. Designs are subject to safety and environmental assessment, the output of which is fed back into the design and site characterisation activities and, as appropriate, the DSS. All of these activities are used to determine R&D needs.

The DSS comprises a functional specification and a technical specification for a single disposal facility for UK higher activity radioactive wastes:

- The purpose of the Disposal System Functional Specification (DSFS) is to identify, document and obtain agreement on the high-level requirements of the disposal system, and any constraints on it [22]. The DSFS is in a form suitable for a wide range of stakeholders.
- The purpose of the Disposal System Technical Specification (DSTS) is to set out and justify requirements, constraints and data relevant to the disposal system [23]. The DSTS is written at a significantly more detailed level than the DSFS, and provides the designers of the disposal system with the requirements that must be satisfied. Each requirement is supported by a justification statement.

Changes to the DSS are controlled by specific procedures set out in [69].

At the current stage of the MRWS Site Selection Process, only generic requirements not specific to a particular geological environment or geological disposal concept can be defined. These requirements come from the MRWS White Paper [4], regulatory guidance such as the GRA [1], and our high-level safety strategy (see Section 3.1.1). As candidate sites are identified, stakeholder requirements specific to the host communities will also be identified.

We have a high degree of flexibility in our design process at the moment, and are considering a wide range of possible geological environments and geological disposal concepts, as discussed below. This allows us to include consideration of inventory uncertainty, and alternative geological environments and geological disposal concepts in option analyses.

As we move forward through the MRWS Site Selection Process, it is envisaged that further more detailed specifications derived from the DSS will describe how the requirements of the functional specification and the technical specification will be met. At some point during Stage 6 (underground operations), it is envisaged that we will need to have a design specification that describes GDF design in some detail. Contract specifications would then translate technical requirements from the design specification into descriptions of the work that can be carried out by individual designers or implementation contractors on specific components of a GDF. Contract specifications would contain acceptance criteria for each requirement.

We are currently developing a requirements management system that we plan to use to track design requirements and to ensure that any modifications to design requirements are implemented effectively and are traceable. Contract specifications may be linked to the system acceptance criteria that are in the requirements management system to ensure that contractually specified text and acceptance criteria for components or sub-systems are consistent with the requirements of the geological disposal system as a whole.

3.1.3.2 Design development

As part of our preparation for implementing geological disposal of higher activity radioactive waste, we have developed a detailed understanding of the options for design of a GDF for different waste types:

- We have catalogued and studied disposal concepts in other countries, and considered their applicability to UK wastes and geological environments [28, 29].
- We have catalogued and studied materials that could be used in the EBS, and considered their applicability to UK wastes and geological environments [30].
- We have considered the suitability of the Nirex Phased Geological Repository Concept for the disposal of ILW/LLW to a range of geological environments [31].

We have used these options studies to help identify the relevant illustrative geological disposal concept examples set out in Appendices A, B and C, as discussed in Section 4.1.4. Design issues relevant to these examples are discussed in more detail in the Generic disposal facility designs report [24] that forms part of the DSSC Tier 2 document set. We have also drawn up a preliminary set of Design principles to inform future design work [70].

The identification of candidate sites will allow us to consider the application of existing UK and overseas disposal concepts to specific sites. Implementation of our site investigation and R&D strategy, set out in Section 3.1.5, will enable us to identify and assess the possible range of geological disposal concepts that could be implemented at each site, taking account of the waste inventory for disposal. As the programme progresses, we will move from high-level decisions on the concepts to be considered, to concept-specific design decisions on the range of materials to be considered and, eventually, to decisions on the designs and materials to be used. As discussed in Section 3.1.2, we can only make meaningful selections of concepts and design components when we have sufficient site-specific information.

As indicated in Section 3.1.1.3, we expect to develop at least two different EBS designs, one for ILW/LLW and one for HLW, to suit the properties of the different waste types. These two disposal concepts will be designed to ensure that interactions that might result from co-location of the ILW/LLW and HLW disposal areas would not compromise the performance of the geological disposal system. We have undertaken studies to investigate the potential implications of co-location, and recognise that this will be an important aspect of the ESC for such a facility [e.g. 71, 72]. For the generic ESC, the design concepts for ILW/LLW have been extended to include consideration of DNLEU, and the design options for HLW have been extended to include consideration of SF, Pu and HEU. However, we have initiated work to catalogue geological disposal concepts for uranium and plutonium

Following Stage 5 of the MRWS Site selection process it is expected that will have a substantial understanding of the sites(s) from information gathered during surface-based investigations. As this understanding is developed during Stage 5, this will enable us to select a preferred concept and design for the site(s). In Stage 6 of the MRWS Site Selection process, it is expected that we will focus on optimisation of the design, operation and closure of a GDF for a given site and inventory, in terms of overall radiological safety.

We will follow an iterative approach to facility design and development of the ESC as results are progressively obtained from the site characterisation, safety assessment and R&D activities, and as the exact inventory of higher activity radioactive wastes requiring geological disposal becomes better defined, with the designs becoming more detailed as work proceeds towards implementation.[10]

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[10] The same approach will be taken for development of the OSC, with requirements coming from the DSS and underlying specification documents, and the design being reviewed and defined in more

With regard to *design issues*, our documentation for the “initial site evaluation” during Stage 5 of the MRWS process (Figure 2.1) will:

- develop outline engineering design(s) appropriate to the geological environment(s) at the candidate site(s), and demonstrate consistency with safe operation [73] and acceptable operational and post-closure environmental performance;
- describe the main features of the safety strategy that might be adopted in the design process for each site, and explore how well the strategy might be met in terms of whether there is a multiple-function and multiple barrier environmental safety approach;
- identify the key technical challenges for design and implementation of a GDF at each site and devise strategies that will allow them to be overcome; and
- explain how the design for construction, operation and closure of a GDF at each site will avoid any unacceptable effects on the performance of the geological disposal system.[11]

Information will be drawn in part from our Stage 4 desk-based site assessments (see Section 2.1 and [33]).

3.1.3.3 Retrievability

Our approach to the issue of **waste retrievability,** that is, the ability to recover waste from a GDF prior to closure albeit with some effort, is based on the position of the UK Government set out in the 2008 MRWS White Paper ([4], paragraphs 4.20-4.22]:

> *“Government acknowledges that there is a divergence of views on the issue of waste retrievability, but on balance considers that CoRWM’s conclusion was correct, i.e. that ‘leaving a facility open, for centuries after waste has been emplaced, increases the risks disproportionately to any gains’ [74]. Closure at the earliest opportunity once facility waste operations cease provides greater safety, greater security from terrorist attack, and minimises the burdens of cost, effort and worker radiation dose transferred to future generations.*
>
> *“…it is likely to be at least a century…until final closure of an entire facility is possible [74]. In practice it could be longer. This timescale provides sufficient flexibility for further research to be undertaken.*
>
> *“Hence Government’s view is that the decision about whether or not to keep a geological disposal facility (or vaults within it) open once facility waste operations cease can be made at a later date in discussion with the independent regulators and local communities. In the meantime the planning, design and construction can be carried out in such a way that the option of retrievability is not excluded. Any implications for the packaging of waste will be kept under review.”*

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detail as more site information becomes available, and more safety assessment, and research and development is undertaken.

[11] This parallels the consideration of how site characterisation could affect site integrity, discussed in Section 2.2.2 and Section 3.1.5.1.

Therefore, we need to maintain flexibility in the design to keep open the option of retrievability of waste. However, excavations in certain rock types will require extensive engineered support to keep a GDF (or the disposal areas within it) open for extended periods. Early engagement with potential host communities will be necessary if proposed design decisions may impact adversely on the option of retrievability. We will seek to discuss both the potential benefits and the potential disadvantages of retrievability with candidate communities at an early stage to ensure that we consider suitable disposal concepts and appropriately consider socio-economic factors in optimisation of the preferred concept. Consideration of these issues forms part of our forward programme of R&D (see Section 3.1.5.2 and Section 6.3).

Our position on retrievability is summarised in [75] and the implications of retrievability are discussed in the Disposal System Technical Specification [23]. With regard to GRA Requirement R12, we will need to be able to demonstrate, whatever our eventual design, that a GDF can be constructed, operated and closed so as to avoid unacceptable effects on the performance of the geological disposal system. In particular, if it is proposed to keep a facility open that would otherwise be ready for closure in order to maintain the option to retrieve waste emplaced in the facility, then the ESC would need to demonstrate that processes such as degradation of waste packages would not unacceptably affect either operational environmental safety or post-closure safety.

3.1.4 Waste packaging and waste acceptance criteria

GRA Requirement R13: Waste acceptance criteria

The developer/operator of a disposal facility for solid radioactive waste should establish waste acceptance criteria consistent with the assumptions made in the environmental safety case and with the requirements for transport and handling, and demonstrate that these can be applied during operations at the facility.

The process of developing waste acceptance criteria (WAC) will begin once we begin to develop a site-specific understanding of the safety issues at candidate sites; these criteria will continue to be refined up to the point where we apply for an environmental permit to begin waste disposal. In the meantime, we are using a disposability assessment process to evaluate proposed waste packages; if our assessment shows that a proposed waste package would be disposable, we endorse it by the issue of a Letter of Compliance (LoC). We summarise below our current disposability assessment process (Section 3.1.4.1), the relationship between this process and the future development of WAC (Section 3.1.4.2), and the role of the ESC in the disposability assessment process (Section 3.1.4.3). The current status of the disposability assessment process, in terms of the percentage of the inventory by waste type for which we have issued a LoC, is summarised in Section 4.1.2.1.

3.1.4.1 The disposability assessment process

Under the disposability assessment process we interact with waste producing organisations in three complementary areas:

1. provision of specifications for waste packaging, based on geological disposal;
2. assessment of individual packaging proposals and provision of disposability assessment and, where appropriate, formal endorsement; and
3. formal quality management system assessment and audit of the process for development and manufacture of waste packages.

The disposability assessment process is made available to all nuclear licensed site operators in the UK, in line with regulatory requirements for production of Radioactive Waste Management Cases [76]. This includes sites in Scotland where the Scottish Government policy is currently for on-site storage of higher activity radioactive wastes [77].

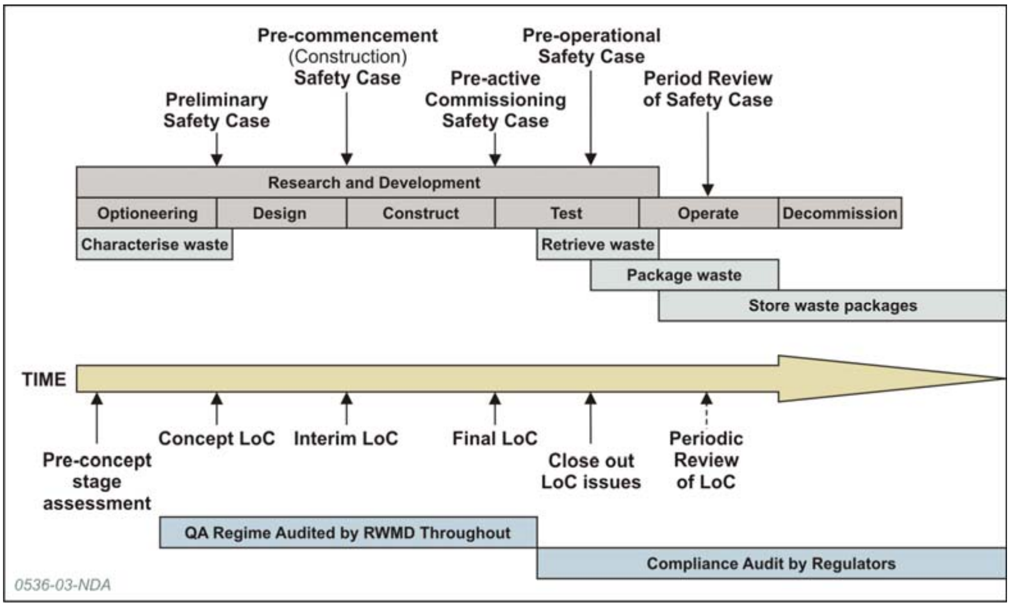
The disposability assessment process operates as described below:

- **Waste packaging proposal.** The disposability assessment process is initiated when a site operator submits a waste packaging proposal that includes reasoned arguments to demonstrate that, following conditioning, the waste is likely to be compatible with existing and future storage, transport and disposal operations. The waste packaging submission must describe the nature of the wastes and plans for retrieval, characterisation and assay, the conditioning process proposed, and the form of the proposed waste package. The packaging specifications are defined in generic waste package specification reports [78, 79]. For example, the wasteform is required to be stable, to meet certain criteria regarding the content of fissile material (to provide criticality safety), to meet certain criteria regarding the durability of the waste packaging and handling features, and not to contain substances that might be detrimental to GDF performance.
- **RWMD disposability assessment**. We then undertake assessment of the proposal, judging the waste package against the published specifications and for compliance with the safety, environmental and security assessments that support transport and geological disposal (i.e. those that underpin the DSSC, see Figure 1.2). Over the last few years we have been evaluating compliance of waste packaging proposals against generic reference disposal concepts and associated generic safety and environmental assessments that were designed to be conservative [e.g. 80, 81, 82, 83], so that we can have confidence that packaged wastes will eventually be acceptable for disposal in a GDF. Our disposability assessment considers the impact of each new waste packaging proposal in light of the total inventory we are likely to have to manage in a GDF. The generic ESC forms a further development of this approach and provides a basis for our disposability assessments going forward. Our approach provides a consistent basis for each waste packaging evaluation, and will continue to be maintained throughout the MRWS Site Selection Process.
- **RWMD response.** We respond to the waste packaging proposal with a disposability assessment report that describes the assessments undertaken and sets out our rationale as to why (or conversely why not) the proposal is judged to be compliant with plans for future waste management considering safety and environmental assessments for transport, and GDF operations and post-closure. The disposability assessment report is also the vehicle for identifying further information needs, or setting out the need for further research and/or development. Where a proposed waste package is found to be compliant with packaging specifications and the associated safety and environmental assessments for our reference transport and disposal concepts, we signify this through the issue of a Letter of Compliance.

The disposability assessment process will generally be applied in three stages throughout the development of the waste package (Figure 3.3): at the concept stage, when the packager is looking to define a preferred conditioning and packaging option; at the interim stage prior to commencing ‘design and build’ of the waste packaging plant; and at a final stage when the waste packaging plant is constructed and prior to commencement of operations. By the time the final stage is reached, the disposability case should have been developed to completion, with the final stage LoC signifying that to the best of knowledge at that stage, the waste package will be ‘disposable’.

Figure 3.3 Our staged disposability assessment process, and the relationship to the design, construction and operation of a waste packaging plant by a waste producer

The safety cases shown are the responsibility of the waste producer, not RWMD.



A key issue is the mechanism by which the LoC and associated disposability assessment is kept up to date. A process called 'periodic review' has been developed so that the disposability case and LoC can be revisited and updated on a periodic basis [84]. The object of the review is to give confidence to all stakeholders that the currency of the LoC is maintained and that the waste package remains disposable when judged against the latest information and safety cases. The steps to be taken for any waste package that may no longer be considered disposable would need to be considered as part of optimisation studies (see below and Section 3.1.2). Decision-making for such packages would include consideration of both repackaging and GDF design modifications. However, given the conservative approach to disposability assessment and the range of geological environments and conditions we are considering in the generic DSSC (see Sections 4.3.2 and 5.2.2 and [17]), we consider it unlikely that previous waste packaging decisions will later need to be reversed.

The disposability assessment process is recognised by regulatory guidance as an important component of the strategy for licensing waste packaging activities.

The disposability assessment process, as currently implemented, is flexible and enables site operators to develop new package types that may better meet the needs for their specific types of waste. The disposability assessment process provides the mechanism by which such proposals can be assessed and, as appropriate, accommodated within the design of a GDF. As GDF design and safety cases are developed and become increasingly site-specific, so too will the waste packaging specifications.

3.1.4.2 Relationship between disposability assessment process and the development of WAC

As noted above, the disposability assessment process will eventually become a waste acceptance process. WAC will be formalised on the basis of the assumptions in the ESC, OSC and TSC, considering, among other things, physical, chemical, thermal, mechanical, and radiological constraints on waste packages to ensure that we will be able to transport, handle and dispose of the wastes safely. Once we have permission to dispose of wastes and site operations have commenced, waste consignments will be checked against the WAC.

When setting the WAC, we will need to consider the impact on wastes already packaged or in waste streams for which a LoC is already in place. It is unlikely to be optimal to make operators re-process/re-package wastes, incurring expense and significant worker doses, if we can design a GDF that can take the existing packages. This consideration forms part of our 'upstream' optimisation process considering the entire lifecycle of the waste.

3.1.4.3 Role of the ESC in the disposability assessment process

There is a close link between the generic ESC and the disposability assessment process:

- The environmental safety assessment calculations we have undertaken in support of this generic ESC (and the additional safety assessment calculations we have undertaken in support of the OSC and TSC) are intended to underpin decisions taken to endorse waste packages as part of the disposability assessment process. This approach is set out in our DSSC Tier 2 report Radioactive wastes and assessment of the disposability of waste packages [14].
- Another important link between the ESC and the disposability assessment process is that GDF design work is conditioned by the presence of a large volume of wastes that will have been packaged in line with the receipt of a LoC in an earlier stage of the project. In fact, this is already the case for the generic ESC, as discussed in Section 4.1.2.1, and will be even more relevant for future site-specific updates of the ESC as increasingly more waste is packaged.

Our approach to safety assessment calculations at the generic stage is to make use of a reference benchmark model derived from the illustrative geological disposal concept examples for a geological environment characterised by higher strength rock. We discuss the benchmark models used with regard to the generic ESC in Section 4.3.2 and Section 5.2.2 and in more detail in our Generic Post-closure Safety Assessment (PCSA) report ([27], Section 4.5.4). We also consider the implications for disposal in other geological environments, and argue that basing decisions under the disposability assessment process using this reference benchmark model is likely to be conservative. That is, we consider that wastes packaged according to a LoC issued at the generic stage will also be disposable if we are presented with other geological environments to consider as part of the MRWS Site Selection Process.

We note that the *precise* values of calculated peak risks or other performance measures in the generic PCSA cannot be particularly meaningful at this generic stage, because they depend on parameters that represent quantities that cannot be known until a candidate site(s) and site-specific geological disposal concepts have been identified. However, we can obtain useful information about the *relative* contributors to the calculated peak risk. This informs us which of the different components of the waste contribute most to the total risk, and which radionuclides in the inventory contribute most to the total risk. This helps focus our design and research work on issues that matter and, similarly, ensures that we address *relevant* issues when assessing packaging proposals as part of the disposability assessment process.

3.1.5 Site investigation and R&D

GRA Requirement R11: Site investigation
The developer/operator of a disposal facility for solid radioactive waste should carry out a programme of site investigation and site characterisation to provide information for the environmental safety case and to support facility design and construction.

3.1.5.1 Site investigation

We have established a Preparation for Surface-based Investigations Project, which is considering issues associated with a characterisation programme covering a site for a single GDF for all higher activity radioactive wastes [e.g. 85]. The issues considered include:

- information needs, i.e. what to investigate / measure;
- techniques, i.e. how to investigate / measure; and
- implementation, i.e. when to investigate / measure.

Consistent with approaches adopted in other countries, site characterisation, safety assessment and design programmes will proceed in a staged but integrated fashion. We will need to find a compromise between completely parallel programmes, where site investigation has no feedback from the developing safety case, and the ideal of evaluating and thinking about the data from each campaign before deciding what to do next, which is unrealistic in time, cost, and continuity. The approach should allow prioritisation of the site investigation activities on the basis of previous safety assessment and design studies, albeit studies based on data that come from site investigations preceding the last cycle of investigation.

We intend to develop an integrated description of the candidate site(s) using data from site investigations and the R&D programme. This description will be built up from ‘site descriptive models’ covering disciplines such as geology, hydrogeology, hydrochemistry, geotechnics, radionuclide transport, including multi-phase flow, thermal properties, and biosphere processes. The site descriptive models will provide more detailed descriptions of the characteristics of the site than will the environmental safety assessment models. They will also underpin and provide data for the assessment models.

Section 2.1 describes the purpose of site evaluation at the desk-based studies stage of the MRWS Site Selection Process (Stage 4). The nature of the geological environment will be one of the aspects considered in evaluating the candidate site(s). We will want to understand how easy it may be to develop the requisite level of confidence in the performance of the geological barrier. Our proposed evaluation criteria also include consideration of the degree of difficulty in characterising the site adequately.

Section 2.2.2 notes that the ESC update for surface-based intrusive site investigations will include an assessment of the extent to which such characterisation activities might disturb the site and any implications this might have for the environmental safety of a possible future GDF.

3.1.5.2 Research and development

In consultation with stakeholders, we have developed and published a strategy for implementing an R&D programme that has the specific objective of supporting geological disposal in the UK [86]. This strategy provides a basis for identifying R&D needs to support a process of disposal system specification, disposal concept optioneering and design optimisation, and assessment of particular concepts or designs. The strategy also considers prioritisation of R&D needs considering their significance (or potential impact) on safety and/or delivery and the current level of understanding. R&D work is scheduled to take into account when the information is required, the time needed to complete each activity, and the need to make effective use of resources.

We are building strong relationships with our stakeholders and seeking to increase their involvement in our R&D programme planning and evaluation. We plan to set up a programme of technical workshops to engage with our stakeholders on R&D and to help them influence our future programme.

We have defined the following specific R&D themes for consideration at present:

- develop and expand our HLW and SF R&D programme to develop designs for disposal of these materials and to assess their safety;
- support the development of future management strategies for materials such as separated plutonium and uranium by developing the technical understanding of disposal issues associated with them;
- continue R&D into ILW disposal, focusing on specific topics that have been identified as important for ensuring safety, or for optimising conditioning and packaging arrangements;
- address implementation issues by carrying out R&D into topics such as technical aspects of retrievability and the implications of a single GDF for all higher activity radioactive wastes;
- prepare for site characterisation by developing appropriate skills and techniques to support development of a GDF in a range of geological environments; and
- investigate the social aspects of implementing a GDF.

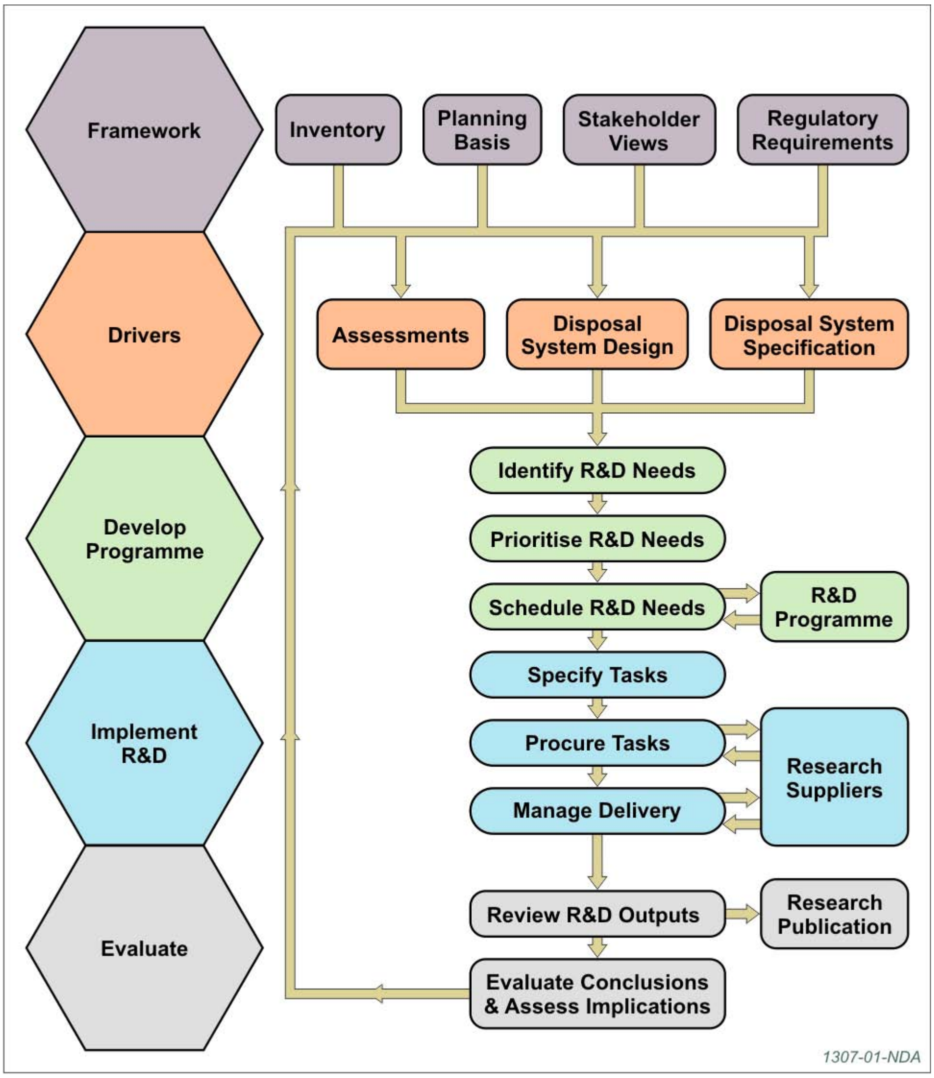
The first three of these themes relate directly to advancing our understanding of the types of wastes and materials that may require geological disposal (including HLW, ILW, SF, and separated plutonium and uranium). The remaining three themes recognise that the voluntarism approach to site selection means that we need to understand the implications of disposal in a range of geological environments and prepare accordingly for site-based work.

In setting out our R&D strategy, we recognise that we are in the early stages of a project that will last for several decades. We will need to remain responsive to the changing needs of the project as it proceeds through site investigation, to construction and operation of a disposal facility, and adapt our R&D accordingly. We will periodically update and revise our site characterisation and R&D programmes, and refine our needs, as part of the iterative disposal system specification, design optioneering/optimisation and ESC cycle. We recognise that the site characterisation and R&D programmes need to be sufficiently flexible to address a wide range of possible new needs that will emerge at later stages of the project, including those that:

- are identified by site characterisation and R&D themselves;
- result from new developments such as new waste packaging techniques or materials; and
- result from new regulatory or legislative requirements that could occur over the many decades it will take to implement a GDF.

A summary of our current R&D work programme is provided in Section 6.3. Our process for delivering R&D is illustrated in Figure 3.4.

Figure 3.4 Our process for delivering R&D



3.1.6 Monitoring

GRA Requirement R14: Monitoring   **In support of the environmental safety case, the developer/operator of a disposal facility for solid radioactive waste should carry out a programme to monitor for changes caused by construction, operation and closure of the facility.**

The GRA [1] defines monitoring as taking measurements so as to be aware of the state of the geological disposal system and any changes to that state. We consider monitoring to cover **continuous or periodic** observations and measurements of engineering, environmental or radiological parameters to help evaluate the state and behaviour of the geological disposal system or the impacts of a GDF and its operation on the environment. This is consistent with the definition proposed by the IAEA for monitoring of a GDF in [87]. The site investigation activities or one-off measurements referred to in Section 3.1.5 are outside our monitoring strategy, except insofar as such activities are used to define the baseline for monitoring.

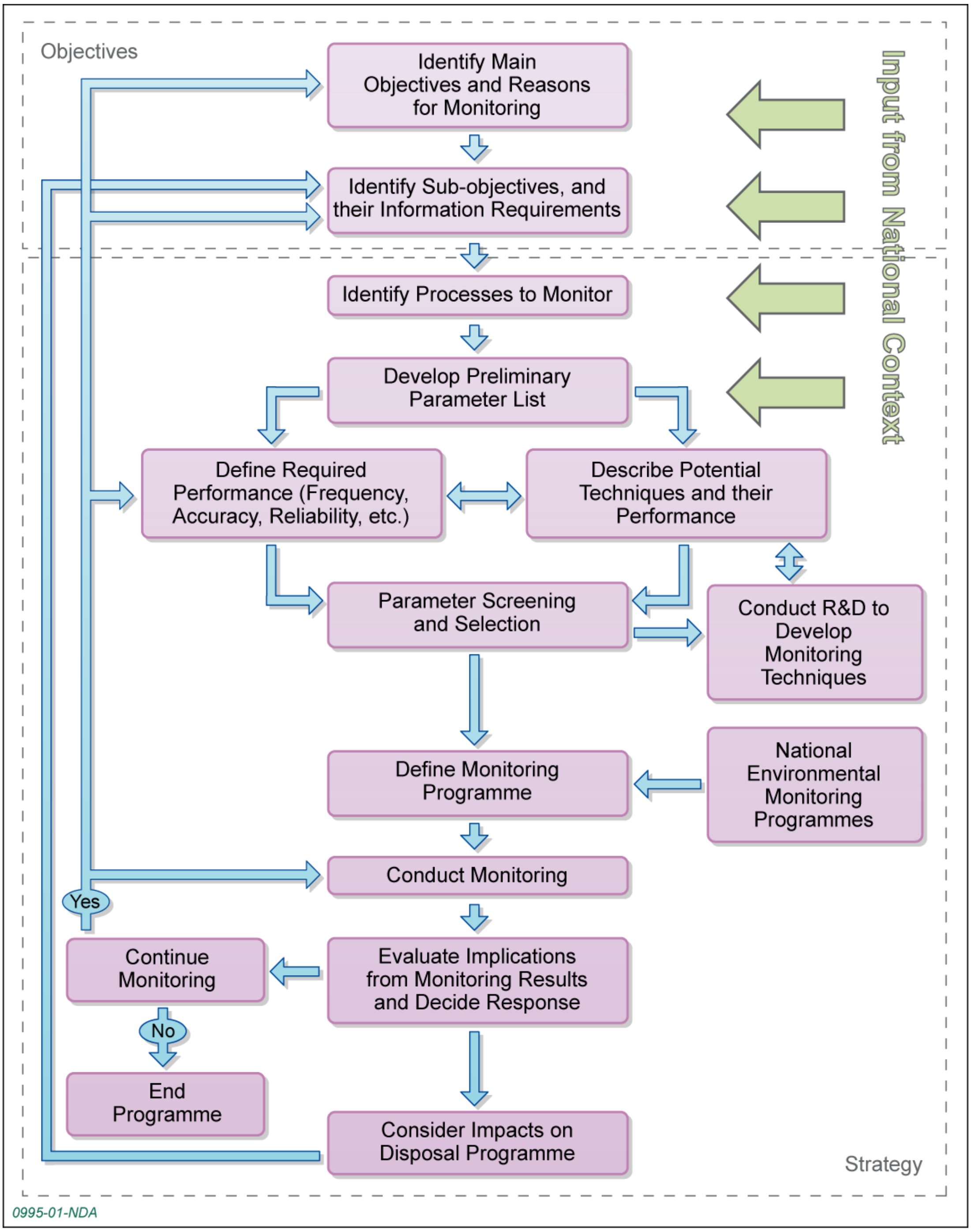
The development of our monitoring strategy is being informed by participation in international R&D programmes on monitoring of GDFs [88, 89]. Our approach to monitoring strategy development is illustrated in Figure 3.5. Note that we will conduct further R&D where needed to develop and implement our monitoring strategy.

Many different activities in a GDF programme would require data from monitoring; not all of these activities are related to environmental safety. Key monitoring objectives would be derived from:

- ***Environmental Safety Case***. Data from monitoring can be used to support and build confidence in modelling activities and safety arguments. Safety assessments can be used to prioritise potential monitoring objectives and associated monitoring parameters that are most significant to environmental performance and/or confidence in performance. This may include monitoring to ensure that engineering has been implemented to the standard assumed in the ESC, and to build confidence in our understanding of the extent and timing of impacts on the host rock from excavation and waste emplacement activities. However, as discussed in Section 3.1.3, a GDF would be designed to be passively safe, such that there is no reliance on monitoring in the post-closure period to provide safety, and that ultimately active institutional control can be withdrawn.
- ***Operational Safety Case***. Conditions must be monitored to assure the site licence conditions are met. This would include radiological exposure of the operating staff and non-radiological safety issues related to working in a GDF and associated facilities. Again, the OSC assessments can be used to identify and prioritise monitoring objectives and parameters.
- ***Stakeholder consultation***. Several monitoring objectives might be identified in addition to those for the activities discussed above to inform the public and build wider confidence in the safety of a GDF. Examples include monitoring of the facility environment and structure to ensure that waste retrieval - if required - could be undertaken safely, monitoring of waste management developments elsewhere, and post-closure monitoring of groundwaters to prove no unexpected release of radioactivity.
- ***Strategic Environmental Assessment (SEA) and Environmental Impact Assessment (EIA)***. Both the SEA and EIA for a GDF would include monitoring commitments that are intended to identify environmental impacts requiring mitigation during construction and operation of a GDF. These commitments could be consolidated into objectives for monitoring.

The monitoring objectives overlap in terms of information requirements that could be fulfilled by monitoring. Therefore, we would develop an integrated monitoring strategy that considers all the objectives and associated information requirements together. This would ensure that, wherever possible, there is a common dataset underpinning different parts of a GDF development programme. We also note that the monitoring plan(s) would need to be tailored to the candidate site(s) and GDF design(s).

Figure 3.5 Development of monitoring strategy for a GDF



We expect that the monitoring programme would include:

- measurements of pre-existing radioactivity in appropriate media to establish baseline conditions;
- monitoring of geological, physical and chemical parameters that are relevant to safety and that might change as a result of site characterisation, GDF construction and waste emplacement;
- radiological monitoring during the period in which an environmental permit is held to confirm that operation of a GDF is not causing an unacceptable impact on the environment;

- monitoring during the operational phase to confirm that GDF operation is being carried out safely and to inform decisions on how best to manage the facility (e.g., when and how to backfill, when and how to seal); and
- non-radiological parameters during construction and operation of a GDF to confirm our understanding of the effects of construction, operation and closure on the characteristics of the site and the wider local environment and community.

We may also selectively monitor aspects of the geological disposal system after closure to provide confidence that it is evolving in the way we expect. However, monitoring that could compromise the safety of a GDF would not be undertaken.

The monitoring activities can be divided into two categories: initial monitoring to obtain baseline data and ongoing monitoring to ensure safety. For monitoring activities that are designed to ensure safety, for example convergence monitoring of excavations, we would define trigger levels for action and an action plan to be executed should the trigger levels be exceeded, before we began the activity we were monitoring.

The duration of each monitoring programme would be defined in terms of the phases of development of the disposal facilities, as set out in the GRA [1]:

- ***Pre-construction***. We would establish the baseline against which we would monitor at later stages. We would ensure that our monitoring strategy is developed and implemented in time to ensure that, where necessary, the monitoring baseline could be established adequately before any construction starts. We would also monitor the impact of site investigation work to ensure there were no significant adverse effects on the environment and the local community.
- ***Construction***. We would need to monitor for non-radiological impacts related to construction. However, where possible, we would also use monitoring to build confidence in our understanding of the geological disposal system and to provide assurance that it is appropriate to move to the next stage of the programme. Thus, predictions would be made prior to activities such as underground construction against which actual data could be compared.
- ***Operations***. Operational safety issues will be a paramount concern for monitoring during this period. Also, we would undertake radiological monitoring and assessment to provide evidence of compliance with permitted discharge limits, and monitoring to show that the system was evolving in the expected way.
- ***Closure***. We would show that we have closed a GDF in compliance with our design and safety assessment assumptions. The decommissioning of surface facilities would also be monitored to ensure there were no significant adverse effects on the environment and the local community. Monitoring of the response of the geological disposal system to cessation of underground activities and installation of the closure engineering would be compared to model predictions to build confidence in our understanding of the geological disposal system and the post-closure safety assessments.
- ***Post-closure***. Any post-closure monitoring would focus on those parameters where we consider there may be detectable changes or trends that we could use to demonstrate that our understanding of the geological disposal system and the post-closure safety assessments are appropriate. Safety would not depend on monitoring. Although the design intent would be to make post-closure monitoring unnecessary, it may continue for the purposes of reassuring the public, as long as it does not have an unacceptable impact on environmental safety.

3.2 Assessment strategy

GRA Requirement R5: Dose constraints during the period of authorisation [12]

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

GRA Requirement R6: Risk guidance level after the period of authorisation

After the period of authorisation, the assessed radiological risk from a disposal facility to a person representative of those at greatest risk should be consistent with a risk guidance level of 10^{-6} per year (i.e. 1 in a million per year).

GRA Requirement R7: Human intrusion after the period of authorisation

The developer/operator of a geological disposal facility should assume that human intrusion after the period of authorisation is highly unlikely to occur. The developer/operator should consider and implement any practical measures that might reduce this likelihood still further. The developer/operator should also assess the potential consequences of human intrusion after the period of authorisation.

GRA Requirement R9: Environmental radioactivity

The developer/operator should carry out an assessment to investigate the radiological effects of a disposal facility on the accessible environment, both during the period of authorisation and afterwards, with a view to showing that all aspects of the accessible environment are adequately protected.

GRA Requirement R10: Protection against non-radiological hazards

The developer/operator of a disposal facility for solid radioactive waste should demonstrate that the disposal system provides adequate protection against non-radiological hazards.

In accordance with international good practice, the GRA requires us to support the ESC with assessments of safety. Safety assessment is defined in the IAEA safety requirements for geological disposal [7] as the process of systematically analysing the hazards associated with a GDF and the ability of the site and the design of the facility to provide for the safety functions and to meet technical requirements. Assessments may include quantitative modelling, reasoned arguments, and the use of qualitative evidence to build confidence in the assessment [8].

The GRA requires the assessment of several issues associated with radiological and non-radiological hazards, including:

- ***Radiological dose to the public during the period in which an environmental permit is held***, i.e. while disposals are taking place and any period afterwards while the site is under active control (Section 3.2.1). Requirement R5 of the GRA establishes constraints for the annual effective dose to a representative member of the critical group during this period as a safety requirement. However, the safety assessment needs only to consider routine exposures to the public; the potential consequences of accidents and radiological doses to workers are considered under the operational safety assessment [6] which is reported outside this ESC.
- ***Radiological risk to the public after the period in which an environmental permit is held***, including risks from releases to groundwater and as gas, and from potentially disruptive natural events (Section 3.2.2). As a safety measure for this assessment, Requirement R6 of the GRA establishes a risk guidance level for comparison to the calculated radiological risk to a person representative of those at greatest risk.

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[12] As already noted, with the establishment of EPR 2010 [2], reference to an "authorisation" in the GRA should be taken to mean an "environmental permit" for a GDF in England or Wales.

- ***Potential consequences of inadvertent human intrusion into a GDF after the period in which an environmental permit is held***, in terms of radiation doses to individuals undertaking the intrusion and individuals and other organisms who might occupy the area affected by releases from the intrusion event (Section 3.2.3). GRA Requirement R7 does not provide a safety measure, but does require an assessment of potential consequences as an input to optimisation studies. The assessment also needs to consider the consequences of intrusion in a wider geographical sense and for the long-term behaviour of a GDF. Requirement R7 also asks us to consider any practical design measures that could reduce the likelihood of intrusion, again as in input to optimisation studies. However, implementation of any such measures must not compromise the environmental safety performance of a GDF if intrusion does not occur.
- ***Radiological impacts on the accessible environment from releases***, e.g. through damaging habitat quality, and to non-human species (Section 3.2.4). GRA Requirement R9 does not provide a safety measure, and there are currently no nationally or internationally established criteria for determining radiological protection of the environment. However, it is an assumption in our Radiological Protection Policy Manual [73] that measures to protect humans would also be sufficient to protect non-human species. This is an area of ongoing study and we will draw conclusions about the effects of a GDF on the accessible environment using the best available information at the time of the assessment [90, 91].
- ***Non-radiological hazards*** (Section 3.2.5). The radioactive wastes disposed of in a GDF also contain a non-radiological hazardous component. GRA Requirement R10 of the GRA requires us to demonstrate an adequate level of protection from such hazards. The GRA states that ([1], paragraph 6.42): "*There are nationally acceptable standards for disposing of hazardous waste. However, these standards may not be suitable to apply directly to waste that presents both radiological and non-radiological hazards. Accordingly, these standards need not necessarily be applied, but a level of protection should be provided against the non-radiological hazards that is no less stringent than would be provided if the standards were applied*."
- ***The role of individual barriers or components of the geological disposal system in contributing to safety of the overall system*** (Section 3.2.2.5)[13]. There is not a high-level requirement in the GRA that deals specifically with barrier performance; however, there are statements throughout the GRA requiring us to assemble a multi-factor safety case, and our safety case relies on the use of multiple barriers and safety functions. Therefore, we need to discuss the role and performance of individual barriers with regard to the safety functions we expect them to fulfil, and how the barriers work together to provide safety.
- ***Other specific assessment issues, such as criticality safety*** (Section 3.2.6). There is a range of other specific assessment issues we will need to consider, including those discussed by the environmental regulators in Section 7.3 of the GRA [1]. In the generic ESC, we focus on our assessment of criticality safety because of its importance in cutting across the TSC, OSC and ESC. In the ESC we need to demonstrate that the possibility of a local accumulation of fissile material

[13] Strictly speaking, the term safety assessment only refers to an assessment of the potential health impacts on people from the GDF. We therefore often use the broader term "performance assessment" to mean an assessment of some limited part (e.g. a barrier) of the geological disposal system that contributes to overall performance, or the assessment of overall system performance (e.g., dose/risk). The overall performance of the geological disposal system is made up from contributions from the different barriers/components and the need to show an understanding of how the system works is as important as any calculated safety measure.

such as to produce a nuclear chain reaction is not a significant concern. We are also investigating, as a 'what-if' scenario, the impact of a postulated criticality event on the performance of the geological disposal system [21].

All these assessments may involve the use of quantitative modelling (calculations) to a greater or lesser extent and there is the potential for considerable overlap between them. For example, assessments of radiological risks to the public and to the environment require the same calculations of radionuclide releases from a GDF. Therefore, although the assessments are concerned with different performance measures or endpoints, we will apply a consistent strategy in the development of all our assessments. This will assist in transparency and consistency across the assessments and promote the efficient use of resources in areas such as model development.

As illustrated in Figure 1.4, our safety assessments have three key strands: the use of multiple lines of reasoning including qualitative safety arguments, the management of uncertainty, and quantitative modelling. Also, our assessments consider additional performance measures beyond radiological dose and risk that can be used to evaluate the performance of a GDF as well as the performance of individual barriers, e.g. radionuclide fluxes.

In the following sections, we describe the strategies for developing our Operational Environmental Safety Assessment (OESA) and our Post-closure Safety Assessment (PCSA). It should be noted that the PCSA contains more detail than the OESA. This is because many of the assessment approaches for the operational environmental safety of a GDF can be similar to those used for other types of nuclear facility in the UK (e.g. waste stores), even if the issues of importance to the assessment results differ between facilities. In contrast, there is no precedent in the UK for evaluating the post-closure safety of a facility for higher activity radioactive waste; this evaluation is more challenging because the assessment timescales are significantly longer than for the operational period and, once a GDF is closed, the options for taking mitigating actions are much reduced. Therefore, previous research and assessment work in the UK has focused more on the challenges inherent in evaluating post-closure safety. The PCSA is also likely to be more important than the OESA in comparing sites having different geological environments and for designing suitable engineered barrier systems.

For this generic ESC, the OESA [26] focuses on GRA Requirement R5, and the PCSA [27] focuses on GRA Requirements R6 and R7. However, we also refer to other assessments (e.g. environmental radioactivity, non-radiological hazards) as appropriate. In future updates of the ESC, these assessments will form part of the OESA and PCSA.

The assessment strategy presented here will be applied for assessments to support future updates of the ESC, as described below. However, only parts of it have been applied in this generic ESC. The way assessments might be applied to different stages of a programme is described in [92].

3.2.1 GRA Requirement R5 – dose constraint during the period in which an environmental permit is held

3.2.1.1 OESA: overall approach

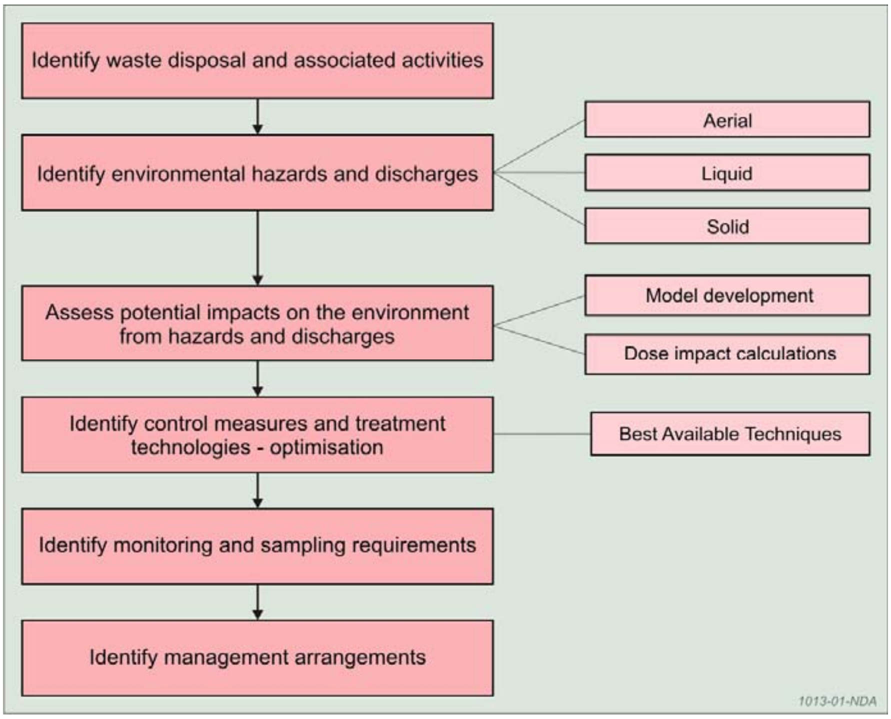
The operational performance of a GDF needs to be assessed against GRA Requirement R5 for effective doses to a representative member of the critical group (Section 3.2.1), Requirement R9 for protection of the accessible environment (Section 3.2.4), and Requirement R10 for protection against non-radiological hazards (Section 3.2.5). This Section focuses on Requirement R5.

There is a higher level of certainty associated with the OESA [26], compared to the PCSA [27]. Many of the environmental hazards and permitted discharges associated with a GDF are anticipated to be comparable to (or lower than) those associated with existing nuclear facilities in the UK, and there is considerable experience of assessing discharges from nuclear sites around the UK. Major accident scenarios that would be considered outside the bounds of normal operating conditions (e.g. vault collapse) are outside the scope of the OESA for the purpose of demonstrating compliance with the GRA. Therefore, the need for scenario development is reduced compared to the PCSA. However, accident scenarios are considered as part of the OSC [6].

The approach to the OESA is shown in Figure 3.6, recognising that there is significant overlap with the OSC, which assesses doses to workers and risks to the public from accidents. The process shown applies to potential radiological and non-radiological discharges to the aerial, aquatic and terrestrial environments. We will assess the radiological impacts of any discharges, considering both humans and non-human biota (Section 3.2.4). 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.

Figure 3.6 Operational Environmental Safety Assessment process

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



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

3.2.1.2 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 [93], 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 [94], 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 [95]. The first EMRAS programme concluded its activities in 2007, and a second programme is now running in the period 2009-2011.

In Section 5.1.1, 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 5.1.1 is at a level of detail appropriate to this generic stage of a GDF implementation programme, and is supported by Appendices A, B and C and 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.

3.2.1.3 OESA: management of uncertainty

The management of uncertainty is discussed in a general manner with regard to the PCSA in Section 3.2.2.3. The approaches described there are also broadly relevant to the OESA and we do not repeat that text here. 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.

3.2.1.4 OESA: quantitative assessment

The general steps in developing quantitative assessment models are discussed in a general manner with regard to the PCSA in Section 3.2.2.4. The approaches and general modelling issues described there are also broadly relevant to assessing the radiological impacts of discharges during the operational period, and we do not repeat that text here.

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) [96] 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.

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 assessment of public doses from operating nuclear facilities [97, 98]. For the purpose of assessing doses to members of the public from radionuclides discharged during the operational period, a GDF can be considered similar to any existing nuclear licensed site, and we therefore make use of the dose assessment guidance set out in [97, 98]. This guidance recommends a staged approach to the assessment of critical group doses. 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. This approach has been followed for the generic OESA [26].

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.

3.2.1.5 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.

3.2.2 GRA Requirement R6 – risk guidance level after the period in which an environmental permit is held

3.2.2.1 PCSA: overall approach

The post-closure performance of a GDF needs to be assessed against GRA Requirement R6 for radiological risk (Section 3.2.2), Requirement R7 for consequences of human intrusion (Section 3.2.3), Requirement R9 for protection of the accessible environment (Section 3.2.4), and Requirement R10 for protection against non-radiological hazards (Section 3.2.5). This Section focuses on Requirement R6.

The main objective of the analysis of post-closure performance and safety is to demonstrate a clear understanding of the geological disposal system and its potential future evolution. The analysis considers the expected evolution of the geological disposal system and less likely events that might affect the performance of the system. We have the same three key strands to our post-closure safety analysis: qualitative assessment, the management of uncertainty, and quantitative assessment. The strategy underlying the development of these strands is described below.

Our experience in developing performance assessments builds upon a series of site-specific and generic assessments of a GDF undertaken since the early 1990s by Nirex [e.g. 80, 99, 100]. We are also participating in international expert groups on post-closure assessment, for example the committees and working groups of the NEA, and we are familiar with overseas performance assessment programmes through international groups and co-operation under bi-lateral agreements.

3.2.2.2 PCSA: qualitative assessment

Our demonstration of understanding of the geological disposal system in the post-closure period comes from a variety of sources:

- Key inputs come from our own R&D programme and from previous R&D work by Nirex to develop understanding of the behaviour of geological disposal systems. As we move forward, we expect further understanding to be developed from iterative feedback between our R&D, design, site characterisation, and assessment activities.
- Experience gained in other waste disposal programmes and in other relevant industries (e.g. mining, hydrocarbons) provides support for the development of assessment models. For example, we participate in experiments in underground research laboratories overseas, and this provides us direct access to *in situ* testing programmes.
- Our own work, previous work by Nirex, and work by other waste management organisations on natural and anthropogenic analogues of the geological disposal system and its materials and environment provide an input to developing the quantitative models and parameter values, and build confidence in the modelling results.
- Research by the wider scientific community on similar past and present natural systems can also be used to build confidence in our understanding of the expected evolution of a GDF and its ability to protect people and the environment.

In Section 5.2.1, we use this understanding to summarise qualitative supporting arguments based on safety functions for the post-closure safety of a GDF, and to outline how we would build confidence in quantitative assessments of post-closure safety through a variety of lines of argument, including results from R&D, long-term demonstration experiments in underground rock laboratories overseas, the use of site-specific natural indicators of safety, and the use of archaeological and natural analogues of components of the geological disposal system. The discussion in Section 5.2.1 is at a level of detail appropriate to this generic stage of a GDF implementation programme, and is supported by Appendices A, B and C and our research status reports.

Progressive development of the ESC will ensure that the information available for the next assessment cycle is fit for the purpose of both qualitative and quantitative assessment.

3.2.2.3 PCSA: management of uncertainty

Uncertainty in our environmental safety assessments for a particular geological disposal concept arises from an incomplete knowledge or lack of understanding of the behaviour of engineered systems, physical processes, and site characteristics. We can never completely resolve all such uncertainty, so we need to find ways to manage uncertainty, including 'designing out' uncertainty as we optimise the design of our preferred disposal concept (see Section 3.1.2). Another area of uncertainty concerns the types and quantities of waste we will eventually have to manage in a GDF.

There are several ways in which such uncertainties can influence the results of a post-closure performance assessment. Different classification schemes for types of uncertainty are possible, but the one we use, consistent with international good practice [101], is as follows:

- Uncertainty concerning the representation of processes in models and computer codes to represent the geological disposal system. This type of uncertainty is often called "model" uncertainty.
- Uncertainty associated with the values of the parameters that are used in the implemented models. These are variously termed "parameter" or "data" uncertainties.
- Uncertainty in future states of the geological disposal system. These are often referred to as "scenario" or "system" uncertainties.
- Uncertainty in future human behaviour (actually a subset of "scenario" uncertainty, but the GRA calls for separate treatment of uncertainties associated with future human intrusion, see below).

The classes of uncertainty are related to each other, and particular uncertainties can be handled in different ways, such that they might be dealt with in one class or another for any single iteration of a performance assessment, depending on programmatic decisions (e.g., on how best to implement assessment calculations or to communicate results) and practical limitations (e.g., on funding or timescales). We also note that while many uncertainties are knowledge-based and therefore reducible in nature (e.g. the value of a corrosion rate), others may be more random and therefore essentially irreducible by further information gathering (e.g., the likelihood of a seismic event of a particular magnitude, or of a particular container being the first one to lose its integrity).

There may be substantial uncertainty associated with the future of the geological disposal system and this uncertainty increases with increasing time. We have developed a methodology for addressing this uncertainty in a systematic way, based on the analysis of all features, events and processes (FEPs) potentially relevant to the evolution of the geological disposal system and the development of scenarios based on these FEPs. A scenario specifies one possible set of events and processes. Different scenarios are specified to provide illustrations of fundamentally different evolutions of the geological disposal system. The use of scenarios enables us to meet the GRA [1] requirement that unquantifiable uncertainties should be kept apart from, and given separate consideration to, quantifiable uncertainties. Our scenario development methodology is described in more detail in Section 3.2.2.4.

For a given scenario, our strategy for handling the uncertainty within the bounds set by the events and processes included in the scenario falls into the following broad categories:

1. Demonstrating that the uncertainty is not important to safety because, for example, safety is dominated by other processes.
2. Addressing the uncertainty explicitly, usually using probabilistic techniques for parameter uncertainties, and showing that the expected situation is acceptable. However, in the case of model uncertainties, where we may be unable to determine preferences or weights associated with the two models, we are more likely to undertake separate assessments using each of the two models as a means of understanding the impact of the uncertainty.
3. Bounding the uncertainty and showing that even the bounding case gives acceptable safety.
4. Ruling out the uncertainty, usually on the grounds of very low probability of occurrence, or because other consequences were the uncertain event to happen, would far outweigh concerns over performance of the geological disposal system, e.g. a direct meteorite strike of sufficient magnitude to disrupt a GDF.
5. Agreeing with the regulators a stylised approach for handling a specific uncertainty, e.g. use of “reference biospheres”.

By using a combination of the above, we can clearly demonstrate how we have treated the uncertainty associated with the models and parameter values on which the performance assessment is based. A safety assessment model can be said to be **robust** when the key performance measures calculated with the model are relatively insensitive to remaining uncertainties. There should be no ‘cliff-edge’ effects where credible changes in a modelling assumption or the value or range of values of a particular input parameter change a decision that would be made based on the results of the assessment. This is essential to provide confidence in the assessment results, and will be needed to satisfy regulatory and public scrutiny.

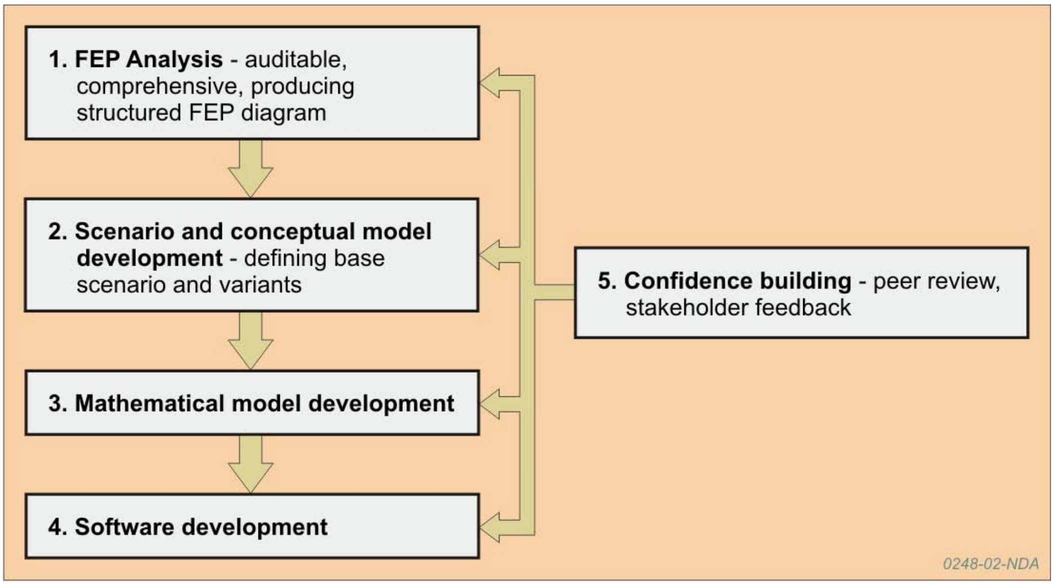
The identification and analysis of those uncertainties to which the performance of the geological disposal system is most sensitive helps us to identify areas where further data acquisition and development in scientific understanding are needed. As illustrated in Figure 1.1, iterative assessment and analysis of key uncertainties helps provide focus in the development of our R&D and site characterisation programmes (see Section 6.3), and can guide us in determining how much we need to know. It also helps with design development, as discussed in Section 3.1.3. In particular, as already noted, we may be able to ‘design out’ uncertainties that would otherwise undermine the robustness of our safety assessment modelling. We will establish and maintain a register of significant uncertainties as our programme develops (see Section 5.3).

3.2.2.4 PCSA: development of quantitative assessment model

There are five steps in development of a quantitative performance assessment model of a given geological disposal system and its implementation in a computer code (Figure 3.7). The five steps are explained below. After the model is assembled, a structured set of calculations is run to explore the behaviour of the geological disposal system and to calculate various performance measures (not shown in Figure 3.7). The types of calculations that may be performed are considered in Section 3.2.2.5. Here we focus on the construction of the model itself and the parameterisation of the model.

Figure 3.7 Five-step approach to development of a quantitative performance assessment model

Feedback loops between the different steps exist, but are not shown. The use of data in modelling (not shown in the figure) is also discussed in this section. We have undertaken all aspects of this approach for this generic ESC, except that we have not produced a FEP analysis specifically for this generic ESC and have only assessed a limited range of scenarios appropriate to the generic stage of a GDF implementation programme.



The performance assessment process begins with wide-ranging consideration of FEPs that could affect the long-term environmental safety of a GDF. This can be done by identifying FEPs that are relevant to the performance of a particular disposal concept, or using existing international FEP lists. The intention is to analyse systematically all potentially relevant FEPs, thereby ensuring that the performance assessment is comprehensive in its coverage.

Nirex previously developed a concept-specific FEP list for disposal of ILW in a higher strength rock environment, based on review of international FEP lists for disposal of ILW, HLW and SF, and expert analysis [102]. This work was reviewed favourably by the NEA [103], and we therefore intend to make use of this earlier work. Furthermore, we note that many national and international programmes have developed FEP lists for a wide range of waste types, but few substantially new FEPs have been identified in the last 10 years or so. This provides confidence in the comprehensiveness of the earlier FEP work.

Features, events and processes (FEPs) that could affect environmental safety

We have not undertaken a new FEP analysis specifically for this generic ESC. We will review our FEP list once we are working with specific sites and designs. It is possible that new FEPs would need to be added to the FEP list depending on the waste types in a GDF, the disposal concept, and the geological environment.

We categorise FEPs into one of two categories, depending on their assessed likelihood of occurrence within the assessment timescale:

- **System FEPs** are those that are certain to exist or occur during the timescale of the assessment. That is, their deemed existence or occurrence can be characterised by a probability of unity. Examples of system FEPs include radioactive decay and sorption.
- **Probabilistic FEPs** may or may not exist or occur during the timescale of the assessment. That is, their deemed existence or occurrence can be characterised by

a probability of less than unity. Examples of probabilistic FEPs include major seismic events, inadvertent drilling into a GDF, and early degradation of shaft seals.

We also decide which FEPs are potentially important for safety and for inclusion in the assessment calculations. Recording which FEPs are represented in the assessment calculations and which are not, together with the reasons for their exclusion, provides a clear framework for auditing and review of the assessment by model developers and by regulators and other stakeholders.

Scenario and conceptual model development

Scenario development

Scenarios can be considered as broad descriptions of alternative futures of the geological disposal system. Multiple scenarios may be defined where it is not possible or convenient to describe the system using a single integrated model. The main objectives for scenario development are [104]:

- to demonstrate or try to ensure completeness or sufficiency in the scope of an assessment;
- to decide which FEPs to include in a performance assessment and how to treat them;
- to provide traceability from data and information to the assessment scenarios, models, parameter values, and calculation cases;
- to structure the assessment calculations;
- to promote transparency and improve comprehensibility of the assessment and the assessment results; and
- to guide decisions concerning future work.

The first two of these objectives are covered as part of the FEP analysis. A systematic analysis of the FEPs relevant to the performance of the geological disposal system leads to the identification of a **base scenario**, which is a description of the likely evolution of the geological disposal system, and a number of variant scenarios that represent potential deviations from the base scenario. The base scenario considers potential gas-mediated and groundwater-mediated radionuclide releases, and includes all “system FEPs”. **Variant scenarios** are triggered by probabilistic FEPs that are not certain to occur, but if they do, may affect GDF performance. Examples of such variant scenarios are early failure of engineered barriers, criticality, and alternative climate change patterns. FEPs related to inadvertent human intrusion are also treated as a variant scenario, consistent with the specific GRA requirements for these scenarios.

We will make use of the scenario development methodology published by Nirex [105, 106]. Although this scenario development methodology was established before the revision of the GRA in 2009, the approach of assessing a base scenario and variant scenarios meets the GRA requirement that unquantifiable uncertainties are kept apart from, and given separate consideration to, quantifiable uncertainties in safety assessment. Although we have not undertaken a full scenario analysis specifically for this generic ESC, we have assessed a base scenario and discuss several variant scenarios that are relevant to a GDF in any geological environment (e.g. human intrusion).

Scenario development needs to take account of the timescales over which the assessment calculations are performed. These timescales are chosen such that we can demonstrate our understanding of the peak risks associated with a GDF. In our quantitative work, we generally present calculated impacts out to one million years. A period of one million years is far beyond the timescale considered in evaluating other kinds of development project, although calculations considering such a period are not unique to radioactive waste disposal. For example, geology, astronomy and climatology consider similar and longer durations – although this is for the purpose of scientific research, not for making comparisons with regulatory criteria. Uncertainty about the environmental safety performance of the geological disposal system increases with increasing time. Given the uncertainties in the biosphere and the stylised approach necessary to evaluate impacts to potentially exposed groups in the post-closure period (see conceptual model development below), calculated impacts for the post-closure period are only ever illustrative. However, the calculated impacts are nonetheless increasingly uncertain for times so far into the future. We consider that beyond about one million years, the uncertainties are so great that quantitative assessments generally cease to be particularly meaningful. Therefore, it is important that we support our calculations with other arguments for safety in the far future, for example comparisons with natural radiation levels, complementary safety indicators, and bounding analyses, and information from natural and archaeological analogues (see Section 3.2.2.2).

Conceptual model development

Conceptual and mathematical models are developed for each scenario and are implemented in software. A conceptual model has been defined by the IAEA [107] as:

> *"A set of qualitative assumptions used to describe a system (or part thereof). These assumptions would normally cover, at a minimum, the geometry and dimensionality of the system, initial and boundary conditions, time dependence and the nature of the relevant physical, chemical and biological processes and phenomena."*

It is important that conceptual models reflect the available data and understanding of the physical, chemical, and biological processes affecting the geological disposal system. Conceptual model assumptions should be consistent with one another and with existing information within the context of the given modelling purpose.

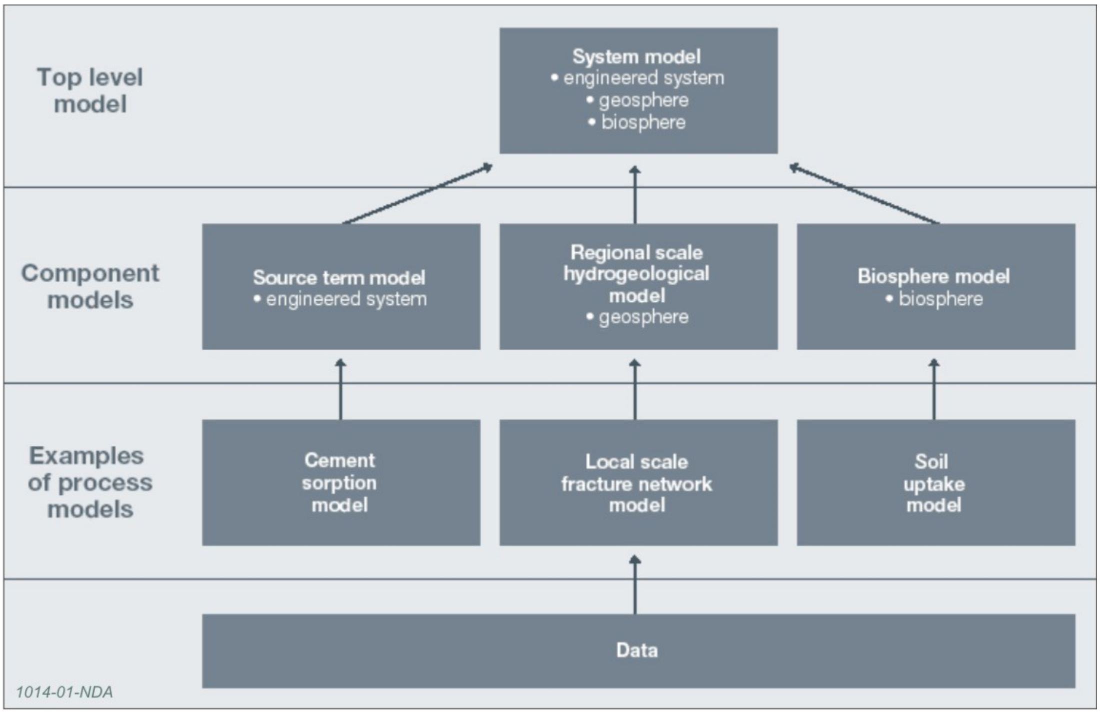
Conceptual models will be developed to represent important aspects of the different components of the system, including:

- the engineered system, comprising the excavated disposal areas and their contents, including the waste materials, waste packaging, buffer/backfill and structural materials, and shafts and access tunnels;
- the geosphere, comprising the rocks in which a GDF is constructed and those that surround them, extending to the surface (see, for example, the list of site descriptive models in Section 3.1.5.1); and
- the biosphere, comprising the environment accessible to humans, including the soil and surface rocks, surface water bodies, oceans and the atmosphere.

A conceptual model also needs to consider how different FEPs interact with one another and how the geological disposal system will evolve with time. In order to build confidence in our modelling of all the different FEPs in each scenario, we need to use a variety of models, on different scales and at different levels of detail. This gives rise to a 'hierarchy' of models, as illustrated in Figure 3.8. This figure is linear; we also need to understand process couplings and interactions between the different components of the geological disposal system, as discussed below.

Figure 3.8 Examples of different levels of model in our model hierarchy

As an example, a model representing the sorption of radionuclides onto cement in the disposal vaults feeds into a higher-level model representing the source term, which in turn supplies information to a model of releases from the engineered system, which forms part of our safety assessment modelling of the entire geological disposal system.



At the highest level of the model hierarchy is a total system model, which is used to calculate performance measures such as radiological risk. Supporting the total system model, and supplying understanding or data to underpin it, may be more detailed process-level models of specific aspects or components of the system. The understanding gained from these process-level models needs to be sufficiently represented in the total system model to enable a calculation of overall system performance that is fit for purpose. However, complicated process-level models can only be justified if there are data and understanding to support them. Therefore, we anticipate that the complexity of our supporting models may increase as our understanding develops and we gather more data with which to parameterise them. However, we recognise that there is a trade-off between increasing model complexity and the uncertainty and reliability of results. We will maintain simple models to the extent reasonably practicable, in parallel with the growing complexity of the supporting models, so that the outputs from complex models can continue to be interpreted in relatively simple terms.

As understanding develops and we gather site-specific and concept-specific data, we may be able to make greater use of detailed supporting models that explicitly consider couplings between key chemical, mechanical, hydrological, thermal, radiological and biological processes, to underpin our total system model. For this generic ESC, we did not consider it appropriate to undertake an integrated suite of calculations at different levels of process complexity. However, our research status reports present conceptual models at a more detailed level.

Our development of conceptual models for the performance assessment will be integrated with our R&D programme and, once site investigations have commenced, the development of the site descriptive models discussed in Section 3.1.5.1. In some cases, there will be uncertainty over the most appropriate model representation. For example, particularly in the early stages of site characterisation, there may be significant uncertainty concerning groundwater flow in the available geological environment(s) and the routes by which radionuclides released from the EBS could be transported to the accessible environment.

We would expect such uncertainty to be progressively reduced as our understanding of the site increases. We will evaluate model uncertainty through implementation and analysis of alternative conceptual models within the performance assessment framework. Possible means of doing this are discussed in [101]. These alternative conceptual models would all represent our system FEPs, but could include different potential pathways to take account of uncertainty in geosphere parameters.

The possible future evolution of the surface environment is particularly difficult to constrain in models, and the way in which people living in the far future could be exposed to radioactivity released from a GDF is highly uncertain. For assessments of the present-day situation in the OESA, a critical group can be defined that is representative of those individuals in the population expected to receive the highest dose equivalent on the basis of activities that can be currently observed to occur. The critical group is considered throughout the period in which an environmental permit is held, during which human activities in relation to a GDF can be monitored and controlled if necessary. However, after the environmental permit is surrendered, when active institutional control of a GDF has ceased, any number and manner of developments of the surface environment and human interactions with it are possible. Some of this uncertainty is dealt with through the development of variant scenarios, such as those to address inadvertent human intrusion of a GDF. Within the base scenario, uncertainty about biosphere evolution is addressed through consideration of a number of potentially exposed groups (PEGs) whose behaviour and the environment they interact with is **stylised** to illustrate the dose / risk that could be received through a particular lifestyle or activity. A stylised approach involves the use of a set of assumptions that are either generally reasonable or clearly conservative, and is accepted by the environmental regulators in the absence of specific information about what would actually happen in the far future. Consistent with international guidance [108], we define the potentially exposed groups on the basis of a series of **reference biospheres**, each of which can be considered as an alternative conceptual model for the biosphere.

Potentially exposed groups have to be chosen so that the range of potential exposure routes is covered, and the habits are consistent with the geological environment under consideration. The potential exposure routes we consider include drinking water, ingestion of crops, ingestion of animal products and fish, inhalation of dust and direct external irradiation. It may be appropriate to define more than one potentially exposed group for use in an assessment to represent the range of potential exposure routes and their variation with time. As we learn more about a GDF site, we would underpin the definition of our potentially exposed groups with more sophisticated analyses, possibly including catchment-scale modelling.

For this generic ESC, the assumptions made concerning the surface environment are consistent with this approach but the modelling of each scenario is simplified to consider only a single exposed group based on an adult member of a farming community making reasonably maximised use of local resources, including a well drilled into a local aquifer. The potentially exposed group is assumed to live in the area where there is the highest concentration of radionuclides discharging to the biosphere via the groundwater pathway

Mathematical model development

Once the conceptual models are developed, we can use mathematical equations implemented in a computer code to represent the processes included in the conceptual models. In this way we arrive at a means of undertaking a quantitative assessment of performance.

As discussed above, we consider that there is value in modelling the system at various levels of complexity. In terms of the development of mathematical models, this may mean, for example, that for a given conceptual model of a particular system or component of that system, understanding can be improved by use of simple analytic expressions as a complement to complex numerical modelling. An approach to doing this is illustrated in [100, Volume 3, Section 8]. A key objective of our model development is to bring together data and understanding at different levels of detail to represent the FEPs identified in the scenario development process at a suitable level of complexity in terms of physics, chemistry, length-scale, and time.

However, for this generic ESC, the approach to undertaking PCSA calculations is relatively simple, and we did not consider it appropriate to undertake calculations based on different levels of complexity.

Software model development and verification and model validation

We use a variety of software to undertake quantitative modelling, driven by our modelling objectives and the type of mathematical representation of the conceptual model. Some functionality may be provided by software packages we use and other functionality is implemented by ourselves.

When we implement a mathematical model in a computer code, we need to test the computer code under a suitable quality regime to confirm that the code correctly implements the mathematical model – this is termed **software verification**. Ideally, verification is supported by independent implementation of a model in more than one code or by more than one modelling team. To this end, we participate in international code comparison and benchmarking work (e.g. [109]).

Unlike software verification, **model validation** applies to the whole model development cycle and represents a *process* of determining whether the mathematical model and its implementation in a computer code provides an adequate representation of the real system for a given purpose. For models at the lower level of the model hierarchy illustrated in Figure 3.8, this may be possible through comparison of model predictions with laboratory and/or field data from our R&D and site characterisation programmes. These programmes can also be used to collect data that help distinguish between alternative modelling assumptions. For models at the higher level of the model hierarchy, model validation processes must rely on other approaches, because it is not possible to conduct experiments or collect field data over the timescales for which we run assessment models. These processes involve demonstrating that the assessment model is fit for purpose by using the methods outlined in Section 3.2.2.3, and by showing how the assessment model provides a reasonable or conservative representation of the processes included at lower levels of the modelling hierarchy (and which may be subject to more traditional forms of model testing).

It is not possible to state the particular computational tools that will be used for the PCSA in later stages of the MRWS Site Selection Process, as such tools are being continually developed and updated owing to their use within and outside the nuclear industry. We are likely to conduct system assessment calculations using industry-standard tools that are available at the time each update of the PCSA is conducted. However, we may continue to develop our own tools for the conduct of some types of supporting calculations where there are issues to be addressed specific to the wastes we have to manage and the disposal concepts we are considering (e.g. for criticality safety assessment, gas generation and migration).

Use of data in modelling

The validity of modelling depends not only on the quality of the models, but on the validity of the parameter values (and uncertainty ranges) used in the modelling and the data from which these values (and uncertainty ranges) have been derived. Data and model parameter values need to be fit for purpose, and our quality assurance procedures for data acceptance will ensure the purpose of the calculations, the conditions being modelled and the spatial and temporal scope are considered. Acceptance checks for parameter values used in modelling will consider the provenance of data on which these values are based and any limitations associated with that provenance. Our quality management procedure RRWI30 [110] covers such issues, and we have applied this procedure to the conduct of the PCSA calculations [27] for this generic ESC – also see Section 3.3.5.

The parameter values used in model calculations will be recorded using a form-based procedure where appropriate. We will record the following kinds of information: parameter name and definition; units (SI units will be used where possible); model, code and context; available data and references; recommended value and justification; assigned uncertainty or probability distribution function; dependence on other parameters (correlation); and applicability. For the PCSA calculations that form part of this generic ESC, such information is recorded within the model implementation of our computer software, and this model implementation can be made available to reviewers.

In the future, separate parameter databases might be developed for other purposes, e.g., those detailing geological, chemical, and/or biosphere parameter values. The databases can draw initially on existing national and international databases, but will eventually need to be tailored to UK wastes and conditions. As our programme develops, these databases will contain more site-specific parameter values, and we will have less need to rely on generic values. The site-specific parameter databases developed to support assessment models will link to any data management systems developed to support our site investigation programme.

We use a formal expert elicitation methodology involving groups of experts to build confidence in the most significant parameter values used in our modelling. We do not have a specific management procedure governing such expert elicitation, but we do make use of a formal protocol that was developed by Nirex [111] and which is consistent with international good practice, e.g. as set out in [112]. For example, we recognise the importance of recording the rationale behind the advice provided by experts, and the need to propagate any uncertainties through the calculations in which the parameters are used.

In this generic ESC, we have not considered it necessary to make use of formal expert elicitation for those parameter values relating to the geological environment used in the PCSA calculations (discussed in Section 5.2.2). At this generic stage in the implementation programme for a GDF, these parameters are assigned a range of values appropriate to the different geological environments we need to address.

Confidence building

As discussed in Section 3.2.2.3, building confidence in the quantitative modelling process is focused mainly on demonstrating that we have identified the relevant uncertainties, and that the assessment results are **robust to uncertainty**.

The model development process uses a systematic methodology with a comprehensive FEP analysis and structured treatment of uncertainty, and hence itself goes some way towards this objective. The software verification and model validation described above are also key factors. The development of quality-controlled parameter databases containing clear explanations of the basis for parameter values and parameter value uncertainty is another means of building confidence.

Beyond this, we will use peer review of key parts of the process and of the overall ESC documentation, and formal independent scrutiny (e.g. reviews of our work by the regulators) to review the coverage and presentation of the PCSA and to provide feedback into future updates. Drafts of this Generic ESC main report and the supporting safety assessments have been subject to formal external peer review and regulatory review.

Further details on how we build confidence in the post-closure safety assessment (to which these models contribute) is provided in Section 5.2.1.3.

3.2.2.5 Using the PCSA

We use our post-closure modelling to calculate a range of key **safety indicators**. A safety indicator provides a measure of the overall performance of the geological disposal system with respect to a specific safety aspect, in comparison with a reference value that can be shown to be safe, or is at least commonly considered to be safe. Safety indicators that we will consider using include:

- radiological dose rate and risk over time (the primary regulatory safety measures);
- concentration of radiotoxic or chemically toxic elements in the biosphere over time; and
- radiotoxicity flux from the geosphere to the biosphere over time.

These indicators are explained and examples of their use provided for a range of geological environments and disposal concepts in [113].

We present radiological risk over time for the PCSA calculations [27] conducted to support this generic ESC (see Section 5.2.2.2 of this report).

We also use our post-closure modelling to calculate a range of key **safety indicators**. A safety indicator provides a measure for the performance of a system component or several components to support the development of system understanding.

Safety indicators that we will consider using include:

- activity over time in components of the geological disposal system;
- radiotoxicity inventory over time in components of the geological disposal system;
- radiotoxicity flux over time from components of the geological disposal system (e.g. from a GDF to the geosphere);
- cumulative radiotoxicity fluxes up to given times from components of the geological disposal system;
- containment factor, defined as the activity released from a GDF divided by the activity disposed, over time; and
- transport time through components of the geological disposal system.

These indicators consider such things as how, and in what quantities, the radionuclides move through the system of multiple barriers with time. These indicators are explained and examples of their use provided for a range of geological environments and disposal concepts in [113]. Other examples of possible safety function indicators are given in our research status reports. We present radionuclide flux over time for the PCSA calculations [27] conducted to support this generic ESC.

We undertake analyses to evaluate the sensitivity of the quantitative model results to modelling assumptions and parameter values. As well as building confidence in the robustness of the results and the assessed safety of the system, these sensitivity analyses are important in developing our understanding and identifying those areas where we most need to reduce uncertainty and improve our understanding in our future work programme.

The post-closure safety analysis also feeds into our optimisation and optioneering work described in Section 3.1.2, so that appropriate decisions are taken at various stages of the programme.

At the desk-based study stage of the MRWS Site Selection Process outlined in Section 2.1, our post-closure safety analysis capability would need to support a determination of whether the geological environment at candidate sites has the potential to host a GDF and is worthy of surface-based investigations.

Prior to site investigations, any assessments would be based only on the already available data for the sites under study. If more than one candidate site is available, we would need to deal with the possibility that there could be a difference in the amount of available data for each site. If few data are available for one site, the assessment may appear simplistic compared to that for a site with a lot of available data. The assessment results for such a site could be subject to a greater degree of uncertainty, compared to that for a site with a lot of available data. However, our confidence in our ability to develop a disposal concept for a GDF appropriate to the geological environment that is likely to meet environmental safety requirements is unlikely to be related only to the current availability of data at a candidate site, so that sites having few data could still be considered favourably in the desk-based studies stage. We also note that while our assessments will consider the degree of difficulty in establishing an ESC at each site, this must be balanced against other factors, as discussed in Section 2.1, and it will be up to the UK Government to decide which site(s) to take forward to the next stage of the MRWS Site Selection Process.

3.2.3 GRA Requirement R7 – human intrusion after the period in which an environmental permit is held

The depth of disposal is expected to put a GDF beyond the reach of many types of human activity. However, although human intrusion into a GDF may be regarded as highly unlikely, it is not impossible.

As part of the PCSA, we will perform assessments of possible future human intrusion. GRA Requirement R7 indicates that uncertainties associated with human intrusion after the period in which an environmental permit is held should be treated separately [1]. Therefore, as required by the GRA, we assume that human intrusion after this period is highly unlikely to occur; however, we will consider and implement any practical measures that might reduce this likelihood still further. Implementation of such measures would be subject to agreement with the environmental regulators. We will also assess the potential consequences of human intrusion after the period in which an environmental permit is held.

Human intrusion scenarios are classified in GRA Requirement R7 [1] as:

1. intrusion with full knowledge of the existence, location, nature and contents of the disposal facility;
2. intrusion without prior knowledge of the disposal facility, e.g. drilling a well or exploratory drilling for mineral resources based on an assessment of the local geology; and
3. intrusion with knowledge of the existence of underground workings, but without understanding what they contain, e.g. an archaeological investigation carried out without knowledge or understanding of radioactivity.

We will develop variant scenarios to serve as a basis for assessing the second and third classes of human intrusion. The environmental regulators do not expect us to consider the first class because they take the view that ([1], paragraph 6.3.38): "*… a society that preserves full knowledge of the disposal facility will be capable itself of exercising proper control over any intrusions into the disposal system*."

For the PCSA calculations [27] conducted to support this generic ESC, we have undertaken a limited number of scoping calculations of human intrusion.

3.2.4 GRA Requirement R9 – environmental radioactivity

We are currently developing our approach to the assessment of radiological impacts on the environment (non-human species - GRA Requirement R9), as discussed in the Biosphere status report [18] and in our Radiological Protection Policy Manual [73]. This is an area where discussions with regulators are needed to agree an acceptable approach and, for this reason, we have not done much work on this topic for this generic ESC. We propose to use methodologies developed by the Environment Agency [90], and within the context of international studies on this topic - in particular, the EC Project “Environmental Risk from Ionising Contaminants: Assessment and Management” (ERICA) [91].

As an example of our proposed approach, an assessment of potential doses to non-human biota during the operational period of a generic GDF is included in the OESA [26]. This is based on consideration of a set of reference organisms. The work we have done to date on assessment of the potential radiological impacts on non-human biota during the post-closure period of a generic GDF is reported in [114] and is summarised in the Biosphere status report [18], and follows a similar approach. Both of these studies indicate that no unacceptable impacts on non-human biota would arise from the disposal of higher activity radioactive wastes in a GDF.

3.2.5 GRA Requirement R10 – protection against non-radiological hazards

We are currently developing our approach to the assessment of non-radiological hazards or chemotoxic impacts (GRA Requirement R10), as discussed in the Biosphere status report [18]. This is also an area where discussions with regulators are needed to agree an acceptable approach and, for this reason, we have not done much work on this topic for this generic ESC. We intend to examine current conventional waste management regulations as a basis for developing an approach [115]. If the wastes being disposed of present any non-radiological hazards requiring assessment, we will modify our radiological assessment tools for this purpose, or use existing tools from the hazardous waste industry such as LANDSIM [116].

Our preliminary assessments of the non-radiological impacts from the disposal of higher activity radioactive wastes in a GDF are reported in [117, 118] and are summarised in the Biosphere status report [18]. We understand the issues and, in developing our assessment approach further, the toxicological properties of the following substances are considered to be worthy of detailed consideration: beryllium, cadmium, chromium, lead and uranium. Our work to date indicates that no unacceptable impacts are expected to arise, and points out the importance of considering the synergistic effects of chemical mixtures and exposures to both chemical and radiological hazards. However, no new calculations have been undertaken specifically for this generic ESC.

3.2.6 Specific assessment issues

We discuss criticality safety separately here as a specific assessment issue because of its importance in cutting across all the DSSC Tier 1 safety cases, and the large amount of work we have already undertaken on this topic to support our disposability assessment process.

In future updates of the ESC, we may discuss other specific assessment issues that we regard as being of particular importance from a methodological point of view. The set of issues considered will include those discussed by the environmental regulators in Section 7.3 of the GRA (“Additional considerations”) that are not otherwise covered elsewhere in the ESC. For example, future updates of the ESC could include here (as a new subsection) a fuller discussion of our approach to the treatment of climate change, the various ways in which expert judgement has been used in our assessments, and more detail on the various means of building confidence in our models.

3.2.6.1 Criticality safety

If enough fissile material is brought together during transport or waste management operations, a criticality could occur, potentially releasing dangerous amounts of radiation to anyone in close proximity and, in certain circumstances, producing significant amounts of energy. Therefore, an important component of our work is the consideration of criticality safety issues associated with operations involving waste containing fissile material, and criticality safety assessments are required inputs for the TSC, OSC and ESC. This work is summarised in the Criticality safety status report [21].

For determining package limits on fissile materials, our assessment methods have been developed from well established methodologies used for several decades in the nuclear industry. However, in the context of the ESC addressing the GRA, the criticality safety assessment must also consider the potential for criticality in the post-closure phase, and significant work on ILW has already been conducted by both Nirex and ourselves [21]. For ILW, the series of barriers inherent in the design of a GDF and integral to features of the waste conditioning and packaging will act to make a criticality event very unlikely. However, in the long term, after deterioration of the physical containment provided by the waste packages, the movement of fissile material and its subsequent accumulation into new configurations could, in principle, lead to the occurrence of a criticality event. We therefore consider that there is a need to assess the potential impacts of a criticality event for ILW disposed of in a GDF, even if a criticality event is considered very unlikely to occur [21].

As a result of changes to the potential inventory for disposal in a GDF, and consideration of alternative disposal concepts and geological environments, we have extended our criticality safety assessment work to consider materials other than ILW. In particular, we have started to consider criticality safety in a GDF for separated plutonium and highly enriched uranium (HEU). We understand the issues – although the total amount of fissile material would increase significantly, changing what might be considered to be a bounding post-closure criticality event, the likelihood of a large accumulation may be vanishingly small because of the emplacement strategy [21].

As site-specific data become available, we will review our criticality safety work to confirm that conclusions drawn from analyses at the current generic stage remain valid.

3.3 Management strategy

GRA Requirement R4: Environmental safety culture and management system

The developer/operator of a disposal facility for solid radioactive waste should foster and nurture a positive environmental safety culture at all times and should have a management system, organisational structure and resources sufficient to provide the following functions: (a) planning and control of work; (b) the application of sound science and good engineering practice; (c) provision of information; (d) documentation and record-keeping; (e) quality management.

We recognise that the technical requirements for a robust safety case are unlikely to be met unless the organisation responsible for delivering safety is properly managed and led and has staff of the right calibre. Therefore, our management strategy is an essential component of our overall safety strategy. An important aim of our management strategy is to ensure that our activities are co-ordinated and integrated to ensure a sound basis for decision-making (see, for example, Figure 1.1). Our management strategy must also ensure that we are able to communicate our plans and findings clearly to regulators, to the Community Siting Partnership(s), and to other stakeholders, and take appropriate actions in response to their feedback.

We summarise below how we promote a positive environmental safety culture through consideration of the main themes identified in Requirement R4 of the GRA:

- planning and control of work (Section 3.3.1);
- the application of sound science and good engineering practice (Section 3.3.2);
- provision of information (Section 3.3.3);
- documentation and record-keeping (Section 3.3.4); and
- quality management (Section 3.3.5).

Having our management strategy in place is an important aspect of demonstrating that we are developing the capability to perform the functions of a Site Licence Company (SLC) in due course. The regulators reviewed our development towards SLC status in late 2009 and concluded that we have "…*made significant progress in working towards the status of a 'Prospective SLC'*" [119]. We are currently functioning as a Prospective SLC under voluntary regulatory scrutiny, as discussed in Section 2.3.

3.3.1 Planning and control of work

We discuss below our competence and culture (Section 3.3.1.1), our governance arrangements (Section 3.3.1.2), and our management system documentation (Section 3.3.1.3).

3.3.1.1 Our competence and culture

The DSSC is owned by us, and, therefore, it is important that we demonstrate that we are a capable organisation, suitably qualified and experienced to develop it. We understand the skills that are required, we recruit capable people - and, where needed, experienced people - to provide those skills, and we keep their training up to date and give them wider opportunities to develop their expertise. This approach to developing and maintaining our corporate competence and culture is discussed in more detail below, specifically considering the issues that are identified by the environmental regulators in the GRA.

Capable organisation

We define the requirements for RWMD to be a capable organisation in the Organisational Baseline Document [66]. The Organisation Baseline Document sets out all our responsibilities for nuclear safety and environmental matters, including the DSSC, and shows how these are covered by each of the posts within our organisation. For each post, there is a defined set of safety and environmental competences that are needed in order to discharge the responsibilities. Our human resource management systems ensure that each person's competence is checked and that any training that is needed to improve their competence is identified and undertaken. As RWMD's work changes, so will the requirements for being a capable organisation, and so our baseline will evolve with time.

Given the MRWS approach to site selection, involving voluntarism and partnership, staff who will be a position of engaging with stakeholders need more than just technical expertise; they also require a sensitivity to the public environment in which we must conduct our affairs, an ability to communicate and work with a range of stakeholders, empathy, and an ability to put into practice a listening culture.

Intelligent customer

An important aspect of being a capable organisation is the ability to act as an intelligent customer for the work that other organisations carry out on our behalf. Being an intelligent customer has a number of aspects, including the ability to specify work that others will carry out, the ability to assess bids from potential contractors, the ability to understand the work that is being carried out, and the ability to know whether a completed piece of work has actually met the safety and environmental needs that gave rise to it in the first place. The development of the ESC includes work undertaken by a wide contractor base, experienced in development of GDF safety cases for a wide range of other waste management programmes.

Our Organisational Baseline Document [66] defines the intelligent customer aspects of each role that carries safety or environmental responsibility. The competence of our staff to act as intelligent customers is assessed and managed in the same way as our other competences.

Our staff are involved in numerous international activities, including organisation of international conferences, participation in international safety case programmes, and involvement in EC projects. These activities support the development of their skills and experience and provide further evidence of their intelligent customer status.

Management of organisational change

RWMD has a Policy for Management of Organisational Change for the GDF Delivery Organisation [120]. This policy covers any proposed change to RWMD's organisational structure, resource or competence requirements. Any proposed change to the organisation is first assessed and categorised on the basis of its potential to affect safety and/or environmental compliance. Where the proposed change is significant, it is discussed by our Nuclear Safety and Environment Committee before the change is made. After a change has been introduced, there is a period of monitoring to ensure that the change had the desired effect and did not produce any unexpected adverse consequence.

One aspect of a change is often that new competence requirements may emerge for the individuals involved in the change. These requirements are identified as part of the assessment for the organisational change.

Learning culture

We foster a learning culture through the following mechanisms:

- Operational experience: We work with several overseas waste management organisations to identify areas of common interest where mutual benefit can be gained through cross fertilisation of ideas and sharing of experience. Often such co-operation is undertaken through bilateral agreements. Entering into an agreement allows for benchmarking, staff training opportunities and access to best international practice on societal and technical matters to do with waste treatment, decommissioning and waste disposal. We work with the following national waste management organisations, among others:
    - Andra – the French "National Agency for the Disposal of Radioactive Waste";
    - JAEA – the Japanese Atomic Energy Agency;
    - Nagra – the Swiss "National Co-operative for the Disposal of Radioactive Waste";
    - NUMO – the Nuclear Waste Management Organisation of Japan;
    - Ondraf/Niras – the Belgian "Agency for Radioactive Waste and Enriched Fissile Materials";
    - RWMC – the Radioactive Waste Management Funding and Research Centre of Japan;
    - SKB – the "Swedish Nuclear Fuel and Waste Management Company"; and
    - US DOE – the United States Department of Energy.

  Examples of ongoing collaborative activities with these organisations include:
    - Our agreement with SKB facilitates the exchange of information and cross review of draft safety reports (e.g. SKB commented on drafts of the generic ESC reports and we comment on drafts of SKB's safety reports).
    - Our work with Nagra provides for access to experimental facilities at the Grimsel underground research laboratory in Switzerland.

  We are a member of the International Association for Environmentally Safe Disposal of Radioactive Materials (EDRAM), which acts an international forum of national waste management organisations. We are also a member of the "Club of

Agencies", which brings together the waste management organisations within the European Union, plus Switzerland and the EC.

- Enhancing learning: We are committed to providing opportunities for our employees to undertake learning activities as part of their career development. We recognise that individual learning and development needs may include external support, for example further education, as well as internal support to provide broader experience and other experiential learning opportunities.

Knowledge management

The NDA has a Knowledge Management Policy that requires knowledge to be shared or preserved where this will support and enable the NDA and Site Licence Companies to meet their organisational objectives. We are in the process of developing a strategy for knowledge management specific to the geological disposal project, based on an analysis of our current and expected future needs. We recognise that the proper recording of decisions, the reasons for them and the reasons why alternative choices were not made is particularly important in the early stages of a programme. One reason for this, as noted in Section 3.1.2, is that early decisions may be built upon, so that they progressively acquire growing significance. It is important to be able to trace the interactions between linked decisions, such that if the information underpinning an early decision changes, there is a traceable record of all related decisions that may need to be re-evaluated.

As noted in Section 3.1.3, we have already begun the development of a requirements management system that we plan to use to capture knowledge relating to requirements, constraints and acceptance criteria for the design of a GDF. It is expected that the system will allow reports and other forms of data to be used to provide traceable evidence that the design meets the Disposal System Specification [23]. The work on the requirements management system forms part of an embryonic process for capturing and transmitting knowledge between teams, ensuring an accessible and durable evidence trail for decision-making and stakeholder engagement.

Other information management systems are discussed in Section 3.3.4.

3.3.1.2 Governance arrangements

Our external and internal governance arrangements are summarised in our Safety and Environmental Management Prospectus (SEMP) [66]. Note that these arrangements are under review, and the SEMP is a living document that will evolve as we move towards Site Licence Company (SLC) status. The current governance arrangements are briefly summarised below.

External governance

The UK Government has established several oversight groups to manage arrangements for the planning and delivery of a GDF, and to ensure that a holistic approach is taken to radioactive waste management across the NDA estate. These groups provide the formal routes through which the UK Government oversees our work and holds us to account.

With regard to the disposability assessment process, the joint guidance issued by the Health and Safety Executive (HSE), the Environment Agency, and the Scottish Environment Protection Agency (SEPA) [121] sets out requirements for the oversight of waste packaging advice for future disposal. We advise waste producers on the conditioning and packaging of radioactive wastes via the disposability assessment process (see Section 3.1.4.1) and waste producers in turn make their case for packaging to the HSE's Nuclear Installations Inspectorate (NII). The environmental aspects of the packaging case are submitted to the environmental regulators, who scrutinise proposed developments of the disposability assessment process under agreements with us, to ensure that the advice we provide to waste producers is fit for purpose. This regulatory scrutiny of the disposability assessment process is carried out by various means, including regulatory review of key disposability assessment outputs (e.g., Assessment Reports) and regulatory review of particular areas of our programme, e.g. criticality, inventory, gas production and migration (see Section 2.3).

Internal governance

We are currently an operating division of the NDA. We are accountable to Corporate NDA via the Repository Development Management Board (RDMB). The RDMB is chaired by an Executive Director of the NDA and comprises members from RWMD, the wider NDA and independent members.

Internal governance is exercised throughout RWMD from the RDMB downwards. The RDMB provides oversight of RWMD and ensures separation of our operational role from the wider strategic role of the NDA. The RDMB is the formal route through which corporate NDA holds us to account, and will become the Board of Directors when we are legally established as a wholly-owned subsidiary of the NDA. The minutes of RDMB meetings will be available to regulators and the public.

The top-level of our structure below the RDMB is represented by the RWMD Executive, comprising the RWMD Managing Director and the Executive Directors. The RWMD Executive is responsible for delivering the business plan, agreed by the RDMB. Our management organisation below Director level is described in detail in the Organisational Baseline Document [66]. The organisational structure provides for clear management lines of communication and decision-making.

We operate in accordance with an annual business plan and budget, agreed with the NDA and approved by the RDMB. Our directors submit a monthly report to the RDMB, which gives details of progress against the business plan.

Project management process

We have established a single Programme Board to oversee our entire work programme and to act as a Project Board for individual projects. The Programme Board reviews each project to cover scope, schedule and cost performance, and discusses issues and risks that require the Programme Board’s involvement. Key projects, e.g. the DSSC, are reviewed every month. This arrangement provides the mechanism for the Programme Board to provide appropriate direction to each of the projects. The Programme Board has a small core membership - the Executive Directors - to provide consistency and overview of the programme. The Board is augmented by our senior staff or by suppliers for specific projects.

Most projects are managed following the PRINCE2 or similar project management process [122]. Details of the scope and execution of individual projects are set out in project initiation documents and project implementation plans.

Other groups and committees

We establish as necessary internal groups or committees, or are involved in external groups or committees to provide more detailed direction and guidance to particular parts of our work, and/or to facilitate our work, as set out in the Safety and Environmental Management Prospectus [66]. The kinds of areas covered include:

- safety, the environment and the disposability assessment process, including safety and environmental aspects of site characterisation;
- processes for Disposal System Specification (DSS), design and safety case development, including the ESC;
- liaison with the regulators, including the environmental regulators; and
- procurement practices, including procurement for all work supporting the ESC and the ESC itself.

We also establish, where appropriate, expert groups to scrutinise aspects of our programme, such as our R&D strategy and programme. As noted in the Acknowledgements, the development of this report has been scrutinised by an Advisory Panel that we established – the DSSC Advisory Panel.

The RWMD Executive receives reports from all such groups and committees.

3.3.1.3 Our environmental safety management system - key documentation

Our environmental safety management system is documented in a suite of linked reports. Key reports include:

- our Safety and Environmental Management Prospectus [66], which provides further information on our safety management system and how we promote a positive environmental safety culture;
- our Organisational Baseline Document [66], which supports the Safety and Environmental Management Prospectus by showing how we define what constitutes a suitable and sufficient organisation, having the resources and competence to ensure that our safety and environment-related activities can be carried out in a controlled, safe and environmentally acceptable manner; and
- our Management Systems Manual [123], which describes our overall management system.

Our management system is part of the NDA-wide system and consists of both procedures that are specific to ourselves and procedures that apply across the NDA. Our Management Systems Manual includes reference to specific procedures for safety and environmental protection, but also covers the more general business management systems such as document control, document review, programme and project management, many of which also support the safety and environmental management systems.

Our Management Systems Manual sets out the process for implementing, assessing and continually improving the management system such that safety and environmental protection are properly addressed. The management system and compliance arrangements apply to the whole organisation up to and including the Repository Development Management Board (RDMB) and RWMD Executive team.

Changes to our environmental safety management system documentation are controlled by specific procedures set out in [124].

Our quality management system is described in Section 3.3.5.

3.3.2 The application of sound science and good engineering practice

We have a variety of methods for ensuring that all our work in support of safety case development applies sound science, and the management structure and processes described in this section contribute to the demonstration of the application of sound science and good engineering practice. In particular, our assessment, R&D and engineering teams maintain an understanding of scientific and engineering developments in radioactive waste management and disposal, both within and outside the UK, through:

- work we and our contractors perform for other programmes;
- co-operation and liaison agreements we have with waste management programmes in other countries and internationally (IAEA, NEA, EC); and
- interfacing with other waste management programmes of the NDA and its Site Licence Companies.

Knowledge of such developments feeds into optimisation and optioneering analyses, including review of past decisions, and planning for future updates of the ESC.

In accordance with the GRA and good practice for the design of radioactive waste disposal facilities, our design and safety strategy makes use of multiple barriers and multiple components with multiple safety functions. Our design work is also based on robust and well established technology, where possible demonstrated in facilities elsewhere. Our design strategy does not rely on human action to ensure safety during the post-closure period, and minimises reliance on human action during the operational period. These are key aspects of our overall safety concept, as outlined in Section 3.1.1.

In accordance with the application of sound science, the ESC and supporting assessments are based on a formal development process (Section 3.2) that conforms to internationally accepted methodology as set out by the IAEA and the NEA [7, 8]. The formalised assessment methodology ensures that all necessary aspects are covered, and requires a structured treatment of uncertainty. For example, the use of a FEP list and formal FEP analysis to support scenario development will ensure that all FEPs have been considered. It will also act as a mechanism for communication within the project. Iterative FEP analysis, combined with a conservative approach to the modelling of events such as inadvertent human intrusion, will ensure that a comprehensive set of potential futures is covered in our assessments. Our environmental safety assessment modelling is undertaken mostly in house by a small team, which ensures that good interfaces exist within the organisation and with any ESC contractors, and that we retain control and understanding of this core component of our ESC.

We collaborate with internationally experienced contractors in developing the ESC, and in associated R&D and engineering studies. These contractors have worked on many national and overseas radioactive waste management projects. Our work is based on consideration of GDF designs and environmental safety assessments for many other projects worldwide and, where appropriate, parameter values selected for the illustrative calculations represent a consolidation of expert judgement across several projects. Further, the designs on which these calculations are based use well established and well understood components and technologies, for which models and assessment data are established in a number of programmes.

The disposal concept option studies [28, 29, 30, 31] and generic design studies [24] that we have conducted recently were discussed in Section 3.1.3. Our engineering design processes will ensure that the requirements of the Disposal System Functional Specification [22] and the Disposal System Technical Specification [23] are addressed in future disposal concept option studies and GDF design work. Future design work would be informed by the results of investigations at candidate sites, changes to the UK Radioactive Waste Inventory, and the results of stakeholder and local community engagement at candidate sites. For example, the identification of candidate sites would allow disposal concepts to be considered for specific sites. Site-specific design work, based on these concepts and incorporating the results of site investigations and other information, would be undertaken progressively and in increasing detail as the MRWS Site Selection Process proceeds towards GDF implementation. Changes to the Disposal System Specification are managed through a formal change control process [69].

The systematic treatment of uncertainty in our assessments is consistent with the requirements of the GRA. Uncertainty analyses examine the potential consequences of recognised spatial and temporal variability, and uncertainty in the assessment calculations. One aim of such analyses is to demonstrate that, notwithstanding the expected overall performance of the geological disposal system as captured in a base scenario, any unexpected under-performance of the engineered or natural barriers over the long term, as captured in variant scenarios, still provides a level of safety consistent with the regulatory guidance and that none of the uncertainties threaten the safety case (see Sections 3.2.2.3 and 3.2.2.4). We are committed to building confidence in our ESC and environmental safety assessment calculations through R&D, design and site characterisation work. R&D, design and site characterisation studies will be conducted to reduce uncertainties in the assessment results, where justified by an analysis of cost versus the importance of the uncertainty and the likelihood of success in reducing it.

In our work we apply standard tools for analysis that are in use by many other waste management organisations. At the generic stage, these include PC-CREAM [125] for OESA calculations and GoldSim [126, 127] for PCSA calculations, as discussed in Section 4.3.

Peer review and scrutiny of the deliverables from our assessment, R&D and engineering studies, as discussed in Section 3.3.5, contribute to ensuring the application of sound science and the use of good engineering practice. Other means of controlling the quality of work include internal and external review of task specifications (‘peer preview’), review of proposals produced by our contractors – including review of proposed assessment, experimental and/or design protocols – and ensuring that both we and our contractors have in place quality management systems appropriate to the proposed work (see Section 3.3.5).

As noted in Section 3.2.2.4, drafts of the generic ESC reports were subject to formal external peer review.

3.3.3 Provision of information

Openness and transparency are founding principles of the NDA, and are part of our culture. Our policy is to share information openly, and be accessible to all.

Information will be published or otherwise made available unless there is good reason for it to be withheld. Most of the documents we prepare, or that are prepared for us by contractors, are freely available via the NDA’s website (http://www.nda.gov.uk/documents/biblio/). Any member of the public can request copies of our documents. The entire suite of generic DSSC reports is publicly available, and updates will also be publicly available.

Our Transparency Policy is operated in accordance with legislative requirements including the Freedom of Information Act 2000, Environmental Information Regulations 2004, and the Data Protection Act 1998. Requests for information under the Freedom of Information Act and the Environmental Information Regulations are answered by us within 20 working days of receipt. The way in which we respond to requests under the Freedom of Information Act is described in [128, 129].

As discussed in Section 2.3, we have entered into a voluntary arrangement with the Environment Agency to subject our activities to regulatory scrutiny and audit. The principal voluntarily regulated activities are:

- development of the DSSC, including ESC documentation;[14]
- development and maintenance of the DSS;
- site evaluation studies in support of the MRWS Site Selection Process;
- stakeholder and community engagement;
- Sustainability Appraisal and Strategic Environmental Assessment;
- GDF design;
- the disposability assessment process; and
- R&D activities.

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[14] As discussed in Section 2, we intend to develop the ESC progressively, and each new update will consider information developed in the other work programmes noted here.

Information can be provided to the regulators in the form of documents, computer codes, presentations, and discussions at meetings. Some of this information will be at ‘draft’ stage and will not be placed on the NDA’s website (e.g., the storyboard and full draft of this report reviewed by the Environment Agency).

It is our responsibility to decide on the delivery programme, and this will consider the requirements of the MRWS Site Selection Process and the GRA. However, once we submit an application for our first Environmental Permit in Stage 5 of the MRWS Site Selection Process (see Section 2.2), the Environment Agency will start to exert formal regulatory control over some of our activities.

As Community Siting Partnerships are established, we will share information on a regular basis with them, initially to support their deliberations on participation in the MRWS Site Selection Process.

3.3.4 Documentation and record-keeping

We operate an electronic document and records management system that is used to control, retain and provide ready access to work/project documentation. A separate system is used to manage and distribute management system documentation. These systems are used to achieve the following:

- provide a searchable database of project documents;
- control the access to documentation, consistent with our transparency and security policies; and
- track changes to documents and revision history.

The document management systems include all internally generated documents and the majority of documents received from external sources. Documents previously generated and owned by Nirex have been transferred into these systems.

Another important system for recording and managing information is the one used by the NDA to update and record the UK Radioactive Waste Inventory of radioactive wastes on a periodic basis [130].

Issues identified by the environmental regulators are recorded and managed using a formal issue resolution process and issue resolution forms (IRFs).

We recognise that other documentation and record-keeping systems will need to be put in place in the future for specific purposes, for example to record and manage site characterisation results, detailed design information, and/or assessment information. A system for managing project decisions and requirements is discussed in Section 3.3.1.1. There is time to develop these systems.

3.3.5 Quality management and peer review

We are certified to ISO 9001:2008 and ISO 14001:2004 by Lloyds Register Quality Assurance (LRQA). Our Management Systems Manual [123, Appendices 1 and 2] contains a set of matrices that indicate where the requirements of ISO 9001:2008 and ISO 14001:2004 are addressed within our management system.

We have set out specific quality assurance requirements in a range of areas relevant to development of the generic ESC [110] – for example:

- the control of documents;
- review and scrutiny of deliverables; and
- checking of computer calculations.

As the project develops, additional quality assurance requirements will be developed to cover other key aspects of our programme such as site characterisation, and GDF design, construction and operation.

Peer review and scrutiny of deliverables is an important principle of our quality management system, and we apply the following review activities to deliverables:

- internal technical review;
- external peer review of major science and engineering reports – this includes the DSSC; and
- reviews by the appropriate Department Head prior to publication.

In addition, regulatory review of deliverables under either voluntary or formal arrangements as part of a staged process provides a valuable means of ensuring the ESC and key supporting documents are consistent with regulatory expectations.

Scrutiny of our activities is also provided by the Committee on Radioactive Waste Management (CoRWM), which provides independent scrutiny and advice to the UK Government on the long-term management of higher activity radioactive wastes and materials that may be declared to be wastes. This includes scrutiny of our proposals, plans and programmes to deliver geological disposal. In order to carry out its work, CoRWM has formed several working groups. One of these is scrutinising plans for, and performance of, UK R&D related to geological disposal and interim storage. We engage regularly with CoRWM to discuss our activities. We also provide key documents to CoRWM for review.

Finally, we present selected technical work as papers at scientific conferences and in scientific journals, where it is subject to peer review by the wider scientific and radioactive waste management community. Publication of our work in peer-reviewed fora builds confidence in its wider technical and scientific credibility.

4 Assessment basis

This section describes the "assessment basis", by which we mean information that underpins the qualitative and quantitative safety assessments provided in Section 5 of this generic ESC:

- Section 4.1 summarises the concept for a GDF, including the waste inventory and uncertainties, waste packaging arrangements, the generic geological environments, and the types of EBS and GDF layouts that are being considered in this generic ESC.
- Section 4.2 summarises the scientific and technical information and understanding that underpins this generic ESC, including a summary of expected evolution of a generic GDF.
- Section 4.3 summarises the models and modelling approaches, including computer codes and databases that we have used to develop and quantify our understanding of generic geological disposal systems.

More detailed information about this underpinning scientific and technical information and understanding is provided in the DSSC Tier 2 safety assessment reports and supporting reports (see Figure 1.2).

The assessment basis represents a snapshot of where we are now, and will change as the MRWS Site Selection Process moves forward. In particular, as we start to consider particular sites, the ESC will become increasingly more detailed and the assessment basis will evolve to ensure that it remains 'fit for purpose', i.e., suitable for the decision required at that stage of the process. We will present information specific to the sites and geological environments under consideration, including describing disposal concepts that are appropriate for each geological environment. We will use the assessment basis to explain why we have selected particular disposal concepts for each of the sites under consideration. Whilst there is precedent and UK and overseas understanding to draw on, some issues can only be resolved through detailed, site-specific work, and so the current assessment basis is of necessity more qualitative than future versions will be.

There is a wide variety of geological environments in the UK that might be suitable to host a GDF, but at this stage we do not know in which environment we will be developing a GDF. The information in this section about the geological environment and the disposal concepts that might be implemented is therefore generic. Further information on the UK application of several illustrative geological disposal concept examples is provided in Appendices A, B and C and in the Tier 2 supporting reports. The information describing the geological disposal system characteristics in these appendices illustrates the issues we might need to address once we have a site, and is indicative of the type of information we will include in this section of the ESC once we have specific sites and disposal concepts to consider.

Different geological disposal systems will require different approaches to the development of the ESC. The generic OESA and PCSA [26, 27] present the results of example calculations of the type we consider to be appropriate to support the ESC at the current generic stage of the MRWS Site Selection Process. The quantitative environmental safety assessments we undertake in Stage 4 of the MRWS Site Selection Process will build on the approaches described in the OESA and the PCSA, as summarised in Section 4.3.

The calculations and illustrative geological disposal concept examples assist us in developing our understanding of the manner in which different types of geological disposal system provide safety through multiple barriers working together, and provide confidence that we can design an appropriate EBS that will provide the required level of performance to work with the geological environment at the candidate site. Doing this allows us to illustrate how we might construct a safety case for a range of different types of real site.

4.1 System concept

4.1.1 The wastes

A variety of different types of waste will be disposed of in a GDF. Important characteristics of each type of waste include the physical form of the waste material (the wasteform), physical and chemical properties of the wasteform, and volume.

The wasteform is an important component of a GDF, contributing to containment of the inventory during both operational and post-closure periods. The different wasteforms that may be disposed of are described in more detail in the Package evolution status report [15].

The MRWS White Paper [4] provides an estimate - the "Baseline Inventory" - 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. The Baseline Inventory is based on the 2007 UK Radioactive Waste Inventory [130, 131]. It includes materials not currently classified as waste - SF and stocks of separated plutonium and uranium. However, as noted in Section 3, it excludes LLW that can be managed under the UK Government's "Policy for the Long Term Management of Solid Low Level Radioactive Waste in the United Kingdom" [65]. The Baseline Inventory is based on UK-wide inventory data, so includes wastes that are expected to be managed under the Scottish Government's policy of interim near-surface, near-site storage or disposal and not through geological disposal.

The Baseline Inventory comprises:

- **Low-level waste (LLW)**. LLW comprises building rubble, soil and steel items such as framework, pipework and reinforcement from the dismantling and demolition of nuclear reactors and other nuclear facilities and the clean-up of nuclear sites. However, currently most LLW is from the operation of nuclear facilities, and this is mainly paper, plastics and scrap metal items. Only a small fraction of the LLW produced in the UK will be consigned to a GDF.
- **Intermediate-level waste (ILW)**. ILW exists in a wide range of physical and chemical forms including metals, graphite, concrete and other rubble, sludges, flocs and various organic materials including oils. Its major components are metal items such as nuclear fuel casing and nuclear reactor components, graphite from reactor cores, and sludges from the treatment of radioactive liquid effluents.
- **High-level waste (HLW)**. HLW is initially produced as a concentrated nitric acid solution containing fission products from the primary stage of reprocessing SF. HLW is currently incorporated into borosilicate glass (vitrified) in stainless steel canisters at the Sellafield Waste Vitrification Plant (WVP). It is expected that, by 2015 most of the HLW that is expected to arise will have been treated in this way. HLW generates significant amounts of heat until such time as the short-lived fission products have decayed. Vitrified HLW will be stored for at least 50 years to allow a significant proportion of the radionuclides to undergo significant radioactive decay such that they are more suitable for disposal. Further packaging of the canisters is likely to be required to create a waste package suitable for transport and disposal.

In addition to these wastes, the UK Government has specified other radioactive materials that could possibly come to be regarded as waste in the future and so are included in the Baseline Inventory for disposal. These materials are:

- **Spent nuclear fuel (SF)**. Most of the SF from the UK's existing civil reactors has been, or will be, reprocessed to separate plutonium and uranium which could be used to make new fuel and so will not be consigned to a GDF. However, some SF from UK advanced gas-cooled reactor (AGR) power stations and all SF from the Sizewell B pressurised water reactor (PWR) may not be reprocessed and so would be managed as waste. A new build programme would generate further arisings of SF.
- **Plutonium (Pu)**. The UK stocks of separated plutonium are not currently declared as waste but are held because they may have a future use, for example in the manufacture of some reactor fuels. If they were however to be disposed of in a GDF, they would first need to be converted into a suitable stable wasteform. For the purposes of the generic DSSC, we have assumed that the separated plutonium is converted into a wasteform that allows it to be disposed of using the same disposal concept as for HLW and SF.
- **Uranium (U)**. Stocks of uranium come from refining uranium ore to make fuel or from reprocessing SF. These stocks include some enriched uranium which is suitable for making fuel for reactors, but the majority of the inventory by volume comprises depleted, natural and low-enriched uranium (DNLEU) residues ('tails') from fuel production. For the purposes of the generic DSSC, we have assumed that the DNLEU is converted into a stable wasteform that allows it to be disposed of using the same disposal concept as the LLW and ILW, and that the small quantity of high-enriched uranium (HEU) is converted into a wasteform that allows it to be disposed of using the HLW/SF disposal concept.

While the UK Radioactive Waste Inventory reflects the best available public domain information concerning the quantities of radioactive materials in the UK, it does not contain sufficient detail to allow it to be used in the development of plans for the long-term management of the waste. For this purpose, we have developed a series of derived inventories [10, 11, 12, 13] for each of the materials included in the Baseline Inventory. While being based on the same data from the UK Radioactive Waste Inventory that was used to produce the Baseline Inventory, the "Derived Inventory" represents a greater level of detail regarding the individual waste streams, and considers this information at the 'waste package level'. Development of the Derived Inventory involved consideration of specific waste packaging concepts which may be adopted for different waste types, some of which have been assessed by way of the disposability assessment process. This allows information on expected waste conditioning factors to be used to determine the actual volumes of packaged waste that would need to be accommodated in a GDF. The term 'Derived Inventory' is used to explain this, but in practice this should be seen as a more detailed description of the Baseline Inventory, rather than a separate distinct inventory.

The Derived Inventory considers 112 radionuclides, 31 of which are considered priority radionuclides of potential significance for the TSC, OSC, OESA and PCSA, and the assessment of criticality safety. We require more detailed information on these 31 radionuclides than on the other radionuclides in the UK Radioactive Waste Inventory. During the operational period, short-lived radionuclides such as caesium-137, strontium-90, hydrogen-3 (tritium) and cobalt-60 are of greatest significance. Radium-226 is also significant because it decays to radon-222, which potentially poses a hazard during GDF operations [6]. After closure of a GDF, long-lived radionuclides such as chlorine-36, iodine-129, carbon-14, technetium-99 and the actinides are significant.

One of the ways in which we evaluate the robustness of our ESC is through consideration of alternative inventory scenarios. We have used the Derived Inventory to define a 'reference case' inventory (equivalent to the Baseline Inventory) and an upper inventory that we consider in assessment and design studies. The assumptions underpinning the Derived Inventory reference case are set out in the Derived Inventory reports [10, 11, 12, 13] and the Radioactive wastes and assessment of the disposability of waste packages report [14]. Full details (e.g. volumes, chemical constituents, radionuclides) of specific waste streams as stored or as conditioned or predicted to arise from existing nuclear facilities are given in these reports and in the DSTS [23], which also provides details of the expected number of each type of waste package. A summary of the information used in this generic ESC, namely packaged volumes and radioactivity for each waste type, is given in Table 4.1.

Table 4.1 Basic data for the Derived Inventory reference case considered in the generic DSSC

Data summarised from [10, 11, 12, 13]. Note that the decay of the inventory at later times is shown in Figure 4.1.

Materials		Packaged volume		Radioactivity (At 1 April 2040)	
	Notes	Cubic Metres	%	Terabequerels	%
HLW	1, 2, 3, 5	1,400	0.3%	36,000,000	41.3%
ILW	1, 2, 5	364,000	76.3%	2,200,000	2.5%
LLW (not for LLWR)	1, 2, 5	17,000	3.6%	<100	0.0%
Spent nuclear fuel	1, 4, 5	11,200	2.3%	45,000,000	51.6%
Plutonium	1, 4, 5	3,300	0.7%	4,000,000	4.6%
Uranium	1, 4, 5	80,000	16.8%	3,000	0.0%
Total		**476,900**	**100**	**87,200,000**	**100**

Notes
1. Quantities of radioactive materials and wastes are consistent with the 2007 UK Radioactive Waste Inventory.
2. Packaging assumptions for HLW, ILW and LLW not suitable for disposal at the existing national LLWR are taken from the 2007 UKRWI. 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 CoRWM Baseline Inventory. 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.

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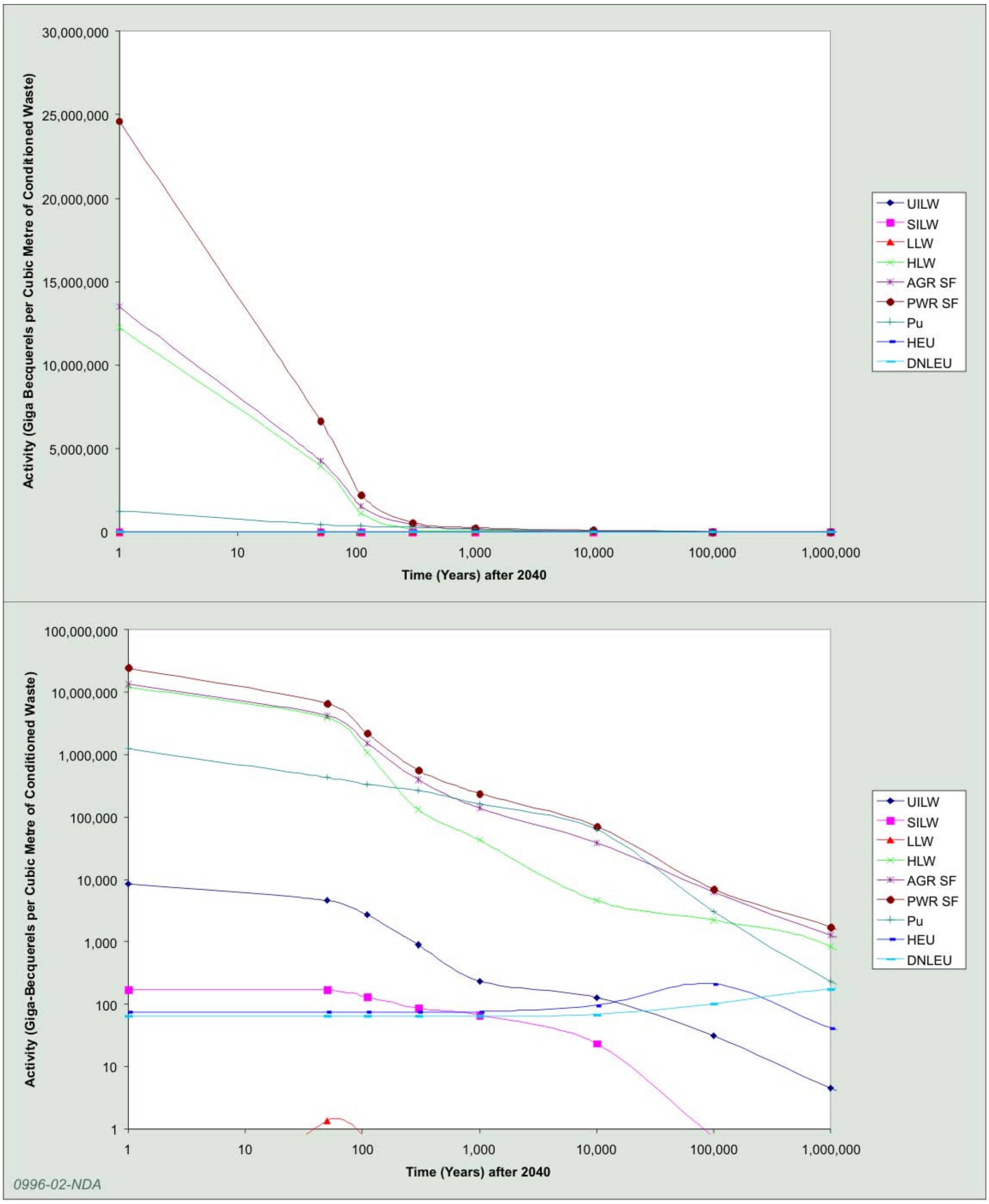
In terms of the hazard presented by the waste, we note that after the first few hundreds of years, a large part of the radioactivity in the wastes will have decayed, and the hazard will be much reduced, as shown in Figure 4.1. This figure contains two graphs: the only difference between the two graphs is that the activity axis in the top diagram is linear and in the bottom diagram is logarithmic. The time axis in both diagrams is logarithmic. The very significant decay that occurs over the first few hundreds of years can be seen clearly in the top diagram; however, little differentiation can be seen beyond this time. The use of axes that are logarithmic allows more detail to be seen at later times (and at lower levels of calculated impact), as seen in the bottom diagram. It is for this reason that the results of our quantitative assessment for the post-closure period are typically shown using logarithmic axes.

Figure 4.1 illustrates a further feature of the inventory. It can be seen from the bottom diagram that the activities of LLW, DNLEU and HEU increase at certain times. This is due to ingrowth of daughter products with long half-lives. Activities reach a peak when these daughters are in equilibrium with the parent radionuclide. Long-lived daughters are accounted for explicitly in the decay curves by calculating the amount of ingrowth that has occurred during each period. Short-lived daughters may be accounted for by assuming that equilibrium is reached soon after disposal. Ingrowth significantly affects the calculated radiological impacts; the effect is most readily seen in our calculations of groundwater transport for DNLEU (see Section 5.2.2.1).

Figure 4.1 The decrease in activity in giga-Becquerels (10^9 Becquerels) per cubic metre of conditioned waste with time for the different types of wastes in the Derived Inventory reference case

Data from [10, 11, 12, 13], which assume all wastes will have been disposed of and a GDF closed by 2090. Note that the actual operational period is likely to be greater, and our planning documents generally assume a period of 100 years [23]. The top diagram has a linear vertical activity axis and the bottom diagram a logarithmic activity axis. The horizontal time axis is logarithmic in both diagrams. It can be seen from the top diagram that a significant fraction of the total activity from all wastes decays within the first few hundreds of years. The bottom diagram illustrates more clearly the decay over longer timescales and shows how the ingrowth of long-lived daughter radionuclides for LLW, DNLEU and HEU can actually increase their activity at certain times (see text).



In addition to the Derived Inventory reference case, this generic ESC considers an 'upper inventory' that includes allowances for uncertainty on volume and composition in relation to the estimates of wastes and materials that have not yet arisen, and in relation to future operation of existing facilities, e.g. how long reactors will operate. For example, the upper inventory assumes that all SF from existing advance gas-cooled reactors (AGRs) and from the Sizewell B pressurised water reactor (PWR) is reprocessed. The upper inventory also includes an estimate of the ILW, depleted uranium and SF that may arise from a proposed new generation of nuclear power stations. We assume that none of the SF from the proposed new generation of nuclear power stations would be reprocessed. We have compiled this upper inventory to allow the implications of these uncertainties to be explored. The upper inventory is not intended to be a maximum inventory and does not set out the largest inventory which could be disposed of in a GDF. A summary of the upper inventory is provided in the Disposal System Technical Specification [23]. In this generic ESC, the implications of the upper inventory are discussed qualitatively – we have not undertaken additional quantitative assessment calculations specifically for the upper inventory. Full details of alternative inventory scenarios are given in the Derived Inventory reports [10, 11, 12, 13].

Some of the waste packages and materials used in construction of a GDF will contain chemotoxic constituents. The **chemotoxic inventory** includes certain constituents of the wastes themselves, such as uranium, that are both radioactive and chemotoxic, and non-radioactive materials such as heavy metals. Other materials present in the inventory may have a detrimental impact on the performance of the geological disposal system, but not be of environmental concern. For example, certain organic materials can form complexes with particular radionuclides, resulting in an increase in the solubility or a decrease in the sorption of these radionuclides (see [20]). Details of the non-radioactive constituents of the waste inventory are also given in the UK Radioactive Waste Inventory [130, 131] and the Derived Inventory reports [10, 11, 12, 13]. Substances of interest include beryllium, heavy metals, phenols and non-aqueous phase liquids. Non-aqueous phase liquids may either be part of the ILW inventory or may be generated as the wastes evolve [20].

4.1.2 Waste packaging and waste acceptance criteria

GRA Requirement R13: Waste acceptance criteria

The developer/operator of a disposal facility for solid radioactive waste should establish waste acceptance criteria consistent with the assumptions made in the environmental safety case and with the requirements for transport and handling, and demonstrate that these can be applied during operations at the facility.

Our approach to meeting GRA Requirement R13 is discussed in Section 3.1.4. Once we have a site and have developed a site-specific design, we will start to develop waste acceptance criteria (WAC) for a GDF. These will reflect our GDF design (for example the EBS materials and the proposed waste emplacement strategy) and the expected conditions in the GDF under consideration. The WAC will be finalised prior to accepting any waste for disposal. This section will be developed further as the WAC are developed.

For the purpose of this generic ESC, the assessment basis concerning waste packages that needs to be considered is drawn from our disposability assessment process, as discussed in Section 3.1.4, and an assumption that materials that have not yet been considered in our disposability assessment process would be packaged in such a way that they would be compatible with the UK application of the illustrative geological disposal concept examples for ILW/LLW and for HLW/SF. We set out below the current status of the disposability assessment process (Section 4.1.2.1), and then, in this context, discuss the waste packaging assumptions made for this generic ESC (Section 4.1.2.2)

4.1.2.1 Current status of disposability assessment process

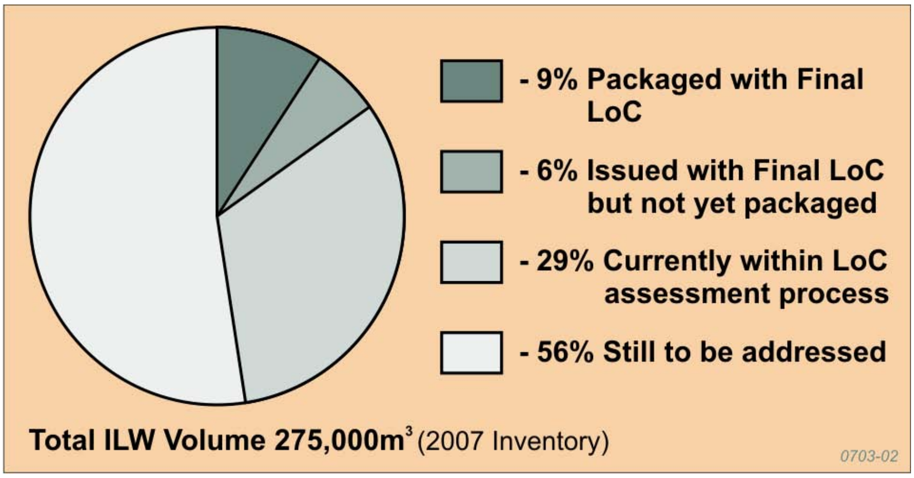
As of the end of March 2009, we had undertaken assessment of 53% by volume of all higher activity radioactive waste in the 2007 UK Radioactive Waste Inventory, and had provided a final stage LoC for 15%, as broken down in more detail below for legacy ILW and legacy HLW [132]. This section also discusses the status of disposability assessment work for other possible materials we will have to manage in a GDF, including the higher activity radioactive wastes from a possible programme of new nuclear power stations ('new build wastes').

ILW

Through March 2009, the cumulative total of final stage LoCs was 87, encompassing some 39,800 cubic metres of conditioned ILW, representing almost 15% of all ILW within the 2007 UK Radioactive Waste Inventory (Figure 4.2). Furthermore, we had issued early stage LoCs and/or disposability assessment reports covering a further 79,200 cubic metres of ILW, for which packaging strategies are being developed. Therefore, as of March 2009, we had considered approximately 44% of UK legacy ILW within our disposability assessment process.

Figure 4.2 Status of disposability assessment process for UK ILW as of March 2009

2007 UK Radioactive Waste Inventory, conditioned waste volume – packaged volume is greater.

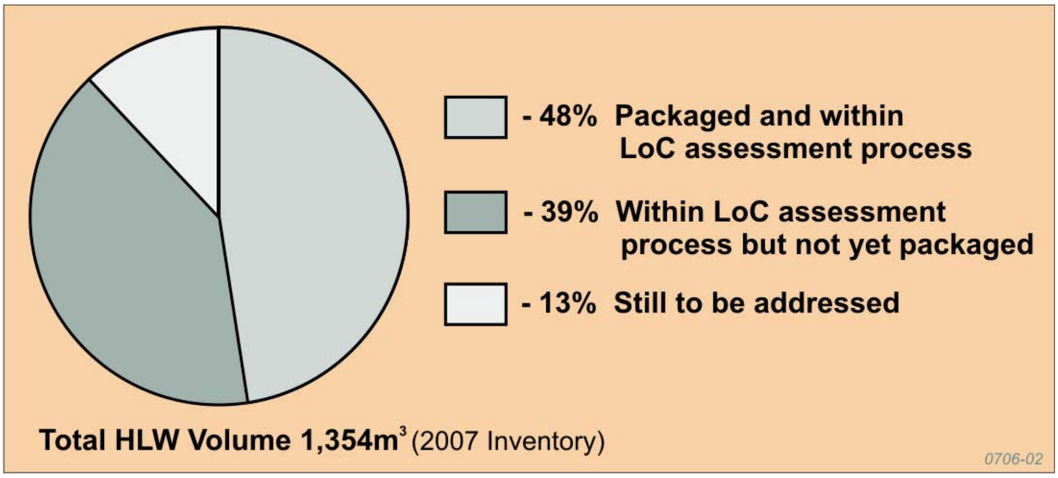


HLW

As of March 2009 we had also undertaken a preliminary disposability assessment for vitrified HLW and issued a detailed assessment report. Approximately 650 cubic metres of conditioned and packaged HLW had been placed into interim storage, representing 48% of the total reported HLW in the 2007 UK Radioactive Waste Inventory (Figure 4.3). A further 39% of HLW had been assessed and is within the disposability assessment process, but had not yet been vitrified and packaged, so that a total 87% of HLW has been assessed.

Figure 4.3 Status of disposability assessment process for UK HLW as of March 2009

2007 UK Radioactive Waste Inventory, conditioned waste volume – packaged volume is greater.



Other possible wastes (new build wastes / Pu / U)

By the end of 2009 we had also issued disposability assessments covering all ILW and SF types to be produced from proposed new build nuclear power stations, as a required input to the Generic Design Assessment for new nuclear power stations [133, 134].

Disposability issues for separated plutonium and HEU are being considered in work carried out by the NDA as part of its wider research programme on the management of these materials [135]. The NDA's research has led to concept stage LoC submissions for plutonium immobilisation and packaging options based on the use of cementitious, polymer, ceramic, glass and mixed-oxide ceramic wasteforms, and we are assessing these submissions.

The options for disposal of depleted and natural uranium, including innovative technologies for using depleted uranium in waste processing and packaging, thereby providing criticality controls, are considered in [136].

4.1.2.2 Assessment basis for generic ESC

Given the current status of the disposability assessment process, detailed waste packaging specifications are already available for most of the ILW and LLW streams [78]. ILW and LLW are usually cemented into waste containers having standardised features:

- unshielded packages which usually require remote handling – e.g. 500 litre drum, 3 cubic metre box (two variants), and 3 cubic metre drum; and
- shielded packages which contain built-in shielding and can be handled using conventional methods – e.g. 4 metre box and 2 metre box.

Much of the ILW inventory has the capacity to generate gas as it undergoes chemical or microbiological reactions; where appropriate the containers for such waste are vented.

Some waste owners use different encapsulants than cement for certain ILW streams, such as polymer encapsulants. Also, some waste owners are considering the use of a vitrified wasteform for certain ILW, and the possibility of disposing of certain wastes directly in suitable containers, without encapsulation. We are open to considering new options for waste conditioning and packaging, but all proposals will need to be assessed via the disposability assessment process [14] to ensure they are compatible with the safety requirements of a GDF.

A waste package specification covering HLW and SF is also available [79], and we have started to consider waste packaging options for separated plutonium and uranium. As none of these materials have yet been packaged into suitable disposal containers, a wide range of options is still under consideration. For the purposes of this generic ESC we have assumed that similar disposal containers would be used for HLW, SF, Pu and HEU. We have also assumed that DNLEU would be disposed of in similar containers to those used for ILW and LLW. Figure 4.4 (on following page) illustrates the various waste types and actual or assumed conditioning and packaging arrangements. Future waste packaging arrangements for all types of wastes are being considered as part of R&D studies, options studies [e.g. 28, 29], and disposability assessments, and will be agreed with waste producers and regulators as we move forward.

4.1.3 Geological environment

GRA Requirement R11: Site investigation

The developer/operator of a disposal facility for solid radioactive waste should carry out a programme of site investigation and site characterisation to provide information for the environmental safety case and to support facility design and construction.

Our approach to meeting GRA Requirement R11 is discussed in Section 3.1.5.

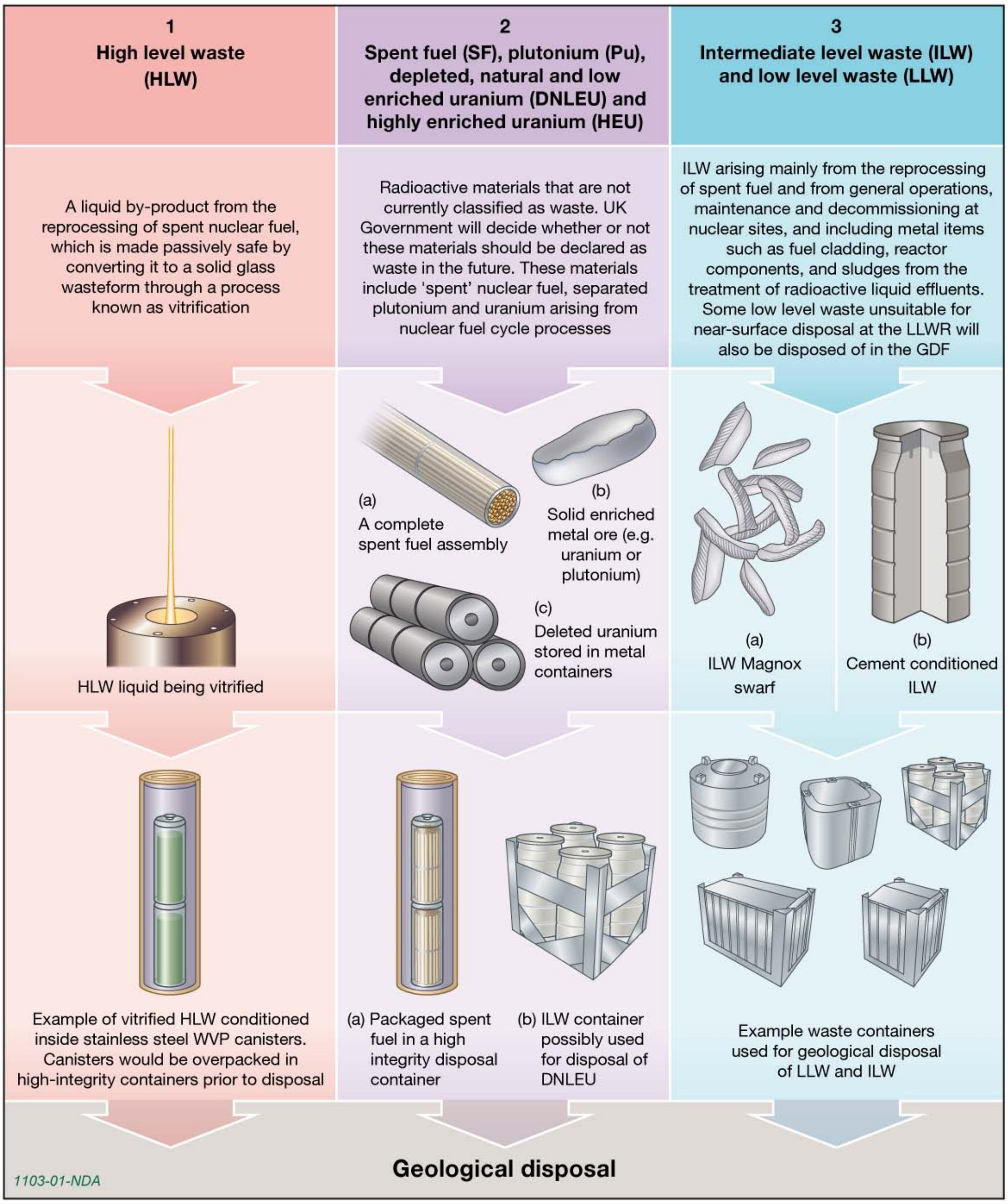
As discussed in Section 3.1.1.3, the geological environment is a key component of the multi-barrier system: it provides long-term isolation of the waste disposed of in a GDF from the biosphere and contributes to containment. The processes by which the geological environment contains radionuclides are discussed in the Geosphere status report [17] and the Radionuclide behaviour status report [20].

There is a wide variety of geological environments that might be suitable to host a GDF in the UK. By the term geological environment, we mean the host rock in which a GDF will be constructed and the surrounding geological formations. Different geological environments have different geological, hydrogeological, geochemical and geotechnical properties. We have carried out studies to evaluate the different geological, hydrogeological, geochemical and geotechnical properties of geological environments in the UK that might potentially be suitable to host a GDF [17, 137]. This has allowed us to define three generic geological environments, which are designed to capture and illustrate the range of behaviour and issues we may need to address when evaluating candidate sites (see Appendices A, B and C for details).

The geological environments that we have defined are based on consideration of the host rock formation (where the disposal areas would be built) and the cover rocks (the geological formations that occur between the disposal areas and the ground surface). At this generic stage in the implementation programme for a GDF, we consider it appropriate to use only coarse descriptions to distinguish different geological environments. As information becomes available on candidate sites, we will take into account more specific information on the geological, geotechnical, hydrogeological and geochemical characteristics that will be important for GDF design.

We have chosen the mechanical strength of the host rock as a key distinguishing characteristic at this stage as, along with the wastes requiring disposal, this has the most influence on the conceptual design of a GDF. This enables us to describe realistic disposal concepts as examples of how geological disposal could be implemented in each of the generic geological environments. The cover rocks could be important in providing additional containment and retardation of any contaminants released from the EBS, so cover rocks are included in our coarse descriptions.

Figure 4.4 Higher activity radioactive wastes and materials considered for geological disposal, and illustrative conditioning and packaging arrangements



The use of generic geological environments does not imply that any specific sites are being considered. The host rock types correspond to three distinct general rock types that occur in the UK and that are considered potentially suitable to host a GDF for higher activity radioactive wastes, based on studies carried out in the UK and overseas.

The three host rock types are:

- Higher strength rocks - these would typically comprise crystalline igneous and metamorphic rocks or geologically older sedimentary rocks where any fluid movement is predominantly through discontinuities[15].
- Lower strength sedimentary rocks - these would typically comprise geologically younger sedimentary rocks where any fluid movement is predominantly through the rock matrix.
- Evaporites - these would typically comprise anhydrite (anhydrous calcium sulphate), halite (rock salt) or other minerals that have been formed by the evaporation of surface water bodies in the geological past. Evaporites are generally found inter-layered with other, lower strength sedimentary rocks.

By 'higher strength', we mean rocks in which it would be possible to excavate stable tunnels and vaults, requiring only limited rock support, with spans of order 20 metres at GDF depths of up to 700 metres, and in which it should be possible to construct such large-span vaults at depths of up to 1,000 metres if appropriate engineering support is provided [139]. One property of higher strength rocks is that they tend to be fractured, with the degree and orientation of the fracturing being important in determining their geotechnical and hydrogeological characteristics. Fracture density in such rocks can vary significantly from rock volume to rock volume, even within the same geological unit.

By 'lower strength', we mean rocks in which it would be possible to excavate tunnels of 10 to 15 metres diameter at GDF depths, although these excavations would be likely to require considerable excavation support, possibly including full linings [139]. Fractures in lower strength rocks may tend to heal with time as a result of creep.

The strength of evaporites varies markedly with composition and the presence or otherwise of impurities. However, evaporite formations are often able to creep and seal voids. In some examples (e.g. Boulby potash mine), large-span excavations might remain stable for decades, whereas in other cases creep of the evaporites means that excavations could only be held open for a few years without extensive engineering support [139]. A key characteristic of evaporites is their isolation from mobile groundwater and extremely low permeability.

A higher strength host rock may extend all the way to the ground surface or it may be overlain by a sedimentary sequence. A lower strength sedimentary host rock may be overlain by other sedimentary rocks. For an evaporite host rock to have been preserved, it will already be isolated by low-permeability formations from any formations containing flowing groundwater, otherwise dissolution of the evaporite would have occurred.

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[15] The term 'discontinuities' applies to a range of structures of any scale that interrupt the continuity of the rock mass; the term therefore includes faults, fractures, fissures, joints, bedding planes, and shear zones.

The rocks that will lie above the host geological formation will be characterised by varying mineralogy, microstructure and permeability and, therefore, varying capacities for providing geological containment with respect to contaminants potentially released from a GDF in the long term. The overlying rocks may also trap any gas that migrates from a GDF. Conversely, some of the overlying rock layers may be aquifers, and these could potentially provide routes for aqueous transport of released radionuclides into the biosphere.[16] Such a process, if it occurred, would result in dilution of the radionuclides moving through the ground. Evaluation of the surrounding rock formations in terms of their potential for containment and retardation would form part of the surface-based investigation stage of the MRWS Site Selection Process.

Table 4.2 summarises the main characteristics of the three different host rock types that are important when considering their ability to contain radionuclides released from the near field of a GDF [17].

Table 4.2 Characteristics of different potential host rock types

Further information is provided in our Geosphere status report [17].

Characteristic	Higher strength rocks	Lower strength sedimentary rocks	Evaporites
Mechanical	High strength. Pattern of fracturing is critical to strength on excavation scale.	Low to medium strength. May swell and creep during or following resaturation.	Strength depends on composition. Deforms in a visco-elastic manner.
Hydrogeological	Permeability likely to be controlled by extent and connectivity of fracture system; fracture-dominated flow.	Generally very low permeability - solute transport likely to be dominated by diffusion, although porous-medium flow is also possible.	Permeability extremely low or undetectable.
Geochemical	Chemically stable and likely to have groundwater with low to moderate salinity. Fracture-fill minerals may have strong capacity for sorption.	Groundwater composition may vary from slightly saline to brine. Chemically stable with significant buffering and sorption capacities.	Liable to dissolution by groundwater. Low sorption capacity. Brine porewater may be highly corrosive.
Thermal	Unlikely to be altered at temperatures envisaged during normal evolution of a GDF.	May be liable to alteration at the upper end of the temperature range within a GDF.	High thermal conductivity and stable at temperatures envisaged during normal evolution of a GDF.

[16] However, as discussed in Section 2.1, note that any sites under consideration must first pass the ‘sub-surface unsuitability’ test in Stage 2 of the MRWS Site Selection Process [4]. Application of the ‘unsuitability’ criteria excludes sites from further consideration where all or part of the GDF host rock is located within a groundwater aquifer, or where all or part of the GDF host rock would be provided by permeable formations that might reasonably be exploited in the future

4.1.4 Engineered barrier system

GRA Requirement R8: Optimisation

The choice of waste acceptance criteria, how the selected site is used and the design, construction, operation, closure and post-closure management of the disposal facility should ensure that radiological risks to members of the public, both during the period of authorisation and afterwards, are as low as reasonably achievable (ALARA), taking into account economic and societal factors.

Our approach to meeting GRA Requirement R8 is discussed in Section 3.1.2. As discussed there, it is not yet possible to address this requirement. However, we summarise here the work we have undertaken to inform ourselves about those aspects of the engineered components of a GDF that are central to a demonstration of environmental safety and that will need to be optimised once we have a preferred site and disposal concept. We also explain the rationale behind selection of the illustrative geological disposal concept examples, and highlight particular safety functions inherent in engineered components of these concepts. Note that the wasteform, waste packages and waste acceptance criteria are discussed in Section 4.1.1 and 4.1.2, but we consider here the engineered barrier system (EBS) in a more holistic sense.

We have carried out a series of options studies to explore the different geological disposal concepts being developed or implemented around the world and that it might be appropriate to implement in each of the potentially suitable host rock types set out in Section 4.1.3 [28, 29, 30]. We have used these studies to identify a set of illustrative geological disposal concept examples for the three potential host rock types to support the generic DSSC. We selected concepts that were well developed and supported by extensive R&D, that allowed for retrieval, and that had been subject to detailed safety assessment, regulatory scrutiny and international review. The illustrative examples are listed in Table 4.3, and the attached notes provide further information on why these examples were selected.

Each of these illustrative geological disposal concept examples is based on the principles of isolation of the wastes and containment of radionuclides and other contaminants using a set of engineered barriers that will complement the natural barrier provided by the geology at the site. The common features of an EBS – wasteforms, waste containers, overpacks (only in some concepts for some waste types), buffers/backfills, mass backfill, seals – were set out in Section 3.1.1. The key features of the UK application of the illustrative disposal concept examples, including the engineered barriers considered in them and the safety functions they provide, are summarised in Appendices A, B and C. We do not intend that any of these illustrative disposal concepts will necessarily be implemented. When we know the geological setting for a GDF, we will select and develop appropriate disposal concepts for the UK higher activity radioactive wastes requiring geological disposal, that may or may not be based on these illustrative disposal concept examples.

With regard to future optimisation of the EBS, we note that some wasteforms have already been created and the wastes suitably packaged under our disposability assessment process, as discussed in Section 3.1.2 and Section 4.1.2. Further wastes will continue to be conditioned and packaged before we have a preferred site and disposal concepts (at the end of Stage 5 of the MRWS Site Selection Process). There is therefore a much greater degree of flexibility for other aspects of the EBS with regard to optimisation. The overall design of the EBS will be optimised once we have a preferred site and disposal concepts, including conditioning and packaging arrangements for wastes still to be packaged.

Table 4.3 Illustrative geological disposal concept examples for different waste types

See [138] for the NEA review of Nagra's assessment of the Opalinus Clay disposal concepts.

Host rock	Illustrative Geological Disposal Concept Examples[d]	
	ILW/LLW	HLW/SF
Higher strength rocks[a]	UK ILW/LLW Concept *(NDA, UK)*	KBS-3V Concept *(SKB, Sweden)*
Lower strength sedimentary rock[b]	Opalinus Clay Concept *(Nagra, Switzerland)*	Opalinus Clay Concept *(Nagra, Switzerland)*
Evaporites[c]	WIPP Bedded Salt Concept *(US-DOE, USA)*	Gorleben Salt Dome Concept *(DBE-Technology, Germany)*

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 review regarded 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.

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Decisions on the materials to be disposed of in a GDF and on waste packaging place certain constraints on the rest of the EBS and the facility layout (see Section 4.1.5). For example, these constraints relate to:

- ensuring that the temperature within the disposal areas remains within acceptable limits;
- ensuring that any gas generated within the disposal areas is able to escape without resulting in significant harm during the operational phase or damaging the barriers during the post-closure period; and
- minimising the potential for a criticality event to occur.

Details of the requirements and constraints on the disposal system, including the EBS, are given in the Disposal System Technical Specification [23].

The range of safety functions that could be provided by individual engineered barriers is discussed in more detail in Section 5.1.1 and Section 5.2.1 and in the Disposal System Technical Specification, research status reports and disposal concept options reports.

4.1.5 Design, construction, operation and closure of underground facilities

> ***GRA Requirement R12: Use of site and facility design, construction, operation and closure***
>
> **The developer/operator of a disposal facility for solid radioactive waste should make sure that the site is used and the facility is designed, constructed, operated and capable of closure so as to avoid unacceptable effects on the performance of the disposal system.**

Our approach to meeting GRA Requirement R12 is discussed in Section 3.1.3.

The design, construction, operation and closure of the underground facility will be site and host rock dependent. As outlined in Section 4.1.3, the geotechnical properties of the host rock will be a key constraint on the design, construction and operation of a GDF. These determine the maximum dimensions of the excavations and the type of excavation support required. As described in Section 4.1.4, the EBS design will be matched to both the site (geological environment) and the wastes.

During the late 1990s Nirex carried out a study of the various large-scale underground caverns that had been excavated around the world [139]. This study covered openings excavated for a range of purposes (hydrocarbon storage, hydro-electric power stations, mines, transport) in a range of different rock types. This 'cavern precedent study' has been used to help describe and categorise - in terms of 'strength' and ability to support openings of various sizes - the three host rock types we are considering (see Section 4.1.3).

The dimensions of the disposal tunnels would be determined by the mechanical properties of the host rock and the requirements of the disposal concept. The number of tunnels required to accommodate the inventory would depend on the combination of tunnel diameter, disposal concept and thermal constraints, which may control the waste package spacing. In general, the lower the rock strength, the smaller the tunnels and the larger the resulting plan area of the underground facilities. However, other factors, such as stress field magnitude and orientation and the presence of fault and fracture zones would also influence the layout and required areal extent of the underground facilities of a GDF.

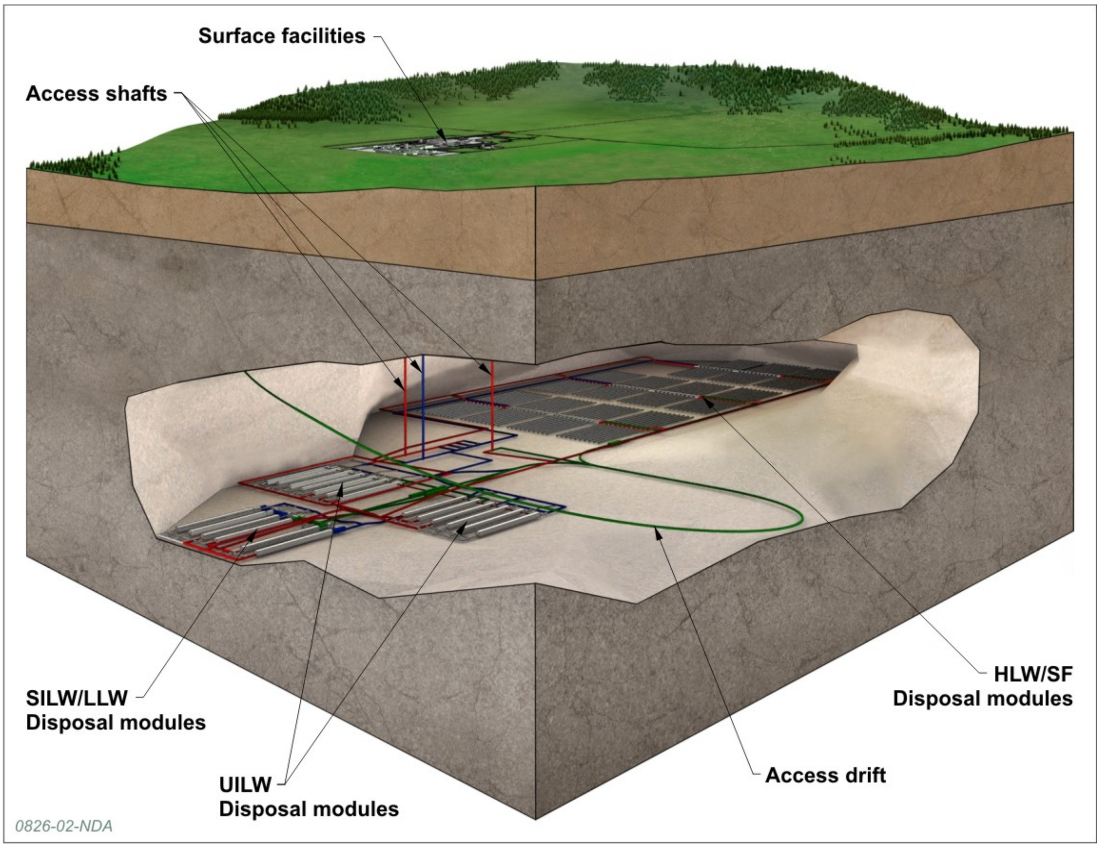
We expect to be able to construct a GDF using conventional excavation technology such as that used to construct large tunnels and caverns in other industries and by other waste management organisations.

As explained in the introduction to Section 3, the UK Government sees no case for having more than one GDF if one facility can be developed to provide suitable safe containment for the entire inventory [4]. Our assessment basis therefore considers a single GDF. We have assumed that there would be two distinct disposal areas implementing different disposal concepts (or EBS designs), one for LLW, ILW and DNLEU, and the other for HLW, SF, HEU and Pu. These disposal areas would share common service areas, access ways and surface facilities, but would be separated from each other by a distance that would ensure that the impact of interactions between the two disposal areas was sufficiently small that it did not compromise the overall performance of a GDF. Figure 4.5 shows a schematic illustration of such a GDF.

For planning purposes we assume that it would take of the order of 90 years to emplace the current Baseline Inventory [23]. Backfilling could take up to 10 years. Our waste emplacement and backfilling strategies are flexible and will be kept under review until such time as we are ready to begin waste emplacement and have received an environmental permit to do so. Our current thinking is that initially only ILW and LLW would be emplaced because of the need for waste producers to clear their stores before the buildings and facilities would require significant refurbishment. Emplacement of HLW and SF may not need to begin until several decades after the start of GDF operations, based on consideration of the volume requiring disposal, the emplacement rate, and an assumed end period to coincide with full emplacement of the Baseline Inventory of ILW and LLW requiring geological disposal. The cooling time for HLW and SF is also a factor affecting their emplacement timing.

Figure 4.5 Schematic illustration of the layout of a generic GDF for UK higher activity radioactive wastes

There are separate disposal areas for shielded ILW and LLW (SILW/LLW), unshielded ILW (UILW), and HLW and SF because of the different handling and disposal requirements of these waste types.



The OSC [6] describes the ways in which we will ensure safety during the construction and operational phases of a GDF. Wherever possible, procedures and activities will be designed to provide passive safety and ‘fail safe’ modes.

During the operational period, a GDF would be ventilated and exhausted via a controlled discharge. The main radioactive gases are expected to be tritium, methane containing carbon-14, and radon-222. Hydrogen and methane represent the main non-radiological hazardous releases.

We anticipate that high-efficiency filtration and other abatement measures should be 99.99% effective at removing vapours and particulates from atmospheric discharges. Updates to the ESC will include consideration of the potential routes for entry of such material to the environment during the operational period (when a preferred site-specific disposal concept is available) and the site-specific characteristics required to assess the impacts of any discharges.

Discharges during the operational period of radioactive liquids or radioactive particles suspended within a liquid are not considered quantitatively within this generic ESC. The design of the disposal containers and GDF would ensure that significant quantities of liquid effluent did not arise from emplaced wastes during routine operations. During the operational period, potentially contaminated liquids and condensates in the underground facilities would be collected using Best Available Techniques, monitored, and transferred to the surface. Effluent treatment facilities at the surface would be designed with optimised abatement systems to prevent or minimise both radiological and non-radiological impacts on the public and the environment.

Consistent with UK Government policy, our current planning and designs maintain flexibility to keep open the option of retrievability, and we are keeping any implications for the packaging of wastes under review. As noted in Section 3.1.3.3, excavations in certain rock types would require extensive engineered support to keep a GDF (or the disposal areas within it) open for extended periods. This is particularly so for evaporites and some lower strength sedimentary rocks that exhibit rock creep.

The closure engineering for a GDF would consider the requirements for backfilling and the installation of seals at key locations within the excavations, as discussed in Section 4.1.4 (see also the Disposal System Technical Specification [23] and the Generic disposal facility designs report [24]).

4.1.6 Active and passive institutional controls

Institutional controls can be active or passive. Active controls are those by an authority or institution permitted under EPR 2010 [2], involving monitoring, surveillance and remedial work at a GDF site, as necessary, as well as control of land use. Passive controls are those that, once taken, do not require any further involvement of the authority or institution under EPR 2010; they apply mainly to the period after the environmental permit is surrendered. Examples of passive controls include retention of information about the site as part of government archives and maps, or in records centres or libraries, government ownership or control of land, and establishment of durable site marker systems (e.g. [140]).

For this generic ESC, this section is relatively brief, and is mainly here as a placeholder for future updates of the ESC.

During the pre-operational and operational periods of a GDF, the institutional controls would be the same as those adopted for any other nuclear facility in the UK. Prior to the operational period, a GDF would become a nuclear licensed site which, in conjunction with the environmental permit, would establish the requirement for operational institutional controls. Any claim for active institutional control would be supported by detailed forward planning of organisational arrangements and a suitable demonstration of funding arrangements. In this regard, we note that funding for the development of a GDF for the disposal of legacy wastes would come from the UK Government through the NDA.[17]

What happens to the site following closure would be a matter for future generations – the site could be farmed, forested, allowed to return to nature, or used for construction or other purposes. The waste would remain isolated within the multi-barrier system in the geological formations hundreds of metres below the ground surface. We would provide records of the location and general contents of the facility to local records offices and the National Nuclear Archive.

The site could also be retained under active control for some period of post-closure monitoring if desired by future generations living in or near the host community. However, the ESC makes no reliance on any post-closure period of active institutional control.

4.1.7 Biosphere

As noted in Section 3.1.1, the biosphere is generally taken to include the atmosphere and the Earth's surface, including the soil and surface water bodies, seas and oceans. The biosphere is an important component of the system being assessed because of its role in determining potential exposure routes once radionuclides enter it from the underlying rocks. As we are at a generic stage of development with no specific site, no detailed

[17] The mechanism for funding disposal of higher activity radioactive wastes arising from the proposed programme of new nuclear power plant development in the UK is still to be defined. However, we expect that the cost of such disposal would be funded by the plant operators.

information on the biosphere is available. Our approach to representing the biosphere in assessing releases from a GDF is described in the Biosphere status report [18].

For the OESA we would need to consider the present-day biosphere at the candidate site(s) since this is the environment that would be affected by any operational discharges from a GDF. Other components of our GDF programme, such as Strategic Environmental Assessment (SEA) and Environmental Impact Assessment (EIA), would also require a description of the present-day biosphere, and we will ensure that the descriptions used in these different studies are fit for purpose and consistent. At the current generic stage, we make use of a series of internationally agreed reference assumptions that are used for assessments of other UK nuclear installations.

Even after we have a site and have characterised it, there will still be considerable uncertainty in the characteristics of the biosphere at times in the far future when any contaminant releases from a GDF would reach the accessible environment. These uncertainties could concern such things as natural changes in landscape driven by climate processes, or changes in the way people interact with the biosphere (e.g. what crops are grown). Such changes are impossible to predict over the timescales considered in the PCSA. Given the uncertainties inherent in projecting the long-term evolution of the biosphere, we adopt a stylised approach in the PCSA, as discussed in Section 3.2.2.4. We aim to keep the approach as simple as we can justify for the PCSA, whilst demonstrating an appropriate present-day understanding of the system for the OESA and other required environmental assessments (SEA/EIA).

Our PCSA needs to consider the possible flux of GDF-derived contaminants from the geosphere into the biosphere. At some sites we would expect this flux to occur within a localised area, perhaps associated with faults, whereas in others this flux could be much more diffuse, with the area over which it occurs possibly changing with time even in the absence of significant climate change [137]. Our illustrative generic PCSA calculations consider this uncertainty. In addition, the amount of dilution that occurs at the geosphere-biosphere interface will also differ between sites. For example, water wells may be important in some environments, whereas in others there may be no ‘aquifer’ formation that could plausibly support a well. Our generic PCSA calculations also consider the uncertainty associated with the amount of dilution a flux from the geosphere will see when it enters the biosphere.

In the generic PCSA, we do not explicitly model the pathways and uptake by humans of radionuclides discharged into the biosphere. Instead we use a set of reference biosphere dose conversion factors that convert unit input of radioactivity into the biosphere into an annual radiological dose. These dose conversion factors have been derived from a separate generic and simplified reference biosphere assessment model suitable for the UK, as described in detail in the Biosphere status report [18], and summarised in Section 4.3.2.1.

4.2 Scientific and technical information and understanding

Our confidence that we will be able to demonstrate the safety of a suitably designed and optimised GDF for higher activity radioactive wastes in the UK is based on an extensive body of scientific understanding and engineering experience, to which we will continue to add [85]. This knowledge stems from our own work, from the work of Nirex, from experience gained by overseas waste management organisations and from the work of international collaborative projects. Our current understanding of the various technical areas we will need to consider as we move forward is summarised in the various reports that underpin the ESC, most notably in the research status reports. At this stage we are using existing research and site characterisation data collected by ourselves, previously by Nirex, and by others to underpin our understanding of the likely behaviour of the geological disposal system. However, the representation of the geological disposal system itself at this generic stage is necessarily stylised. We have, however, attempted to make the representation 'cautiously realistic' so as to provide an understanding of the environmental safety of a GDF that can make a useful contribution to the ongoing discussions that are shaping the programme.

Some of the FEPs that impact on safety, and hence GDF design, are better understood than others, and this will continue to be the case as we move forward. We will need to take account of this uncertainty in our design and assessments. We will carry out targeted R&D and site characterisation to provide us with the site-specific and other information we will need to design, construct and operate a GDF safely and be confident of its long-term safety after closure, as discussed in Section 3.1.5 [85, 86]. However, even following an extensive programme of R&D and site characterisation, we are unlikely to be able to resolve all the uncertainties associated with system performance. For example, it may not be possible to measure the parameters concerned under the appropriate (*in situ*) conditions on the length scales and timescales of interest, or the act of making the measurements may carry an unacceptably high risk of damaging the site. If we cannot resolve particular uncertainties that potentially impact adversely on system performance, we would design the EBS to be robust to the uncertainties concerned (see Section 3.2.2.3).

We also recognise that data we acquire may support the establishment of more than one conceptual model of a feature, event or process. Our understanding of the structure and processes at candidate sites is likely to be incomplete until well into Stage 6 of the MRWS Site Selection Process. We will carry forward in our assessments the uncertainty associated with alternative conceptual understanding or models that might possibly explain the data, as discussed in Section 3.2.2. Where such uncertainty could have a significant impact on decision-making, we will consider whether further R&D or site characterisation could distinguish between the alternative conceptual models, or whether the parts of the geological disposal system that we control could be modified in such a way to 'design out' the uncertainty.

In cases where uncertainties cannot be quantified over the timescale of our assessments, such as the consideration of future human actions, the behaviour of potentially exposed groups, and changes in the biosphere (particularly in the post-closure period), we adopt stylised approaches, consistent with the GRA [1].

4.2.1 Description of expected evolution

The expected evolution of a GDF depends on the geological environment, GDF design, and any external environmental change that could impact on the geological environment and/or the GDF at a particular location.

We describe the expected evolution of a generic geological disposal concept in terms of three periods:

1. construction and operational period;
2. early post-closure period (establishment of long-term conditions, to a few tens of thousands of years, depending on disposal concept); and
3. late post-closure period (to one million years and more).

At this generic stage there is little point in providing a more detailed breakdown of the post-closure period, not least because different characteristics of the illustrative geological disposal concept examples mean that they would evolve at different rates.

1. Construction and operational period

An excavation-disturbed zone would form as a result of GDF construction. The extent and future evolution of this would depend on the properties of the host rock, the excavation method and degree of excavation support provided and its timing. Excavation and fitting out of new disposal areas would occur in parallel with the emplacement of waste in other disposal areas. The layout of the underground facilities would be designed to ensure that these two activities can be carried out safely at the same time.

During the operational period, the environment within the underground facilities would be controlled by drainage and ventilation systems. There would be some gaseous releases from corrosion and microbial waste degradation, which would be removed via the ventilation system. The host rock around the underground facilities would desaturate during this period and oxidation reactions may occur; however, disposal areas that have been filled with waste and closed off may begin to resaturate while disposal operations and desaturation are taking place elsewhere. (Note, in an evaporite host rock there would be very limited mobile groundwater present and resaturation would not be relevant.)

2. Early post-closure period

Following closure the disposal areas would fully resaturate. The time required for resaturation will depend on the properties of the host rock and on the design of the disposal areas (geometry and amount of void space). In some host rocks the disposal areas may fully resaturate on a timescale of years to decades, whereas in other host rocks full resaturation may take tens of thousands of years. (As noted above, in an evaporite host rock, there would be very limited mobile groundwater present and resaturation would not be relevant.)

Once ventilation was stopped, the temperature in the HLW/SF disposal areas would start to rise owing to radioactive decay of short-lived radionuclides in the inventory. As a result the temperature in the disposal areas and the surrounding rock may rise to several tens of degrees above ambient temperature, before slowly decreasing. The peak temperature in the EBS is likely to occur within a few decades of waste emplacement, but we expect temperatures in the HLW/SF disposal area to remain above ambient for around 10,000 years. There would be a much smaller temperature rise in the ILW disposal areas.

Long-term chemical conditions in a GDF would become established and barriers would evolve to their long-term forms (e.g. swelling of bentonite buffers). The engineered barriers would provide containment, although they would be slowly degrading as they interact with each other and with any groundwater in the host rock. Significant volumes of gas may be generated from corrosion and microbial waste degradation in the ILW disposal areas. If gas is unable to escape from a GDF, the pressure within it may build up until it exceeds the lithostatic pressure at that depth in the rock (referred to as ‘over-pressurisation’). This could temporarily affect the pattern of groundwater flow and rock stress around a GDF, until any pressure build-up within it was relieved. Dissolved radionuclides would be released from ILW packages via the vents, but we expect HLW/SF waste packages to remain intact in the early post-closure period and to provide complete containment.

The excavation-disturbed zone may partially or completely heal as a result of creep.

The geosphere would isolate a GDF from the biosphere and, together with the EBS, completely contain, or retard, the transport of any dissolved and gas-phase contaminants.

3. Late post-closure period

The EBS is likely to become increasingly ineffective as it evolves (e.g. see Figure 3.2), and contaminants would gradually be released from the EBS into the geosphere. However, the evolving EBS would continue to provide retardation; for example, a degraded low-permeability or chemical buffer may continue to provide surfaces for sorption. Gas generation is expected to have ceased or become negligible.

The most likely cause of a major environmental change in the UK potentially significant enough to have an impact at GDF depths is major re-glaciation of the UK landmass. This is unlikely to occur for 100,000 years and could occur much later than this [17, 18], at which time the EBS may be only partly effective. However, a GDF would be designed to ensure that the EBS remains effective until the majority of the inventory has decayed. The geosphere may continue to provide complete containment long beyond this, and thereafter provide retardation of the transport of radionuclides to the surface environment.

Issues specific to these three periods for the illustrative geological disposal concept examples as applied to the UK are summarised in Appendices A, B and C. Further detail is provided in our research status reports and, in particular:

- the Package evolution status report [15] provides details of our understanding of the expected evolution of waste packages in a GDF and the processes by which radionuclides are released from the wasteform;
- the Near-field evolution status report [16] explains how the different components of the EBS in the illustrative geological disposal concept examples might evolve;
- the Geosphere status report [17] explains the key mechanical, hydrogeological, geochemical and thermal features of the illustrative geological environment examples and how they could affect the evolution of a GDF;
- the Gas status report [19] provides information on the processes that control gas generation rates and the migration of gas; and
- the Radionuclide behaviour status report [20] provides details of our understanding of the processes that could retard radionuclide transport.

4.2.2 Monitoring

GRA Requirement R14: Monitoring

In support of the environmental safety case, the developer/operator of a disposal facility for solid radioactive waste should carry out a programme to monitor for changes caused by construction, operation and closure of the facility.

Our approach to meeting GRA Requirement R14 is discussed in Section 3.1.6, including consideration of how we will go about developing a monitoring plan, monitoring objectives, monitoring timescales, and use of the results.

For this generic ESC, this section is mainly here as a placeholder for future updates of the ESC. In future updates of the ESC we plan to summarise our monitoring plan and the results of any monitoring to date, demonstrating how those results support our developing understanding of the candidate site(s) and the impact our activities are having on it.

4.3 Methods, models, computer codes and databases

This section describes the types of quantitative analyses we consider to be appropriate at this stage of the programme. These methods and models have been applied in the generic OESA [26] and the generic PCSA [27], which provide quantitative support to the largely qualitative safety arguments we are making in this generic ESC. Our illustrative assessments aim to show how the information that is likely to be available to us at Stage 4 of the MRWS Site Selection Process could be used to build confidence in the safety of the geological disposal system at candidate sites, and provide guidance to the development of the site characterisation, R&D and engineering design programmes. As described in previous sections, the detail of the quantitative analyses we will undertake and report in the ESC will evolve as we start to obtain site-specific information from the candidate site(s) and to develop site-specific disposal concepts.

At this early stage in the MRWS Site Selection Process, it is more important to understand the relationship between different possible geological environments / disposal concepts and post-closure environmental safety than operational environmental safety because the geological environment is a key part of the safety concept for the post-closure period, whereas it plays a relatively limited role when considering environmental safety during the operational period (see Section 3.1.1). As already noted, our assessment basis and the quantitative environmental safety assessments we have performed in this generic ESC are considerably more thorough for the post-closure period than for the operational period. This is in line with adopting an overall assessment approach that is fit for purpose at every stage of the process.

4.3.1 Operational environmental safety

The generic OESA [26] describes the types of operational discharges we envisage and the calculations we have undertaken to evaluate the off-site consequences of routine radiological discharges during the operational period. As location-specific GDF designs are developed, we will keep under review the potential for non-radiological discharges.

The generic OSC [6] considers the off-site consequences of accidental releases from potential fault scenarios, and describes our methodology for the calculation of dose release ratios (dose per unit release of activity) for such releases. As described in the OESA [26], we use the same methodology for the calculation of doses to the public from routine releases in this generic ESC (see Section 5.1.2).

We have determined that PC-CREAM [125] is fit for the purpose of providing an indication of the likely radiological impact on members of the public from discharges to the environment from normal operation of a GDF. PC-CREAM comprises a suite of modules that can be used to perform radiological dose assessments from routine and continuous discharges from virtually any type of installation, including nuclear power plants and nuclear fuel cycle facilities. PC-CREAM was developed for the EC, but parts of the system have been used throughout the world. PC-CREAM was also used to calculate the dose per unit release values that are presented in the Environment Agency's methodology reports [97, 98], which we have used in our generic OESA calculations.

4.3.1.1 Aerial discharges

The design of a GDF would include a ventilation system to provide adequate ventilation underground, and to other manned areas, at all times, and which would be used and maintained throughout the operational period. The presence of the ventilation system would result in aerial discharges of radioactive contaminants in gaseous form via a stack. Illustrative calculations of off-site radiological impact have been undertaken based on discharges for tritium, carbon-14 (as methane) and radon-222. Doses are calculated for a critical group consisting of a local resident family, and considering exposures from inhalation of activity from the plume, external radiation from the plume, and external radiation from deposited activity, using the PLUME module of PC-CREAM. Doses from the ingestion of contaminated food as a result of aerial discharges have also been calculated using the FARMLAND module of PC-CREAM.

4.3.1.2 Liquid discharges

The design of a GDF would include a liquid effluent treatment and discharge plant as part of the surface facilities. Liquid effluents are anticipated to arise from waste package handling activities at a GDF and from condensation in the ventilation system. Estimated discharges from the liquid effluent treatment and discharge plant could be entered into either the DORIS module (for discharges into the marine environment) or the ASSESSOR module (for discharges into river environments) of PC-CREAM to provide an indication of potential radiological impacts. Such calculations have not been performed for this generic ESC, but we could conduct such calculations as appropriate in future updates to the ESC.

4.3.1.3 Non-human biota

We use the EC's ERICA tool [141] for the assessment of radiological impacts on non-human biota (fauna and flora) to a range of 'reference organisms' appropriate for a terrestrial biosphere. Extensive databases of the occupancy, geometry, concentration factor and dose conversion coefficients necessary to assess dose rates to such organisms have been established over the last decade as part of a series of EC-funded projects, most notably ERICA. Potential pathways from the initial discharge via the discharge stack are as follows:

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

4.3.2 Post-closure safety

Our quantitative post-closure safety analysis is provided in the generic PCSA [27]. This contains illustrative example calculations and/or qualitative discussion for groundwater-mediated, gas-mediated and human intrusion-mediated releases from a GDF. The discussion is set within the context of the illustrative geological environments under consideration, and encompasses a wide range of possible disposal concepts. It also considers criticality safety in the post-closure period. We have access to a range of databases, both our own databases and international databases, which we have drawn upon to support and parameterise our calculations, as described in the PCSA report [27]. We are able to use the generic PCSA calculations to illustrate some aspects of the process of carrying out a performance assessment, but the main driver for including a quantitative assessment now is to inform our disposability assessments of packaging proposals for waste that is being packaged now, in advance of the identification of a site. We intend to use the generic PCSA calculations as the benchmark for undertaking future assessments as part of the disposability assessment process. The methodology for this is summarised in Section 3.1.4. Our generic PCSA summarises the similarities and differences in the current generic assessments with the approaches adopted by Nirex for previous generic assessments conducted to underpin earlier disposability assessment work.

The assessment approach used in this generic ESC will be applied in Stage 4 of the MRWS Site Selection Process. This is because we will not have detailed knowledge of candidate sites that emerge in Stage 4 beyond that which already exists in the public domain. The extent of the candidate site(s) may also be large. We will use existing geoscientific knowledge of the candidate site(s) to assign typical hydrogeological (and other) parameter values (and uncertainty ranges) that would be associated with the geology believed to be present at the site(s), using the approach described in the PCSA and summarised here. During Stage 5 of the MRWS Site Selection Process, our assessments would become increasingly detailed as the possible location(s) of a GDF within the candidate site(s) become better defined and disposal concepts are further developed to reflect site conditions. During Stage 5 we expect to be in a position to significantly develop our representation of the geological environment at the candidate site(s). This will enable us to derive appropriate site-specific hydrogeological parameter values for the site(s), rather than having to assign typical values as in this generic ESC.

The following subsections describe the approach used in the generic PCSA for assessing radionuclide transport by groundwater, the consequences of gas, inadvertent human intrusion, and criticality safety.

4.3.2.1 Radionuclide transport by groundwater

For a GDF situated in water-saturated rocks where there is groundwater movement, the dissolution and transport of radionuclides in groundwater is likely to be the most significant mechanism by which radionuclides could eventually return to the surface environment. The **groundwater flow field** at a particular site will be highly dependent on the rock strata present, the hydrogeological properties of those rock strata (such as their permeability and porosity), and the natural driving forces for groundwater movement (such as surface topography and any pressure or salinity gradients).

The three illustrative generic geological host rocks discussed in Section 4.1.3 (higher strength rocks, lower strength sedimentary rocks and evaporites) are likely to have different groundwater flow field properties, as outlined in Table 4.2 and discussed in more detail in the Geosphere status report [17]. In evaporites there is likely to be little or no mobile groundwater present in the host rock and, hence, radionuclide transport by groundwater would not be a significant issue. Lower strength sedimentary rocks in the UK are likely to be water-saturated, but may be of sufficiently low permeability that there is little groundwater movement. In such an environment, radionuclide transport may be dominated by diffusion processes. Higher strength rocks are generally water-saturated and may contain relatively permeable rock strata and/or fractures through which significant groundwater movement can occur.

In order to represent radionuclide transport by groundwater at the current generic stage of the MRWS Site Selection Process, we need an appropriate way of representing a generic groundwater flow field that is fit for use as part of our disposability assessment process. As discussed in Section 3.1.4.3, we have developed reference benchmark models for the PCSA in such a way that calculated doses are likely to be conservative - that is, for an actual site, they would be unlikely to be greater than those calculated using the reference models and reference parameter values. The reference model for the groundwater flow field has been developed consistent with this approach. Therefore, we have chosen to base our model representation on a possible flow field for a higher strength host rock setting, as this is the most conservative approach in terms of radionuclide transport by groundwater. Groundwater flows are likely to be greater for such a host rock than for a GDF located in a lower strength sedimentary host rock or an evaporite host rock.

We have defined the following four parameters to represent the properties of a generic groundwater flow field that would, when combined with the transport and retardation properties of radionuclides, determine the performance of the geological barrier for our generic assessment of radionuclide transport by groundwater:

q – the specific discharge (in metres per year) through the undisturbed host rock at the location of the GDF. This provides a measure of how the rate of groundwater flow through the EBS could affect calculated results.

T – the groundwater travel time (in years) from the GDF to the surface environment. This provides a measure of how the containment capability of the geological environment could affect calculated results.

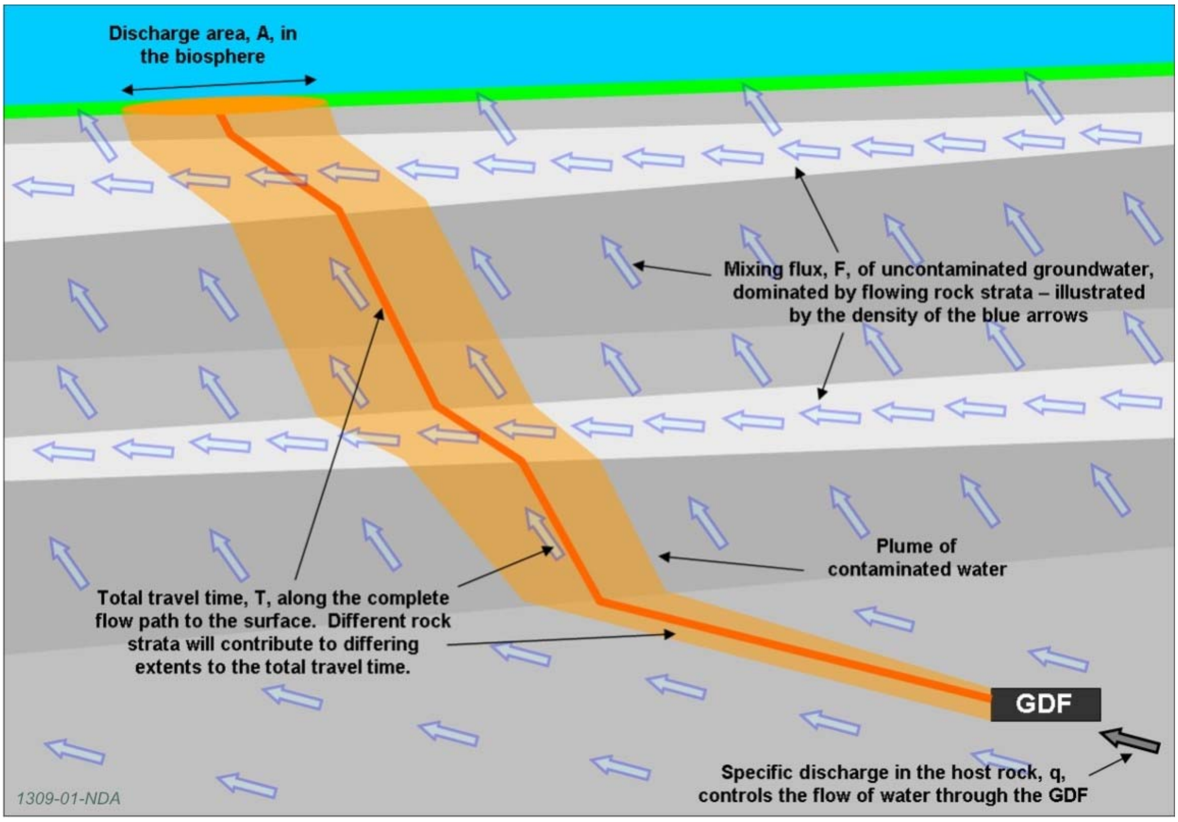
F – the groundwater mixing flux (in cubic metres per year) in the overlying rocks. This provides a measure of how the dilution potential of geological units overlying the host rock could affect calculated results.

A – the discharge area (in square metres) into which the contaminant plume is released at the surface. This provides a measure of how the characteristics of the geosphere-biosphere interface could affect calculated results.

These four parameters are illustrated in Figure 4.6.

Figure 4.6 Illustration of how the parameters q, T, F and A relate to an example groundwater flow field and migrating radionuclide plume

The blue arrows indicate the direction and magnitude of groundwater movements on a vertical section through an illustrative geological environment composed of rock strata with different hydrogeological properties. Groundwater flow is greatest where the density of arrows is greatest. The darker grey shading illustrates lower-permeability rock strata that could potentially overlie the host rock.



Together, the parameters q, T, F and A represent the way in which the groundwater flow field controls the movement of any radionuclides released from the engineered barriers in a GDF. In addition to these four parameters representing the groundwater flow field, there are radionuclide-specific parameters that also influence the transport of radionuclides in groundwater. For example, some radionuclides will take considerably longer than the groundwater travel time (T) to reach the surface environment as they sorb to rock surfaces and undergo processes such as rock matrix diffusion and anion exclusion. These processes are represented in our models and are discussed in more detail in the generic PCSA [27].

We also need to represent the performance of the engineered barrier system, again in the absence of a site-specific EBS design. In terms of the impact on radionuclide transport in groundwater, an important property of the EBS is the period over which radionuclides are contained by the engineered barriers and, therefore, not accessible to be transported in groundwater. On this basis, we have defined a fifth parameter as follows:

C – the time (in years) for which the waste container provides substantially complete containment of radionuclides. This provides a measure of how the containment capability of an EBS could affect calculated radionuclide transport results. We have taken a conservative approach to assigning reference parameter values for C, again in line with the approach to disposability assessment as set out in Section 3.1.4.3. For example, most ILW containers are vented to allow the escape of gas, and we currently do not have confidence in ascribing a reference value of C greater than zero. Ascribing a value of zero maximises the availability of radionuclides for transport by groundwater. In contrast, we expect that the container materials under consideration for HLW and SF could provide a significant period of containment (tens of thousands to hundreds of thousands of years or more) and, therefore, we can justify much longer containment times for HLW and SF containers [15].

As discussed in the generic PCSA [27], there are many other processes and parameters in the models used to carry out these calculations, representing chemical and physical processes in the EBS, geosphere and biosphere, such as solubility limitation in the EBS, sorption to buffer/backfill materials, the effect of organic materials on solubility and sorption, instant release fractions from SF for certain radionuclides present on the surface of the fuel that are more readily dissolved, diffusion through the buffer for the illustrative geological disposal concept example based on disposal of HLW/SF, and sorption to rock in the geosphere. Note that some of these processes have the potential to provide considerable containment within the EBS for many radionuclides after the waste container has failed (e.g., sorption in the backfill, solubility limitation), and these are represented in our models by appropriate radionuclide-specific parameter values. We have assigned all such other parameters either fixed deterministic values in all simulations, or fixed probability distribution functions representing uncertainty, using information from our research programme or recent assessments carried out by Nirex or by overseas waste management organisations. This gives us a reference dataset for these parameters against which we explore the sensitivity of varying the probability distribution functions for q, T, F, A and C. The fixed parameter values and uncertainty distributions making up our reference dataset are tabulated in the generic PCSA [27].

At this generic stage, we consider it appropriate for the quantitative analysis to explore the sensitivity of performance measures such as risk primarily to the parameters q, T, F, A and C. In this way, the results of the calculations can inform site assessment work in Stage 4 of the MRWS Site Selection Process and our initial planning of site characterisation work in Stage 5, as well as our future disposability assessments.

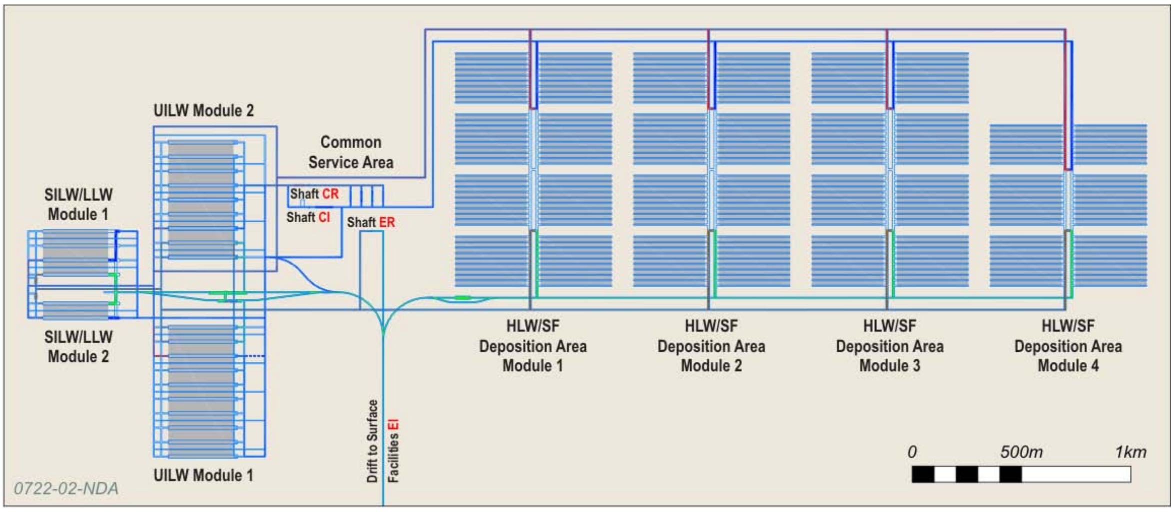
In the calculations carried out for the generic PCSA, we assumed that the transport through the geosphere of contaminants dissolved in groundwater (solutes) is mainly by the movement of the groundwater. This would not be the case for all the UK geological environments we might need to consider - contaminant transport could be much slower and dominated by diffusion in effectively static porewaters in some geological environments. However, we can derive effective values for parameters such as T for environments in which solute transport is dominated by diffusion (e.g., as would be typical for a lower strength sedimentary host rock) by considering parameters such as the diffusion distance and the effective diffusivity of the species concerned. This is a reasonable approximation to make for the sorts of illustrative calculations that are appropriate at the current time. However, should we need to consider such an environment as a candidate site at later stages in the MRWS Site Selection Process, we would develop a more appropriate model based on the site characteristics, when known.

Once we have sufficient site-specific hydrogeological and hydrochemical data and have selected site-specific disposal concepts, we will use that information to build a realistic understanding of the geosphere and containment characteristics at the site. This would then form the basis of our modelling, replacing the q, T, F, A, C approach used here. Different sites may require different approaches and tools. For example, should solute transport in the geosphere at the candidate site(s) turn out to be dominated by diffusion, we would modify our performance assessment approach to include a more accurate numerical representation of diffusion.

We have had to assume a layout for the disposal areas of a GDF to define the dimensions to use in our models; we have used the conceptual layout for a GDF based on the illustrative geological disposal concept examples for HLW/SF and ILW/LLW disposal in a higher strength host rock, as presented in the Generic disposal facility designs report ([24], see Figure 4.7). In this conceptual layout, the plan area of the underground facilities (sometimes referred to as the 'footprint') is approximately 5.6 square kilometres for the derived inventory reference case. Note that the plan areas for the generic GDF underground layouts developed for application to other geological environments are different, but these differences are not significant at the level of detail of our illustrative calculations and, hence, do not need to be considered in the generic post-closure safety assessment.

Figure 4.7 Idealised plan view of the GDF layout used in the generic PCSA calculations

There are separate disposal areas for unshielded ILW (UILW) and shielded ILW (SILW)/LLW/DNLEU (left-hand side), and disposal tunnels for HLW/SF/Pu/HEU (right-hand side). Layout has been developed for disposal of the derived inventory reference case in a higher strength host rock.



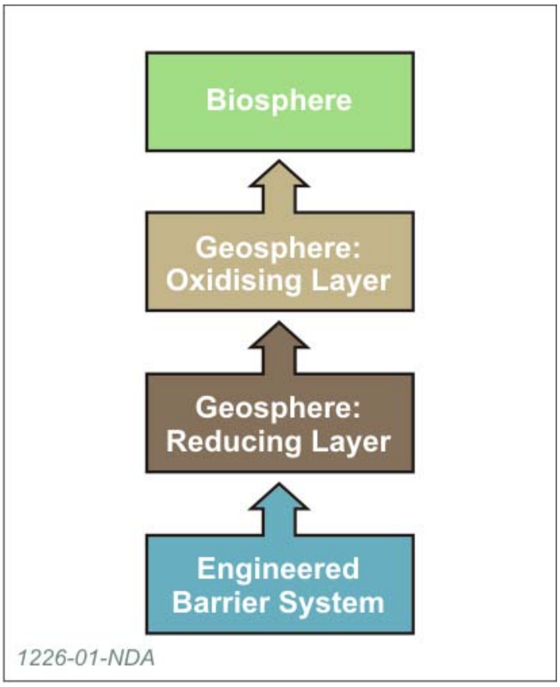
As discussed in the generic PCSA [27] and shown in Figure 4.8, our total system assessment model consists of a simple compartment model accounting for:

- the engineered system, consisting of the disposal areas and their contents and the sealed access tunnels and shafts;
- the geosphere, consisting of the rocks in which a GDF is constructed and those that surround them, extending to the surface, implemented as a multi-layer model; and
- the biosphere, consisting of the environment accessible to humans, including the soil and surface rocks, surface water bodies, oceans and the atmosphere.

We represent this system in terms of the radionuclide species, the contaminant sources, the transport media, and the transport pathways, and solve the underlying system of equations.

We assume that interactions between the ILW/LLW/DNLEU and the HLW/SF/Pu/HEU disposal areas are negligible in terms of their impact on post-closure safety, but recognise that justifying this would be an important aspect of the ESC for such a facility. The work we have undertaken to justify this assumption is presented in [71, 72], but we intend to undertake further work on this topic once we have specific candidate sites to consider. We also note a number of overseas waste management organisations have undertaken significant work on this issue to justify the development of a single GDF for a range of higher activity radioactive wastes (e.g. Andra, France [142]).

Figure 4.8 Schematic illustration of the conceptual model forming the basis of the PCSA calculations



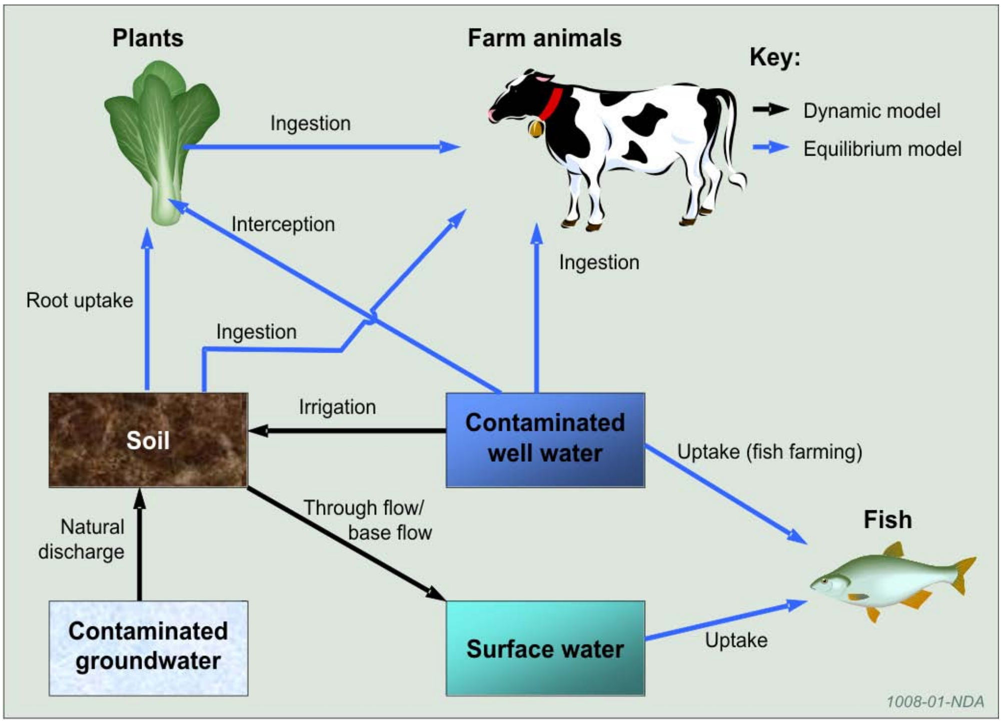
The geosphere is represented in a simplified fashion appropriate for this generic stage of the project. It consists of two layers: a reducing host rock environment into which radionuclides are released from the disposal areas, and an overlying layer with a geological environment that is oxidising. We consider this model to be suitably generic. The mineralogy and geochemical properties of the two layers are assumed to be different and we assign each layer different sorption values appropriate to each of the reducing and oxidising environments [27]. The UK Government's 2008 MRWS White Paper [4] mentions a minimum depth for the disposal areas of 200 metres. One of the considerations in choosing the actual depth of a disposal at a particular site would be to ensure that the disposal areas were sufficiently below any near-surface oxidising zone.

We also note that the value of T (the groundwater travel time from the disposal areas to the surface environment) at a real site would depend on the properties of both the host rock and any overlying sequences of rock. For example, for a higher strength host rock, a geological environment having a sedimentary layer overlying the host rock would be expected to have longer travel times than if the host rock extended all the way to the surface. This also provides a reason for developing a model for the geosphere that contains at least two components that can be assigned separate hydrogeological properties.

For this generic ESC, a set of biosphere dose conversion factors (measured in Sieverts per Becquerel) is used for the illustrative calculations, converting unit input of radioactivity into the biosphere into an annual radiological dose (measured in Sieverts per year) for a representative individual of a potentially exposed group. The basis for these is set out in the Biosphere status report [18]. These dose conversion factors have been derived from a separate generic and simplified reference biosphere assessment model suitable for the UK. The transport processes considered in this model are illustrated in Figure 4.9. Separate calculations are made to determine dose conversion factors for abstraction of contaminated water from a well and for natural groundwater discharge of contaminated groundwater.

Figure 4.9 Contaminant transport processes considered for groundwater-mediated releases to a temperate terrestrial biosphere

The figure indicates whether the underlying model for each process is time-dependent (dynamic), or whether equilibrium assumptions are adopted. Note that the soil zone consists of two layers, a surface-soil layer and a subsoil layer having different properties.



We have determined that the GoldSim software package [126, 127] is fit for the purpose of carrying out calculations of the potential radiological impacts associated with groundwater-mediated releases from a GDF and illustrating barrier performance. GoldSim provides functionality for representing contaminant and radionuclide species, transport media, transport routes, contaminant sources, and receptors, and the coupled sets of differential equations underlying these systems. Its functionality is quality assured by the company that develops and sells it, and it is used extensively by other waste management organisations carrying out similar analyses to us [e.g. 64, 143]. However, to implement our conceptual and mathematical models, it is necessary for us to link together the specialised elements provided by GoldSim.

We expect to be able to make use of similar models and computational tools for hazardous waste impact assessments as we use for radiological impact assessments.

In Section 5.2.2.1, we summarise the results of a series of probabilistic calculations of risk from the groundwater pathway for sets of values of q, T, F, A and C.

4.3.2.2 Consequences of gas

Our generic PCSA [27] also considers gas generation mechanisms – corrosion of waste materials, microbial action, ingrowth of gaseous radionuclides, 'radon-stripping' – and releases from a GDF of the gases generated. Our understanding of the generation and migration of gas under GDF conditions is summarised in the Gas status report [19]. We currently use a computer code called SMOGG [144] to estimate the rate of gas generation. SMOGG calculates the rate of gas generation from corrosion, microbial degradation and radiolysis, including the release of gaseous carbon-14 from graphite and irradiated metals. Gas migration is a highly site-specific process, and gas-mediated releases could, in theory, dominate the assessed post-closure risk for some geological disposal systems. In the generic PCSA [27] we use our knowledge of gas generation and transport processes and the results of previous assessments to explain the role of different barriers and provide a qualitative discussion of the possible consequences of gas for different types of geological disposal system. This work is summarised in Section 5.2.2.2.

Gas pressurisation and two-phase gas/water flow are likely to affect the engineered and natural barriers. However, any such effects would be highly specific to the GDF design and geological setting and have therefore not been considered in the generic PCSA.

4.3.2.3 Human intrusion

We use a computer code called SHIM [145, 146] to estimate the consequences of human intrusion. SHIM calculates the doses that result from activities such as drilling into a GDF and examining the cores recovered and occupation of a GDF site following some type of disruption that exposes waste material at the surface. SHIM can model both direct intrusions into a GDF and intrusions into a plume of radionuclides that may have migrated away from a GDF, although it does not simulate radionuclide transport in groundwater. It does not consider the impact of the intrusion event on the subsequent functioning of the EBS and geosphere barrier.

Our generic PCSA [27] includes some illustrative calculations carried out using SHIM. This work is summarised in Section 5.2.2.3.

4.3.2.4 Criticality safety

The inventory for disposal in a GDF contains radionuclides that are fissile. The potential for, and possible implications of, a criticality event in a GDF are discussed in the Criticality safety status report [21]. Waste conditioning and waste packaging arrangements are used to ensure criticality safety during the operational period. However, during the post-closure period we cannot totally rule out the possibility that fissile material could accumulate to the extent that a criticality event could occur. We assess the possible consequences of criticality events in the post-closure period using a suite of models and computer codes (QSS, RTM, FETCH) that consider the neutronics and thermal-hydraulics of a criticality event [147].

Our generic PCSA [27] includes arguments to explain why we believe a post-closure criticality event to be unlikely, and, if such an event did occur, why the consequences would not be significant in terms of environmental safety. These arguments are summarised in Section 5.2.2.4.

5 Environmental safety analysis

GRA Requirement R5: Dose constraints during the period of authorisation

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

GRA Requirement R6: Risk guidance level after the period of authorisation

After the period of authorisation, the assessed radiological risk from a disposal facility to a person representative of those at greatest risk should be consistent with a risk guidance level of 10^{-6} per year (i.e. 1 in a million per year).

GRA Requirement R7: Human intrusion after the period of authorisation

The developer/operator of a geological disposal facility should assume that human intrusion after the period of authorisation is highly unlikely to occur. The developer/operator should consider and implement any practical measures that might reduce this likelihood still further. The developer/operator should also assess the potential consequences of human intrusion after the period of authorisation.

GRA Requirement R9: Environmental radioactivity

The developer/operator should carry out an assessment to investigate the radiological effects of a disposal facility on the accessible environment, both during the period of authorisation and afterwards, with a view to showing that all aspects of the accessible environment are adequately protected.

GRA Requirement R10: Protection against non-radiological hazards

The developer/operator of a disposal facility for solid radioactive waste should demonstrate that the disposal system provides adequate protection against non-radiological hazards.

Our approach to meeting GRA Requirements R5, R6, R7, R9 and R10 is discussed in Section 3.2. In this section we develop and describe the different lines of reasoning and environmental safety arguments for:

- operational environmental safety (GRA Requirements R5, R9 and R10), which is assumed to be largely independent of the geological environment (Section 5.1); and
- post-closure safety (GRA Requirements R6, R7, R9 and R10), which is considered for a range of generic geological environments and disposal concepts (Section 5.2).

Section 5.1 presents the high-level qualitative safety arguments and quantitative assessment for operational environmental safety. In Section 5.1.1 we describe supporting evidence for environmental safety, including:

- the overall approach to achieving safety through design and management;
- how that approach is applied to provide specific features in the design to provide protection to the public for operational activities which could give rise to off-site releases and public exposures to radiation; and
- experience of similar activities at existing facilities which gives confidence that safety can be achieved.

In Section 5.1.2 we provide an example of the results from the OESA, focusing on potential aerial discharges from waste emplaced in the underground facilities of a GDF. Discharges from the surface facilities are not considered quantitatively in this generic ESC, but are discussed qualitatively in Section 5.1.1. The quantitative results are illustrative only, and are discussed in relation to a generic surface environment and disposal concept.

Section 5.2 presents the high-level qualitative safety arguments and quantitative assessment for post-closure safety. In Section 5.2.1 we describe supporting evidence for safety, including:

- the overall approach to achieving safety through design of a GDF using multiple barriers;
- the safety functions typically provided by each of the multiple barriers; and
- how we use different types of evidence to build confidence in the performance of the multiple barriers.

In Section 5.2.2 we provide an example of the results of a suite of probabilistic calculations of the potential radiological impacts associated with groundwater-mediated releases from a GDF. The results are illustrative only, and are discussed in relation to the UK application of the illustrative geological disposal concept examples. The consequences of gas generation and release from a GDF are also discussed, as are the issues of inadvertent future human intrusion and criticality safety.

Section 5.3 identifies key uncertainties in the ESC at this generic stage in the MRWS Site Selection Process.

Appendices A, B and C contain more detailed qualitative safety arguments pertaining to the UK application of each of the illustrative geological disposal concept examples considered in the generic ESC.

Our aim in presenting illustrative calculations is to provide an indication of the kind of results we would expect at Stage 4 of the MRWS Site Selection Process, and to support assessment of future waste packaging proposals under the disposability assessment process (see Section 3.1.4). For an actual site and facility design, the safety analysis would include site-specific qualitative and quantitative safety arguments, based on our understanding of the conditions at the site, its expected evolution, and the safety functions associated with the different barriers in the facility design and how they operate over different periods.

5.1 Operational environmental safety assessment

5.1.1 OESA: qualitative safety arguments – supporting evidence

5.1.1.1 Approach

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 [23]. As discussed in Section 3.3.1, 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 wastes will be packaged before transport to a GDF in accordance with detailed waste package specifications [e.g. 78, 79]. Strict quality assurance requirements are imposed to ensure these packaging specifications are met.

The packaged waste has a number of inherent safety features [15]:

- 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.[18] The wastes do not contain liquids or pressurised gases. Any wastes that are expected to evolve significant gas (e.g. Magnox) 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.[19]

Appropriate limits on the content of fissile materials established through our disposability assessment process ensure that waste packages contribute to criticality safety.

The generic OESA [26] considers the following factors for normal operating conditions:

- waste disposal activities;
- environmental hazards from discharges; and
- assessment of impacts to the public and the environment (non-human biota).

5.1.1.2 Potential for off-site release and public exposure

The activities that could give rise to public exposure to radiation and environmental releases of radionuclides during GDF operation, and the safety features that provide protection, are outlined below. We do this by describing, step by step, the journey the waste would take from receipt at the surface facilities of a GDF to its disposal in the underground facilities.

Receipt and buffer storage of radioactive waste at a GDF surface facilities

Rail wagons would be transferred to on-site sidings and road vehicles would be parked in a designated area. External doses to the public would be reduced by providing shielding, for example in the form of embankments built from excavated rock spoil, and locating these areas away from the boundary fence. Information on the safety of waste transport to a GDF is provided in the TSC [5], and information on buffer storage and waste handling in the surface facilities of a GDF is provided in the OSC [6]:

- 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 IAEA transport regulations [61]. 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.

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[18] 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.

[19] 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 [5] and OSC [6].

- Less hazardous ILW/LLW will be packaged mainly in “Type 2 Industrial Packages”, again as defined in the IAEA transport regulations [61]. 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 – these will be monitored.

Checking and unloading of waste at the surface facilities
Loading onto on-site wagons for transfer underground

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.

Transfer underground

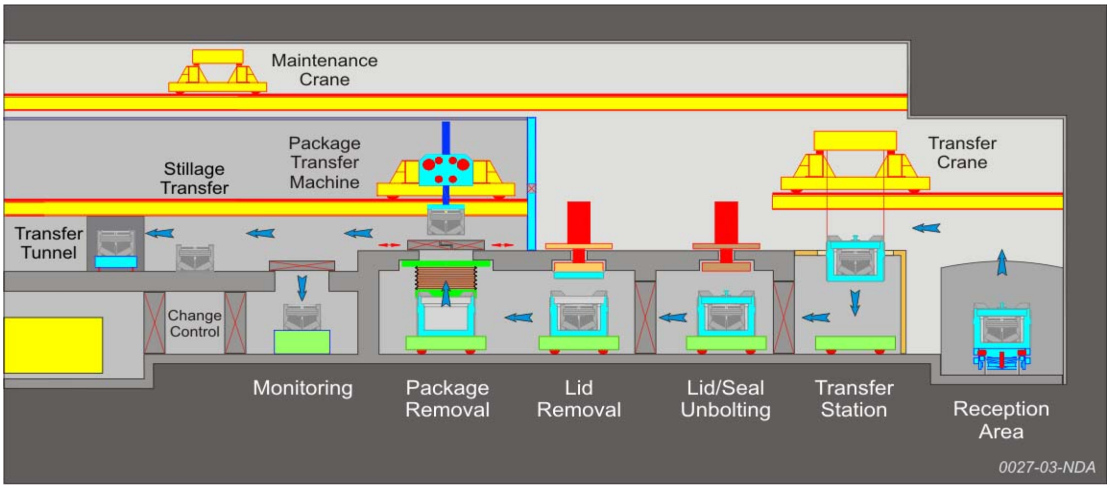
Waste would be transported underground using an inclined access tunnel or, possibly, a shaft. The Type B reusable transport containers and the Type 2 Industrial Packages would continue to contain the wastes during transfer from the surface facilities to the underground facilities. The only potential 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.

Unloading of unshielded waste packages from reusable shielded transport containers and emplacement in disposal areas via remote handling.
Direct emplacement of Type 2 Industrial Packages in disposal areas

The unloading of unshielded waste packages from reusable transport containers would be carried out underground in cells equipped with high-integrity lifting equipment. A simplified cross-section through a conceptual loading cell for unshielded ILW packages is shown in Figure 5.1.

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 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.

Figure 5.1 A simplified cross-section though a loading cell for unshielded ILW packages



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 - as illustrated conceptually in Figure 5.2 - and with less potential for release of radioactive particulates.

Figure 5.2 Emplacement of shielded ILW packages in evaporite

Maintenance of empty reusable shielded transport containers and transport vehicles

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.

Maintenance of disposal areas once they are filled with waste

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 ESC focuses on the potential for – and environmental safety implications of – discharges of radioactive gases from the underground facilities.

5.1.1.3 Operational experience

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 (see Figure 5.3).

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.

Figure 5.3 Use of half-height International Standards Organisation freight containers at the UK national LLW disposal facility near the village of Drigg in West Cumbria

Photo courtesy of LLWR Ltd.

5.1.2 OESA: illustrative quantitative results

In this section we summarise the results of our illustrative quantitative assessment calculations for gas-phase discharges of radionuclides from disposed waste during the operational phase of a GDF, based on work reported in our generic OESA [26]. We have undertaken numerical modelling to quantify the off-site impact of potential gaseous discharges from waste emplaced in an operational GDF followed by its dispersion and subsequent uptake by the public and the environment (including non-human biota). The waste-derived radioactive gases of interest are tritium, carbon-14 bearing gases and radon-222.

The Environment Agency methodology [97, 98] (as briefly discussed in Section 3.1.1.4) calculates the dose per unit discharge (DPUR) for seven different groups of the public (based on their location and habits) and to four age groups (including the foetus) [148]. The exposure group most at risk for members of the public located off-site is likely to be the local resident family. The DPUR for the most limiting age group (that is, the age group who would receive the largest dose) within this exposure group for each of the gas-phase radionuclides of concern has been calculated, accounting for three key exposure pathways: external irradiation, inhalation and consumption through food.

The RWMD Radiological Protection Policy Manual (RPPM) [73] sets out the radiological protection policy and criteria within and against which all our work 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, including the GRA [1]. 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 the facilities, will meet all current statutory radiological protection requirements.

For the OESA, the relevant criteria against which the assessment results will be compared are presented in Table 5.1; see the OESA [26] for further details of the source of these values.

Table 5.1 Dose limits and targets for normal operation to be applied during the period of licensing and/or environmental permitting

	Dose (millisieverts per year)
Effective dose limit	1
Maximum value of dose constraint to members of the public from a new facility	0.15
Design target	0.01

Key modelling assumptions in calculating off-sites doses from gaseous discharges relate to the height of release of gaseous discharges from a GDF (in the OESA this is assumed to be at a stack height of 15 metres), the distance to the off-site local resident family from the point of release (in the OESA this is assumed to be 100 metres), and the distance to where the local resident family's food source is produced (in the OESA this is assumed to be 500 metres from the release point).

Doses to the off-site public relating to gaseous discharges from an operational GDF have been calculated using the gas generation data in the Gas status report [19], as presented in Table 5.2 below ("Discharge Bq per year" data). We assume that tritium and carbon-14 bearing gases are released from a container at their generation rate. However, for radon-222 (which has a half-life of 3.82 days), we use an 'emanation coefficient' to account for retention of the gas within the waste package. The emanation coefficient corresponds to the fraction of radon-222 that is released from a waste package in comparison with the in-package radon-222 generation rate.

Table 5.2 presents calculated doses to members of the public in the local resident family receptor group located at 100 metres from a GDF and consuming food grown at 500 metres from a GDF, for a stack height of 15 metres, using average gas release rates for the operational period.

Table 5.2 Calculated doses from off site discharge to the local resident family receptor group, using average gas release rates for the operational period

Radionuclide	Discharge (becquerels per year)	Food consumption DPUR	External irradiation DPUR	Inhalation DPUR	Dose (milli-sieverts per year)
Tritium	1×10^{12}	5.7×10^{-13}	–	1.4×10^{-12}	0.0004
Radon-222	2×10^{12}	–	4.5×10^{-16}	2.4×10^{-10}	0.043
Carbon-14	5×10^{11}	3.3×10^{-11}	4.5×10^{-16}	2.4×10^{-10}	0.009
Total					0.052

The total dose to members of the public from the operational period average gas generation rates is calculated to be 0.052 millisieverts per year. The majority of this dose arises from radon-222 (0.043 millisieverts per year), 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 millisievert 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 millisieverts per year, and below the maximum value of dose constraint to members of the public from a new facility of 0.15 millisieverts per year that we have adopted in our RPPM [73]. However, although the dose from carbon-14 and tritium is below the design target of 0.01 millisieverts per year, the dose from radon-222 exceeds the design target.

Backfilling of the GDF 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 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, reported in the OESA, 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 millisieverts per year. The majority of this dose (0.11 millisieverts per year) 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 millisievert per year and 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 millisieverts 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 millisieverts 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 millisieverts per year that we have adopted in our RPPM.

Radon-222 doses presented in the OESA assume an emanation coefficient of 2×10^{-3}. Work undertaken to date as part of the LoC disposability assessment process has established that this emanation coefficient can be reduced using a multi-barrier approach to waste packaging. 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 millisieverts per year can readily be achieved by application of appropriate packaging measures.

We also report studies in the OESA 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 is also reported in the OESA. The results of this assessment indicate that the assessment of dose to non-human biota from off-site gaseous radioactive discharges from a GDF requires 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 [19]. Mitigation measures could be taken to reduce the potential for gaseous discharges from a GDF that would, consequently, also reduce any off-site doses.

In addition to the emplaced radioactive inventory, the host geology may also be a source of naturally occurring radon. We note the need to assess the implications of this naturally-occurring radon at a site-specific level.

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.

5.2 Post-closure safety assessment

Post-closure safety relies on the intrinsic passive safety of an appropriately sited and designed GDF. A post-closure safety assessment needs to explain why we can have confidence that the barriers and safety functions of a GDF will provide the required level of safety. This confidence is achieved by demonstration of a detailed understanding of the way the various barriers work together to isolate and contain the wastes as they evolve over time. Based on this understanding we can present qualitative safety arguments (see Section 5.2.1 below).

A post-closure safety assessment also needs to present a quantitative analysis of GDF performance over long timescales. For this quantitative analysis it is appropriate to focus on the mechanisms by which radionuclides from the wastes could be returned to the biosphere and their impact on people and the environment. At this generic stage, in the absence of knowledge of the site or design for a GDF, we present the results of illustrative calculations to understand the potential environmental impact of a GDF and to underpin our disposability assessment process (see Section 5.2.2).

5.2.1 PCSA: qualitative safety arguments – supporting evidence

5.2.1.1 Approach

Some of the wastes under consideration will remain hazardous for extremely long times. The function of a geological disposal system is to protect humans and the environment from the harmful effects of the wastes over the required timescales. This is achieved through a system of multiple barriers that results in the almost complete decay of many radionuclides before they can enter the biosphere, and sufficient containment and retardation of any remaining, undecayed radionuclides so that they enter the biosphere only at very low concentrations.

Our strategy for implementing geological disposal is based on isolation and containment of the disposed waste. Isolation and containment are provided by a series of barriers whose functions and relative importance may vary at different times during the evolution of a GDF, as set out in Section 3.1.1.3. As described in the research status reports, we have a mature understanding of the majority of the processes we will need to consider when designing the EBS to complement the wasteforms and geological environment. This experience is relevant to a wide range of geological environments, providing a sound basis from which to move forward.

The DSTS [23] sets out general requirements for the **safety functions** of a GDF. The generic post-closure safety functions for the principal barriers within a GDF are summarised by barrier in Table 5.3 and discussed in more detail in Section 5.2.1.2.

The specific safety functions relevant to the illustrative geological disposal concept examples considered in our post-closure safety assessment are described in Appendices A, B and C, which draw together material from the research status reports. These appendices provide an indication of how the generic safety functions presented in Table 5.3 can be tailored to particular geological disposal concepts.

Table 5.3 Generic post-closure safety functions of the principal components of multi-barrier systems

Note that each barrier will not necessarily provide all the post-closure safety functions listed, and the importance of particular safety functions will depend on the geological disposal concept, which itself is linked to the geological environment that is available and to the waste types in question. In addition, the importance of safety functions will vary in time, as discussed in Section 3.1.1.3. The overall post-closure safety of a geological disposal system will not depend on any one of these safety functions alone, but on how the functions interact with each other over time, as the closed GDF evolves.

Barrier	Generic post-closure safety function
Wasteform	• provide a stable, low-solubility matrix that limits the rate of release of the majority of radionuclides by dissolution in groundwater that comes into contact with the wasteform
Waste container	• protect the wasteform from physical disruption (e.g. by movement in the bedrock) • prevent groundwater from reaching the wasteform • after corrosion has breached the container, act as a partial barrier limiting the movement of groundwater in and around the wasteform • control the redox conditions in the vicinity of the wasteform by corrosion reactions, thus controlling the solubility of some radionuclides • allow the passage of gas from the wasteform into the surrounding EBS
Buffer or backfill around the waste container	• protect the waste container from physical disruption (e.g. by movement in the bedrock) • control the rate at which groundwater can move to and around the waste container (e.g. by preventing flow) • control the rate at which chemical corrodents in groundwater can move to the waste container • condition the chemical characteristics of groundwater and porewater in contact with the container and the wasteform so as to reduce corrosion rate and/or solubility of radionuclides • control the rate at which dissolved radionuclides can move from the wasteform into the surrounding rock • control or prevent the movement of radionuclide-containing colloids from the wasteform into the rock • suppress microbial activity in the vicinity of the waste • permit the passage of gas from the waste and the corroding container into the host rock
Mass backfill	• restore mechanical continuity and stability to the rock and engineered region of a GDF so that the other engineered barriers are not physically disrupted (e.g. as a clay buffer takes up water and expands) • close voids that could otherwise act as groundwater flow pathways within a GDF
Seals	• cut off potential fast groundwater flow pathways within a backfilled GDF (e.g. at the interface between mass backfill and host rock) • prevent access of people into a closed GDF

Barrier	Generic post-closure safety function
Geological barrier	• isolate waste from people and the surface environment by providing a massive radiation shield • protect the EBS from human and natural events and processes occurring at the surface and in the near-surface rocks (e.g. major changes in climate, such as glaciation) • protect the EBS by providing a stable mechanical and chemical environment at depth that does not change quickly with the passage of time and can thus be forecast with confidence • provide rock properties and a weakly active hydrogeological environment that limit the rate at which deep groundwater can move to, through and from a backfilled and sealed GDF, or that completely prevent flow • ensure that chemical, mechanical and hydrogeological evolution of geological environment is slow and can be forecast with confidence • provide properties that retard the movement of any radionuclides in groundwater – these include sorption onto mineral surfaces and properties that promote dispersion and dilution of radionuclide concentrations • allow the conduction of heat generated by the waste away from the EBS so as to prevent unacceptable temperature rises • disperse gases produced in a GDF so as to prevent mechanical disruption of the EBS

5.2.1.2 Post-closure safety functions and safety arguments

This Section discusses the high-level safety functions of isolation and containment, and describes how they are provided by the various barriers within a GDF (expanding on Table 5.3). Where appropriate, we also indicate how the safety functions are represented in the illustrative generic PCSA calculations that we present in Section 5.2.2.

As already noted, for post-closure safety assessment calculations we need to consider the mechanisms by which radionuclides from the wastes could be released through the barriers and return to the biosphere. We have identified four ways by which radionuclides (and other contaminants) in the waste could, in principle, return to the biosphere in the post-closure period:

- transport of radionuclides by groundwater from a GDF through the overlying rocks to the surface environment where humans could be exposed to external radiation or ingest radionuclides in food or water;
- release of radioactive gases, owing to waste degradation processes such as corrosion and microbial activity, and their subsequent migration, as free gas or dissolved in water, to the surface environment where humans could be exposed to external radiation, inhalation or ingestion of radionuclides in food or water;
- release and return of radioactive material directly to the surface environment as a result of inadvertent human intrusion into the disposal facility as a result, for example, of drilling or excavation activities; and
- geological events and processes that might lead to direct exposure of wastes at the surface (e.g. through uplift and erosion, or through volcanic activity).

The discussion in this section considers how the engineered and natural barriers could operate for the high-level safety functions of isolation and containment in the context of these potential release routes. As discussed in Section 3.1.1.3, the importance of particular post-closure safety functions will depend on the geological disposal concept, which itself is linked to the geological environment that is available and to the waste types in question. In addition, the importance of safety functions will vary in time. The overall post-closure safety of a geological disposal system will not depend on any one of these safety functions alone, but on how the functions interact with each other over time, as the closed GDF evolves.

Isolation

As discussed in Section 3.1.1.3, isolation means removing the waste and its associated hazard from people and the surface environment. The underground vaults and disposal tunnels are likely to be located at a depth of between 200 and 1,000 metres in a stable rock formation in a region of naturally low groundwater flow, but even the shallowest depth would provide sufficient radiation shielding for anyone at the surface. The actual depth would depend on the geological environment at the site in question. The UK is located in a geologically stable region and does not suffer the volcanic eruptions or major earthquakes that cause significant damage in some parts of the world – the last volcanic eruptions in the UK occurred in the Hebrides over 50 million years ago [149]. The surrounding host rock and other geological formations between the disposal areas and the surface provide a natural barrier. At the end of disposal facility operations, the access routes would be backfilled and sealed to provide isolation and prevent human access. The disposal areas would then be isolated from natural disruptive events and human intrusion in a physically and chemically stable environment.

A disposal depth of at least 200 metres would provide adequate isolation from disruption from natural events (such as erosion resulting from glacial action and climate change in the very long term) while the wastes are still hazardous [139]. Such a depth would also provide protection from disruption due to terrorism or acts of war. In practice, based on precedents from other waste management programmes and past UK experience, the disposal depth is likely to be greater than 200 metres.

One potential route for the return of radioactivity to the surface environment is through direct human intrusion as a result of, for example, drilling for exploitation of natural resources such as coal, oil, gas or metal ores or drilling for geothermal heat extraction. The MRWS Site Selection Process [4] ensures that areas with exploitable resources at depth are screened out at an early stage (see Section 2.1). This would minimise the likelihood of future human intrusion at any time in the post-closure period. Furthermore, drilling records from the UK and overseas show that the number of boreholes drilled to depth reduces as the depth increases (e.g. [150]).

In our illustrative generic PCSA calculations for radionuclide transport by groundwater [27], the disposal depth influences the time it takes for groundwater that passes through the GDF to return to the surface environment (i.e., the parameter T, see Figure 4.6). The depth of a GDF also influences the likelihood of a human intrusion scenario.

Containment

As discussed in Section 3.1.1.3, the objective of containment for a GDF is to retain radionuclides within various parts of the multi-barrier system for as long as required. A number of engineered components of the geological disposal system provide containment by reducing the accessibility of the waste to groundwater and subsequent transport of radionuclides to the surface environment via groundwater flow [16]. Limiting water availability also limits gas generation in a GDF [19].

'Impermeable' barriers, such as the waste containers, any overpack and many wasteforms, provide effective containment until there is physical failure of the barrier, for example owing to corrosion processes. Diffusion is the only process that transports radionuclides through these impermeable barriers while they are intact, and this process is extremely slow for materials such as the metals used in waste containers (effectively there is no transport). A substantial proportion of the disposed activity may decay before the barrier fails; this proportion is therefore completely contained because it is never released. Once the barrier no longer provides complete containment, it may continue to offer resistance to release through various retardation processes, for example through sorption of radionuclides onto corrosion products. For low-permeability barriers, retardation processes play an important part in providing an initial period of containment.

In other engineered barriers (e.g., buffer, backfill), the combination of low permeability and retardation results in very slow migration through the barrier, effectively achieving a significant period of containment within that barrier (i.e. a significant period before there is any release). The radionuclides are being transported (slowly) through the barrier during the period of containment and, following the 'failure' of containment, the processes that result in this slow transport will continue to operate.

We discuss below the specific roles of the wasteform, waste container, buffer/backfill, mass backfill, seals and host rock in providing containment of radionuclides and indicate how these are represented in our illustrative generic PCSA calculations. Further information on the specific application of the safety functions discussed below to the illustrative geological disposal concept examples is provided in Appendices A, B and C, our research status reports [15, 16, 20], and our geological disposal concept option reports [28, 29, 30].

Wasteform

The principal objective of all wasteforms is to provide a stable, low-solubility matrix that limits the rate of release of the majority of radionuclides by dissolution in groundwater that comes into contact with the wasteform. For some wastes, for example HLW and SF, the wasteform is essentially fixed and the characteristics that provide these safety functions are readily identifiable, whereas for other types of waste, such as ILW, there is scope for tailoring the characteristics of the wasteform when the waste is conditioned and packaged. This is an important consideration during the post-closure assessment of waste packaging proposals during our disposability assessment process [14].

Most of the wasteforms disposed of in a GDF would be inherently durable under conditions in the disposal areas (see the Package evolution status report [15]) and, thus, would contain the radionuclides for a period even once the waste container no longer provided complete containment, as discussed below:

- Some of the ILW and LLW has already been packaged in a way that is compliant with our disposability assessment process. Most of the ILW that has been packaged has been conditioned using a cementitious grout. Studies of cements [151] have shown that they retain their physical strength for long periods.
- A significant quantity of the HLW vitrified wasteform has already been produced. HLW glass has been shown to be a durable wasteform [152]. The radionuclides in the wasteform are bound in a form that, once the container fails, dissolves only slowly in the chemical environment of a GDF. Radionuclides can only be released as the glass dissolves – this is represented by the glass dissolution rate parameter in our generic PCSA calculations [27].
- SF comprises uranium oxide in which fission products are distributed in the crystalline matrix and, after the container fails, radionuclides can only be released as the uranium oxide is slowly dissolved in groundwater. However, some radionuclides are present on the surface of the fuel and are more readily dissolved (this is termed the 'instant release fraction'). The generic PCSA calculations represent the instant release fraction and then apply a dissolution rate for SF to represent the slow release of the remaining radionuclides [27].

Sorption of radionuclides on corrosion/wasteform-alteration products in containers could further retain the radionuclide within the waste package. Sorption acts to temporarily remove radionuclides from solution.

Waste container

Prior to disposal, the post-closure performance of all waste containers will have been considered as part of our disposability assessment process [14]. One of the aims of this process is to ensure that all waste packages (that is the waste container and its wasteform content) are consistent with the required safety functions of a GDF.

In the post-closure period, the primary safety function of the waste container, including the overpack if present, is to provide physical containment of the wastes. HLW and SF containers are designed to contain the waste and its inventory by preventing groundwater from reaching the wasteform for a long period; physical isolation of HLW and SF from near-field porewater and containment of the radionuclides within the waste packages for thousands of years is a key safety argument for these concepts.

In contrast, many ILW containers are vented because the wastes have the potential to generate gas, and it is important to avoid pressure build-up within the container. Such containers are designed to allow the passage of gas from the waste package into the surrounding environment. These waste packages resaturate on a timescale that is controlled by the properties of the vent and wasteform, and the permeability of the surrounding engineered barriers and host rock. However, the vents are relatively small and, while the overall container is intact, it is unlikely that a direct flow path through the waste package would be established. As a result, there would effectively be no water flow through the package. Solute transport would be by diffusion and the radionuclides would be effectively contained within the waste package until the waste container fails, for example through corrosion. Long-term corrosion studies in chemical conditions representative of a GDF show that the rate of uniform corrosion of the materials typically used for ILW containers is very low [e.g.153].

As waste containers corrode, they may establish a chemical environment in the near-field porewater that limits radionuclide mobility (e.g. by the creation of chemically reducing conditions that limit the solubility of some radionuclides). Additional safety functions of the waste container are to protect the wasteform from physical disruption and, after the container has been breached, to act as a partial barrier limiting the movement of groundwater in and around the wasteform.

Finally, waste packages are unlikely to all degrade at the same rate, because of variations in local conditions in a GDF and in their manufacture. Although this is not a design objective and, therefore, not a formal safety function, any spread of package degradation rate would spread the release of the inventory (i.e. there would be a spread of times at which containment is lost) and, thereby, dilute the flux of radionuclides that becomes available for transport by groundwater.

In the generic PCSA calculations [27], the containment parameter, C, represents the period for which the waste container is intact. For vented ILW containers, this parameter is conservatively set to a value of zero years.

Buffer/backfill

EBS designs generally include additional physical barriers (i.e. low-permeability materials) emplaced adjacent to the waste package, such as bentonite (whose clay minerals swell on contact with water) or crushed salt (which is effectively impermeable once creep processes have sealed any void space). These materials contribute to containment provided by the waste package by limiting the rate at which groundwater, reactants and reaction products can be transported to and from the waste package. For example, a low-permeability material may be able to exclude potentially detrimental species such as microbes and colloids that could promote corrosion of the waste package or enhance contaminant transport. A low-permeability barrier may also provide containment in its own right because of the substantial length of time required for the radionuclides to migrate through it by diffusion.

Chemical buffering within cementitious buffers/backfills (and within cement-grouted ILW containers) is an important safety function for cementitious disposal concepts. Chemical buffering can provide a favourable chemical environment limiting corrosion of the waste container. In addition, dissolution of minerals in the cement controls the solubility of key radionuclides in the waste in contact with the cement-conditioned porewater by providing a chemical environment that reduces their solubility. The solubility limit represents the maximum possible concentration that may be attained in solution. However, it is unlikely that the solubility limit will be reached if the rate at which radionuclides are leached from the wasteform is slow.

Radionuclides are strongly retained or retarded on clay and cementitious barriers. Sorption is the key mechanism by which this occurs, although precipitation may be an additional mechanism, particularly near the excavation-disturbed zone for disposal concepts relying on a cementitious buffer/backfill.

The buffer/backfill also serves to retain the waste package in its required position, and protect the waste container from physical disruption. Where gas generation is potentially important (e.g. ILW), the backfill can be engineered to provide sufficient porosity to allow the passage of gas from the waste package into the host rock, to avoid overpressurisation within the GDF. Thermal properties of the buffer/backfill may help to dissipate heat from heat-generating wastes.

The generic PCSA calculations represent the solubility and sorption properties of all relevant radionuclides in the near field [27]. The parameters for solubility and sorption take into account the chemical buffering provided by the backfill. The porosity and density of the backfill are also represented.

Mass backfill

Mass backfill refers to material that is placed in the excavated openings when part or all of a GDF is being closed after waste and any buffer material has been emplaced. The range of characteristics of mass backfill may be similar to buffer/backfill placed immediately around the waste packages. Mass backfill can restore mechanical continuity and stability to the rock and engineered region of a GDF so that the other engineered barriers are not physically disrupted (e.g. as a clay buffer takes up water and expands), close voids that could otherwise act as groundwater flow pathways within a GDF, and prevent easy access of people to the waste packages.

The generic PCSA calculations [27] assume that a GDF has been backfilled effectively and that there are no voids around the waste packages.

Seals

Seals support the isolation function of the geological barrier, and can provide long-lasting containment [16]. They would be designed to be compatible with the host rock and to achieve a low permeability similar to the host rock, in order to stop access ways from becoming preferential pathways for the transport of radionuclides. The seals are likely to use cement-based concretes and natural materials such as swelling clays to achieve their containment function. The spaces between seals would probably be filled with crushed rock.

Our generic PCSA calculations [27] assume that a GDF has been sealed effectively so there are no preferential pathways for the transport of radionuclides. In more detailed site-specific studies, we envisage undertaking variant calculations to explore the impact of seal performance on overall performance of the GDF.

Geological barrier

As outlined previously, the rock overlying the disposal areas would serve to isolate the waste from the biosphere by providing a massive radiation shield. The geological barrier also protects the EBS from any human and natural processes and events occurring at the surface and in the near-surface rocks (e.g. major changes in climate, such as glaciation).

The mechanical stability of the host rock helps to protect the EBS. The mechanical strength of the host rock is also important from the perspective of ease of facility construction and GDF layout.

The host rock can contribute to the containment function of the EBS by providing a stable chemical environment and limiting the rate of water flow through the EBS. The rate of groundwater flow through the EBS is represented by the specific discharge parameter, q, in the generic PCSA calculations [27]. The majority of the degradation reactions that affect the engineered barriers are water mediated, so limiting the volume of water flowing through the near field limits both the rate of barrier degradation and the rate at which dissolved contaminants can be transported away from the waste packages. However, it is important to note that some buffer and backfill barriers must be fully resaturated before they can function properly, and a lower than expected resaturation rate (i.e. unexpectedly low groundwater flow) may prevent the barrier functions from becoming established on the expected timescales.

A relatively high thermal conductivity in the host rock would promote the conduction of heat generated by the waste away from the EBS, and could allow a higher emplacement density of heat-generating wastes (HLW/SF) in a GDF while preventing any unacceptable temperature rises.

Geological containment results from the length of time required for the radionuclides to traverse the geological barrier, and the decay of radionuclides released from the near field within the geological barrier. In the generic PCSA calculations [27], the length of time taken for a radionuclide to traverse the geological barrier is represented by the groundwater travel time, T, multiplied by a radionuclide-dependent sorption retardation factor.

For long-lived radionuclides that are able to traverse the geological barrier, the concentration reaching the accessible environment is further reduced by processes such as:

- the spreading out or dispersion of the radionuclides as the water migrates from depth; and
- dilution of water carrying GDF-derived dissolved radionuclides by mixing with uncontaminated water in the overlying rocks.

In the groundwater flow model implemented in our generic PCSA calculations [27], these processes are represented by the parameters F, groundwater mixing flux, and A, discharge area.

Different geological environments can achieve containment in different ways. For example, all three of the illustrative geological environments considered in this generic ESC can have low permeabilities, but in some evaporites the permeability may be so low as to be unmeasurable. The migration of radionuclides through lower strength sedimentary host rocks and evaporite host rocks is expected to be controlled by diffusion, which is a much slower process than advection. Low hydraulic gradients, low water content in the host rock, and rock matrix diffusion can also all contribute to containment of radionuclides in the geosphere.

Clay minerals occur commonly in lower strength sedimentary rocks, and are strongly sorbing towards many radionuclides. This results in radionuclides such as the actinides being strongly retarded in these rocks, and consequently increases their travel time through the geosphere. In higher strength rocks, clay minerals and iron oxyhydroxide minerals (also strongly sorbing) often occur as surface coatings on the discontinuities in the rock along which groundwater flow and radionuclide migration mainly occur. Therefore, even though these may be only trace minerals in the bulk rock, they can have significant containment functions. Radionuclide retention and retardation is a key aspect of such geological environments. In contrast, the potential for radionuclide retention by sorption or precipitation in evaporites is much lower, because of the low sorption capacity of evaporite minerals and the high salinity of any groundwater in these rocks. Evaporites act to contain radionuclides principally by isolating the waste package from mobile groundwater.

Impacts of gas on the safety functions of GDF barriers

As noted previously, gases are generated by the waste and these need to be able to escape from the disposal facility without causing a build-up of pressure that could disrupt the EBS or drive groundwater flow in the host rock. The overlying rocks may help disperse the gas so that it is released over a large area, thereby reducing the potential radiological dose from any radioactive gas component. The overlying rocks may also provide a large volume of water into which the gas can dissolve – this would reduce the quantity of gas available to move as a free gas phase, and, in some geological environments, could be enough to prevent breakthrough of a free gas phase at the surface. Alternatively, the overlying geological layers may contain 'traps' within which the gas could accumulate and be contained in the geosphere, in a similar way to gas accumulating in natural hydrocarbon traps.

Gas generation in a GDF should not pose a threat to the containment provided by impermeable barriers such as metal overpacks. Most gas generation processes require water and the amount of water within these sealed waste packages is extremely small. Even when a container has been breached and water is able to contact the wasteform, if the availability of water at the surface of the waste package is low, gas generation rates may be water limited. On the other hand, gas generation could potentially threaten the containment functions of engineered barriers that provide containment as a result of a combination of low permeability and retardation. If the gas generation rate is sufficiently high that a free gas phase forms, then the escaping gas may disrupt the barriers as well as transporting radionuclides. It is, however, possible to engineer backfill materials to have a relatively high permeability for gas and a low permeability for water, and thereby facilitate the safe release of any excess gas pressure. The EBS resaturation rate may be coupled to the gas generation rate in host rocks where gas is not able to escape easily from a GDF.

5.2.1.3 Building confidence in post-closure safety assessments

An ESC needs to be based on a variety of lines of argument, reasoning and results to evaluate the long-term safety of a GDF. In this section we discuss how we can build further confidence in the safety arguments, and provide examples of different lines of work designed to provide the required variety of environmental safety arguments. This approach is aimed at ensuring that undue reliance is not placed on any single line of argument, including the quantitative assessment calculations, in evaluating the post-closure safety of a GDF.

R&D programme and long-term demonstration experiments

We have developed an R&D strategy [86] and have an active R&D programme. Recent results from this programme are summarised in our research status reports, and the forward programme is outlined in Section 6.3.

Where appropriate, we will carry out long-term experiments, which may be planned to last for many years. In addition, we will carry out long-term demonstrations of technology and barrier performance, in which we will predict the expected evolution of a test system (for example, backfill in contact with waste materials, or the resaturation of a buffer under realistic GDF conditions) and then compare the results with the predictions, to confirm our understanding and build confidence.

Both demonstrations and experiments can be undertaken in conditions that simulate those in a GDF, for example the demonstrations and experiments currently being undertaken by other waste management organisations in underground rock laboratories overseas, such as those at Äspö in Sweden [154] or Grimsel and Mont Terri in Switzerland [155]. Some of these demonstrations and experiments may cover several years, or even tens of years. However, all these timescales are a long way short of the hundreds of thousands of years that need to be addressed within the ESC.

Natural indicators that provide confidence in safety functions

As discussed in Section 3.2.2.4, quantitative safety assessment calculations undertaken as part of the ESC are likely to be necessary for a timescale on the order of one million years. Calculated radiological doses from assessment models will be more uncertain at longer times into the future. At such times other arguments can be used to illustrate safety and build confidence in the long-term safety of a geological disposal system. For example, natural indicators may be used to provide information on the travel time through parts of the geosphere to corroborate the results obtained from numerical models (see Section 3.2.2.5).

The use of natural indicators is being progressed in several overseas radioactive waste disposal programmes, and examples of natural indicators used to strengthen a safety case are considered in [113]. Natural indicators of safety typically considered during site characterisation and the development of the post-closure safety arguments include:

- indicators for a long groundwater travel time;
- indicators for radionuclide retardation in the geosphere; and
- indicators for geochemical stability in the host rock.

The development of these indicators could draw on information from:

- palaeohydrogeology - groundwater origins, ages, past flow patterns and mixing histories derived from:
  - spatial distribution of hydraulic properties such as over-pressurisation and under-pressurisation, location of recharge and discharge areas, and hydraulic gradients;
  - spatial distribution of groundwater composition, including variations in total dissolved solids and the presence of main and trace ionic species and isotopes; and
  - rock/water interactions and their influence on groundwater composition, isotope signatures and fracture-infill minerals.
- rates of release of naturally occurring radionuclides (such as the various isotopes of uranium) from geological formations to the biosphere;
- buffer capacity, providing information on groundwater composition; and
- natural analogue studies (see below).

Additionally, radioactive and chemically toxic species are present in the natural environment. Measuring the concentrations of naturally occurring radioactive and toxic species and the fluxes of those species in groundwater in the vicinity of the candidate site(s) could provide a useful baseline against which to compare calculated values of GDF-derived fluxes or concentrations of radioactive or chemotoxic species in the post-closure period.

We will use a range of natural indicators to support the ESC. The indicators listed above will become more relevant as the ESC is developed to consider actual sites.

Comparisons with analogues

To build confidence in the understanding of the behaviour of components of a GDF over very long timescales, analogues in archaeology and nature can be sought. Nirex previously commissioned a series of reports [151, 156, 157, 158, 159, 160] summarising understanding based on extensive work on natural analogues that has been carried out worldwide over several decades. We will use analogue information to build additional confidence in the performance of the barriers and their associated safety functions. As we develop the ESC and move forward to site-specific and concept-specific assessments, the selection of analogues will be tailored to the particular disposal concepts and geological environment(s) being considered.

We provide some examples below of analogues from both archaeology and nature.

Archaeology

Many of the materials being considered for use in the engineered barriers of a GDF are similar to materials that people have been using for a long time, and it is possible to find archaeological analogues that have existed on the timescale of thousands of years.

For example, HLW is solidified into a form of glass. Studies of ancient glass artefacts [161] show that glass is stable and is slow to dissolve over long timescales (e.g. see Figure 5.4). Analogue glasses with a range of different compositions are available, including volcanic glasses which are many millions of years old. Studying the behaviour of these different glasses allows us to build confidence in the likely performance of the HLW glass.

Figure 5.4 Photograph of a 14.5-centimetre-long Egyptian fish-shaped glass vessel (3,300 – 3,400 years old)

The vessel has been perfectly preserved, even in the surface environment [162].

0997-01-British Museum

Iron is one of the materials we are considering for use as an overpack for disposal of HLW. Although iron rusts in the atmosphere, in the absence of oxygen, iron artefacts can be preserved for long times. There are many examples of well-preserved iron artefacts from the Roman era, including a large hoard of iron nails, buried by Roman soldiers in clay soil almost 2,000 years ago at Inchtuthil in Scotland (Figure 5.5), which show how large masses of iron corrode [160]. Once a GDF has been backfilled and sealed, there would be little oxygen left, so we expect iron waste containers to corrode slowly in the post-closure period.

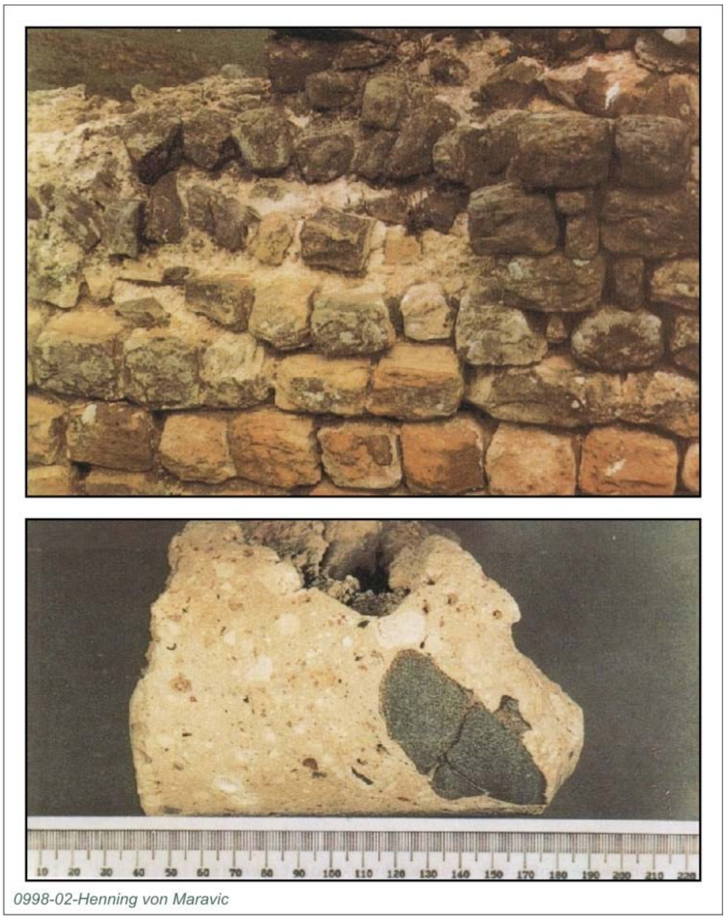
Figure 5.5 Photograph of a 30-centimetre-long Roman nail from the central part of the Inchtuthil hoard

Photograph courtesy of Bill Miller.

Most of the ILW that has been packaged has been conditioned using cementitious material, and the illustrative geological disposal concept examples include cases where cement is also used as a buffer/backfill material. There are examples of cement materials that have remained stable for thousands of years, even when wet and in an oxidising environment. For example, Hadrian's Wall (Figure 5.6) was constructed about 1,900 years ago and illustrates the longevity and stability of Roman cement, even when present in a harsh surface environment [151].

Figure 5.6 The upper photograph shows a section of Hadrian's Wall, and the lower photograph shows a detail of cement from the Wall

The cement is in good condition, despite its long exposure to the elements. The scale of the lower photograph is 22 centimetres across. Photograph courtesy of Henning von Maravic.

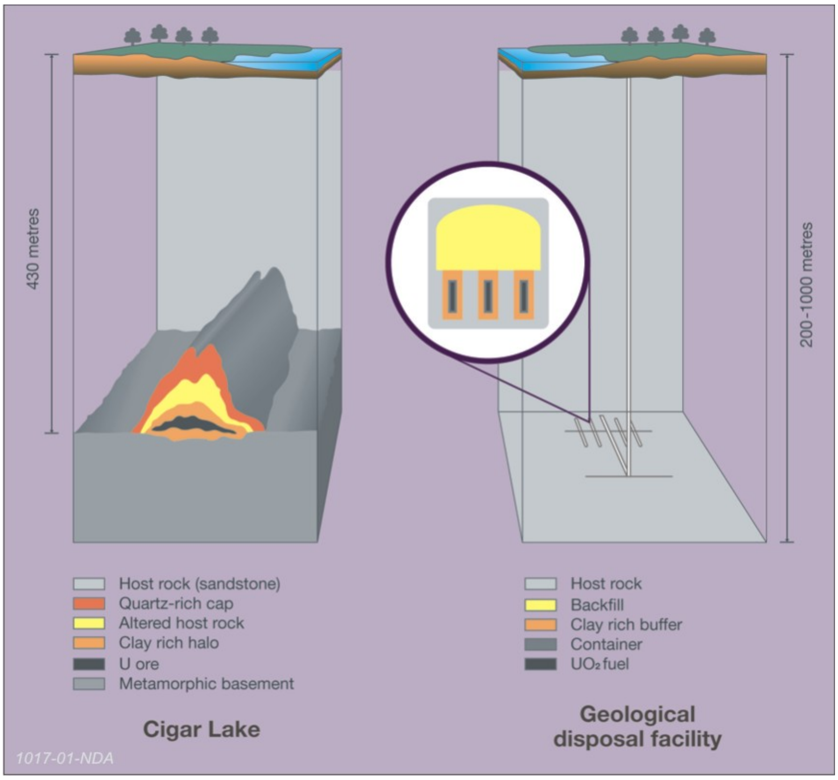


The materials used by ancient civilisations differ from those that would be used in a GDF, and the conditions under which such artefacts have been preserved also differs in detail from the likely conditions in a GDF. Although these examples can give some confidence in the longevity of the materials we are considering, we recognise that when presenting such safety arguments it is important to acknowledge their limitations. In particular, we note that what such analogues show is that artefacts can survive for long periods, not that they regularly do survive. In that sense, the information from archaeological analogues can be considered somewhat anecdotal.

Nature

As well as making comparisons with archaeological artefacts, aspects of the multi-barrier system can be compared with natural systems. For example, study of uranium-ore deposits indicates that the natural barrier can function effectively for long periods. An example of natural concentration and long-term isolation of uranium is provided by the Cigar Lake uranium deposit in northern Saskatchewan, Canada [163]. It occurs at a depth of about 430 metres, similar to the potential depth of a GDF (Figure 5.7). The Cigar Lake ore body was formed some 1,300 million years ago, yet there is no significant trace of uranium derived from the ore body at the surface today. The rocks that surround the ore have prevented the release of uranium radionuclides to the surface for a period that is more than a thousand times longer than we need to consider in the ESC. Cigar Lake has been extensively studied as a natural example of a geological disposal system limiting the movement of uranium over extremely long periods.

Figure 5.7 Illustration of Cigar Lake uranium-ore deposit, showing similarities to a geological disposal system



Rather than being a direct support to system-level modelling of risk from a GDF, analogues often support the underpinning models (e.g. models of cement dissolution). For example, during the 1990s there was a collaborative international natural analogue project [164] studying the Maqarin site in northeast Jordan. This is an unusual site because it has an active groundwater system and a naturally occurring portlandite (as might be used as backfill in a cementitious disposal facility). Water/rock interactions produce secondary phases analogous to those that may be found around a GDF. This analogue complements other modelling and experimental work undertaken [165].

5.2.2 PCSA: illustrative quantitative results

In this section we summarise the results of our illustrative quantitative assessment calculations for groundwater-mediated releases of radioactivity from a GDF in the post-closure period (Section 5.2.2.1). The details of the models and assessment methods are summarised in Section 4.3.2, and discussed in more detail in the generic PCSA [27]. We focus on potential groundwater releases because that is the most likely means by which radionuclides would be transported from a GDF back to the surface environment.

Our aim in presenting these illustrative calculations is to provide an indication of the sort of results we would expect at the desk-based stage of a site investigation programme, and to continue to provide a quantitative benchmark for assessing proposals for packaging waste and providing advice to waste producers on packaging. The calculations also illustrate the sensitivity of calculated radiological risk to the performance of the waste container and the geological barrier. As discussed in Section 3.1.4.3 and in the generic PCSA [27], given the generic stage of our programme, these calculations should be regarded as providing indicative information concerning the calculated radiological risk.

No new illustrative calculations have been conducted specifically for this generic ESC for the consequences of GDF-derived gas. Our Gas status report [19] indicates that gas-mediated release of radionuclides to the surface environment in the post-closure period may be unlikely and would be dependent on the nature of the geological environment at quite a detailed level, as summarised in Section 5.2.2.2.

Our work on considering the consequences of inadvertent human intrusion is summarised in Section 5.2.2.3.

Our work to demonstrate criticality safety of waste emplaced in a GDF is summarised in Section 5.2.2.4.

5.2.2.1 Radionuclide transport by groundwater

The generic PCSA [27] presents calculations of risk for radionuclide transport by groundwater for various combinations of the q, T, F, A parameters representing the performance of the geosphere, and C representing the containment time of radionuclides within the waste container, with the aim of illustrating the sensitivity of calculated radiological risk to these key parameters. The values of the q, T, F and A parameters depend on the nature of the regional groundwater flow at a site (as discussed in Section 4.3.2.1), and in a site-specific assessment would be determined by more detailed modelling underpinned by data from a substantial site-investigation programme.

Calculations have been carried out for the Derived Inventory reference case for ILW and LLW, HLW, SF, Pu, HEU and DNLEU. The specific radionuclides considered in the calculations and their inventories (in Becquerels) for the Derived Inventory reference case are given in the generic PCSA [27]. In this generic ESC, we consider the impact of uncertainties in the inventory in two ways:

- The generic PCSA [27] provides separate and combined presentation of quantitative results for materials that are already considered as wastes (ILW/LLW and HLW) and for materials that are not yet defined as wastes by the UK Government (SF, Pu, HEU and DNLEU).
- Uncertainties in the inventories of legacy materials that make up the reference case are discussed in the Derived Inventory supporting reports [10, 11, 12, 13]. As discussed in Section 4.1.1, we consider these uncertainties in deriving an upper inventory, which also includes ILW and SF from a proposed programme of new nuclear power stations. The potential impacts of disposing of the upper inventory are discussed qualitatively in Section 5.2.2.5.

In the generic PCSA [27] we present a set of 13 sensitivity calculations for ILW/LLW, a set of 16 sensitivity calculations for HLW and SF, and a set of four sensitivity calculations for DNLEU. The generic PCSA also presents separate results for Pu and HEU. These results demonstrate that the long-term radiological impact of disposing of Pu and/or HEU would be several orders of magnitude lower than SF or DNLEU (see Figure 5.17), and we therefore only discuss the radiological implications of disposing of these materials briefly in this report. This suite of calculations gives an indication of how the calculated risk depends on the values of key parameters representing a wide range of potential geological environments.

The calculations are presented by waste type in order to clarify the key sensitivities and issues that are specific to particular waste types. However, we note that the potentially exposed group for all sets of calculations could be the same, in which case the total calculated dose or risk for the Derived Inventory reference case, for a real GDF, would need to be summed across all wastes contributing to the exposure. Examples of summed impacts are illustrated in the generic PCSA [27], and one example is given at the end of this section (Figure 5.17).

To aid comparison of the different cases, all calculations were performed on the same basis, as set out in Section 4.3.2:

- They were probabilistic, with values of q, T, F, A and C sampled from probability distribution functions representing uncertainty, with uncertainty ranges as set out in the generic PCSA (generally a factor of 3-10 either side of a central value) – see also Table 5.4 and Table 5.5.
- Other uncertain parameter values, such as radionuclide solubilities, sorption coefficients and diffusion coefficients, were also sampled from probability distribution functions, as detailed in an appendix to the generic PCSA, except for two calculation cases specifically discussed below for DNLEU. We note in particular that relatively poor radionuclide sorption properties were adopted for the upper oxidising layer in the geosphere model (see Figure 4.8), i.e. sorption values appropriate for a sandstone rather than a clay. The distribution functions for these parameters were unchanged in our sensitivity calculations, with the exception of the two DNLEU cases.
- The primary radionuclide transport process modelled was advection.
- GDF dimensions were taken from our generic design for higher strength rock.

Calculated risks are generally shown to a time of one million years after GDF closure, for the reasons set out in Section 3.2.2.4. However, for DNLEU we show illustrative results beyond 1 million years, because the peak risk is only reached at later times. However, our confidence in these results is low.

The values of q, T, F, A and C that we have used for the ILW/LLW sensitivity calculations are shown in Table 5.4, and for the HLW sensitivity calculations in Table 5.5. The parameter values used for the SF sensitivity calculations were identical to those used for the HLW calculations. The parameter values used for the DNLEU cases were the same as those used for case ILW1, with some exceptions as discussed separately below.

We provide more detail on cases ILW1, HLW1, SF1 and DNLEU1 as the parameter values used in these cases are consistent with the reference values used by Nirex in previous generic performance assessment work [e.g. 80]. These cases were used previously as a basis for the conduct of disposability assessments. Considering these cases allows us to demonstrate a connection between past and current work on UK GDF development and disposability assessments. In addition, the assumptions on which this set of calculations is based are consistent with assumptions that form the basis of the quantitative assessments considered in the TSC [5] and OSC [6]. The generic PCSA [31] includes detailed analysis of this case, including analysis of intermediate performance measures (in particular, the flux of radionuclides from the engineered barrier system against time).

Table 5.4 Central values of q, T, F, A and C used in the PCSA calculations for ILW/LLW

Uncertainty ranges are factor 10 for q and T and factor of 3 for F and A either side of the central value. For C, because of the nature of the vented waste packages and the types of waste, we assumed that the ILW/LLW inventory was available for release from the start of the post-closure period. The one exception is Case ILW13, in which the effect of choosing C equal to 1,000 years was evaluated, with no uncertainty or variability in the other parameter values. See text for further explanation.

Case	q (metres per year)	T (years)	F (cubic metres per year)	A (square metres)	C (years)
ILW1	6×10^{-4}	100,000	3×10^{5}	10^{7}	0
ILW2	6×10^{-4}	10,000	3×10^{5}	10^{7}	0
ILW3	6×10^{-4}	1,000,000	3×10^{5}	10^{7}	0
ILW4	6×10^{-4}	100,000	3×10^{4}	10^{6}	0
ILW5	6×10^{-4}	10,000	3×10^{4}	10^{6}	0
ILW6	6×10^{-4}	1,000,000	3×10^{4}	10^{6}	0
ILW7	6×10^{-6}	100,000	3×10^{5}	10^{7}	0
ILW8	6×10^{-6}	10,000	3×10^{5}	10^{7}	0
ILW9	6×10^{-6}	1,000,000	3×10^{5}	10^{7}	0
ILW10	6×10^{-6}	100,000	3×10^{4}	10^{6}	0
ILW11	6×10^{-6}	10,000	3×10^{4}	10^{6}	0
ILW12	6×10^{-6}	1,000,000	3×10^{4}	10^{6}	0
ILW13	6×10^{-4}	100,000	3×10^{5}	10^{7}	1,000

Table 5.5 Central values of q, T, F, A and C used in the PCSA calculations for HLW

Uncertainty ranges are factor 10 for q, T and C and factor of 3 for F and A either side of the central value. Note that the parameter values used for the SF sensitivity calculations were identical to those used for the HLW calculations.

Case	q (metres per year)	T (years)	F (cubic metres per year)	A (square metres)	C (years)
HLW1	$6x10^{-4}$	100,000	$3x10^{5}$	10^{7}	500,000
HLW2	$6x10^{-4}$	10,000	$3x10^{5}$	10^{7}	500,000
HLW3	$6x10^{-4}$	1,000,000	$3x10^{5}$	10^{7}	500,000
HLW4	$6x10^{-4}$	100,000	$3x10^{4}$	10^{6}	500,000
HLW5	$6x10^{-4}$	10,000	$3x10^{4}$	10^{6}	500,000
HLW6	$6x10^{-4}$	1,000,000	$3x10^{4}$	10^{6}	500,000
HLW7	$6x10^{-6}$	100,000	$3x10^{5}$	10^{7}	500,000
HLW8	$6x10^{-6}$	10,000	$3x10^{5}$	10^{7}	500,000
HLW9	$6x10^{-6}$	1,000,000	$3x10^{5}$	10^{7}	500,000
HLW10	$6x10^{-6}$	100,000	$3x10^{4}$	10^{6}	500,000
HLW11	$6x10^{-6}$	10,000	$3x10^{4}$	10^{6}	500,000
HLW12	$6x10^{-6}$	1,000,000	$3x10^{4}$	10^{6}	500,000
HLW13	$6x10^{-4}$	100,000	$3x10^{5}$	10^{7}	10,000
HLW14	$6x10^{-4}$	100,000	$3x10^{5}$	10^{7}	0
HLW15	$6x10^{-4}$	10,000	$3x10^{4}$	10^{6}	10,000
HLW16	$6x10^{-4}$	10,000	$3x10^{4}$	10^{6}	0

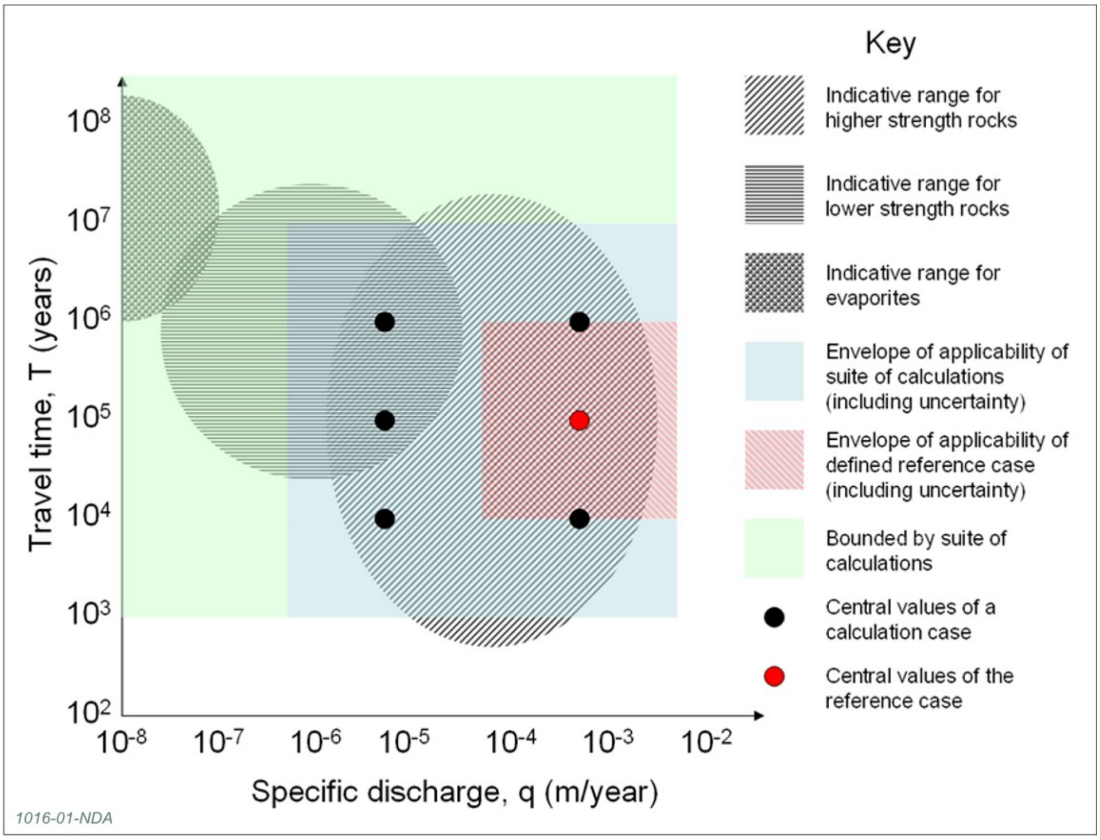
The mean value of 500,000 years for the 'reference case' containment time for HLW/SF containers is derived from the Package evolution status report [15], which justifies a lifetime for copper canisters (as used in our illustrative example disposal concept for higher strength rock) of at least 100,000 years and potentially much longer. SKB expects containment times of at least 1 million years for the copper canisters in their KBS-3V concept (See Appendix A). A mean value of 500,000 years for the C parameter was therefore chosen as representative of a very long-lived container, such as would be required for HLW/SF.

Figure 5.8 presents a 'map' of the q and T values considered in our suite of calculations. In the calculations, q and T are treated as independent parameters, representing different aspects of the groundwater flow field, as described in Section 4.3.2.1 (Figure 4.6). The hatched elliptical areas in Figure 5.8 represent indicative ranges of q and T values for the host rocks in our three illustrative geological environments, namely higher strength rocks, lower strength sedimentary rocks and evaporites. Figure 5.8 therefore shows how the calculations cases can be mapped to representations of the different host rocks. The relationship of the F and A parameters to different geological environments depends mainly on the sequences of rock overlying the host rock, rather than the host rock itself, and these parameters are therefore difficult to illustrate on such a figure.

Each large dot in Figure 5.8 represents a set of central values for q and T as used in the generic PCSA sensitivity cases. Each sensitivity case shown in Table 5.4 and in Table 5.5 corresponds to one of the six dots. The pale blue region shows the area of q and T values covered by the suite of calculations, including the uncertainty ranges considered for q and T. The pale green region shows the area of q and T values that fall outside the suite of calculations, i.e. q is less than the minimum value considered and/or T is greater than the maximum value considered. The indicative ranges of q and T values for different geological environments vary widely. In general, lower values of q and higher values of T are more favourable from the viewpoint of geological containment with respect to groundwater-mediated transport. However, such geological settings may be less favourable in other respects, for example impacts of gas generation, human intrusion or ease of construction and operation.

Figure 5.8 Relationship of values of specific discharge (q, metres per year) and groundwater travel time (T, years) to different host rocks

Lower values of q and higher values of T are more favourable from the viewpoint of the geological containment safety function. Note that both axes are logarithmic. See text for explanation of figure.



Our sensitivity analyses cover nearly the complete range of possibilities for higher strength host rocks and a significant part of the possible parameter value range for lower strength host rocks. The cases considered with the lowest value of q and the largest value of T (cases ILW9/HLW9 and ILW12/HLW12) are illustrative of geological environments having very high containment potential. For example, for a model run with a groundwater travel time of one million years, there would be essentially no impact from groundwater-mediated releases from a GDF for about one million years (although dispersion in the geosphere could give rise to limited releases in advance of one million years). However, as Figure 5.8 illustrates, some geological environments could provide an even greater degree of containment than shown in our calculations. The suite of calculations therefore does not cover the full range of geological environments in the UK, but, as discussed in Section 3.1.4.3 and in the generic PCSA [27], it is likely to provide a conservative basis for decision making under the disposability assessment process for geological environments in which advective flow is negligible and radionuclide transport is dominated by diffusion.

The rationale behind our choice of values for the C parameter is as follows:

- For whatever ILW/LLW concept we eventually choose, it is unlikely that it will be possible to claim much of an absolute containment time in the waste package because of the nature of the vented waste containers and the types of waste.[20] Therefore, zero containment time in the waste package was assumed for all cases apart from Case ILW13, in which the effect of choosing C equal to 1,000 years was evaluated.[21] With C set to zero, the entire ILW/LLW inventory is assumed to be available for release from the container from the start of the post-closure period. We are undertaking supporting assessment work in which we model in greater detail the release of radionuclides from an ILW/LLW package and radionuclide movement through the near-field in the post-closure period. This work may allow us to justify longer containment times in due course.

- For the HLW/SF/Pu/HEU disposal concept, various types of container material could be adopted. In most of the calculations, a central value of 500,000 years for C was assumed (with an order of magnitude variation each way), representative of a long-lived container material (such as, but not restricted to, copper). There is also a factor of two variability between containers (i.e., at the lower end of the uncertainty range, containers fail gradually over the period 50,000 years to 100,000 years, and at the higher end of the uncertainty range, containers fail gradually over the period 5 million to 10 million years). Variant cases were considered as shown in Table 5.5 for a shorter-lived container material such as iron (central value of 10,000 years), and for no containment time. This choice of values is discussed in the generic PCSA [27], and issues associated with container lifetime are considered in the Package evolution status report [15].

We recognise that for some geological environments, notably those where there are substantial regions of negligible advective flow, or where very low risks are calculated, a performance assessment might actually be put together rather differently. For example, it might be possible to make a safety case by making bounding arguments rather than carrying out probabilistic assessment calculations, and a model of diffusive transport might be required.

The calculated results from the sensitivity cases are compared to each other and to the risk guidance level for the post-closure period given in the GRA [1]. The results provide an indication of the extent of engineered containment that could be required to be consistent with the regulatory risk guidance level, depending on the properties of the geological environments available at the candidate site(s). We can use this information in the MRWS Site Selection Process to provide an early insight into the kinds of disposal concept and engineered containment that would be needed, and the way in which we would need to construct a safety case for particular environments, characterised by the parameter value ranges we have used in our calculations. As noted previously, we will also use the calculated results as a benchmark under our disposability assessment process for the packaging of wastes.

The results of the PCSA calculations for the groundwater pathway are discussed in more detail in the sections below.

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[20] However, even if containment in the waste package is minimal, the EBS may play a significant role in retarding the transport of radionuclides into the geosphere – see discussion of this issue in Section 5.2.1.2. Our PCSA calculations consider the wider containment properties of the EBS, but these are not varied from one sensitivity case to the next.

[21] For the ILW variant with C = 1,000 years, no uncertainty or variability in other parameter values was included.

ILW/LLW

The calculated mean annual individual risk versus time for ILW/LLW case ILW1 is shown in Figure 5.9 for key radionuclides. The calculated mean risk is near zero for the first 5,000 years following GDF closure. The highest fluxes of radionuclides from the EBS into groundwater in the geosphere reducing layer at early times are due to caesium-137, strontium-90 and americium-241. These radionuclides are relatively short-lived, with half-lives of 30, 29 and 433 years respectively. In reality, we would expect these radionuclides to be contained within the EBS, but in all but one of our ILW calculation cases (ILW13), it is cautiously assumed that they are immediately available to dissolve in porewater within the backfill. As their half-lives are small compared to the groundwater travel time, they give rise to negligible risk.

Figure 5.9 Calculated mean annual individual risk versus time for groundwater-borne releases for ILW/LLW case ILW1, showing the contribution to risk from key radionuclides

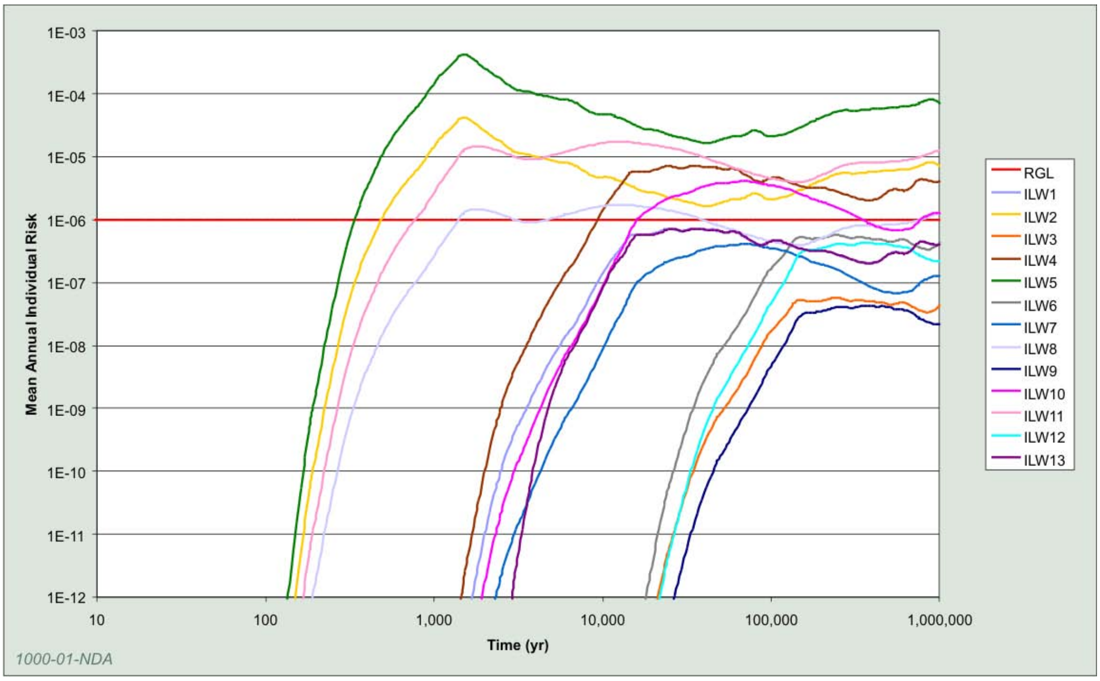
Horizontal red line is the risk guidance level (RGL) from the GRA.

From 5,000 years until after 100,000 years, the calculated mean risk is predominantly due to iodine-129 and chlorine-36. The fluxes of iodine-129 and chlorine-36 leaving the EBS are both comparatively low – in terms of activity, together they make up less than 0.1% of the initial radionuclide flux leaving the EBS. Both iodine-129 and chlorine-36 have long half-lives (15.7 million years and 302,000 years respectively), and their chemical properties are such that they are highly soluble and not readily sorbed, either in the EBS or in the geosphere. The calculated peak risk from these radionuclides is therefore largely determined by the time it takes them to migrate through the geosphere (determined by T) and the extent to which any emerging radionuclide plume is mixed with other groundwater in the overlying rocks (determined by F).

Beyond 100,000 years, the calculated mean risks are mainly due to decay products of uranium-238 (radium-226, lead-210 and uranium-234). Between 100,000 years and one million years, there is also a contribution from neptunium-237 and its daughters (thorium-229 and uranium-233). Beyond about 200,000 years, the only radionuclides still emerging from the EBS at a significant level are uranium-238 and its daughters. Uranium-238 has a very long half-life, approximately 4,500 million years. Uranium is not very soluble, and is strongly sorbed onto the backfill and onto the rocks within the geosphere. Therefore, uranium is only released slowly from the engineered system and migrates slowly through the geosphere to the biosphere. The multiple barriers in the disposal concept work together to delay and disperse its eventual release, so that it does not reach the biosphere in significant concentrations. Figure 5.10 shows the calculated mean annual risk from all radionuclides for the 13 sensitivity cases, each case representing a different set of probability distribution functions for q, T, F, A and C, as shown in Table 5.4.

Figure 5.10 Calculated mean annual risk versus time for groundwater-borne releases for ILW/LLW sensitivity studies

Horizontal red line is the risk guidance level (RGL) from the GRA.



Comparison of cases ILW1, ILW2 and ILW3 shows that the effect of increasing travel time through the geosphere (T) is to decrease the value of the calculated peak risk, whilst increasing the time at which the peak risk occurs.

Comparison of cases ILW1 and ILW7 shows that decreasing the specific discharge velocity in the near field (q) for a long geosphere travel time (100,000 years) decreases the calculated peak risk as less activity is released into the geosphere. The same variation is seen with the comparison of cases ILW3 and ILW9, which also have a long geosphere travel time (1,000,000 years). For the cases where there is a relatively short travel time of 10,000 years (cases ILW2 and ILW8), the decrease in q results in a decrease in calculated peak risk, but the time of peak risk has increased in case ILW8. In each of these cases, the peak is primarily due to radionuclides such as chlorine-36 and iodine-129, which are highly soluble and not strongly sorbed in either the EBS or the geosphere.

The effect of decreasing the groundwater flux (F) and discharge area (A) is shown in the comparison betweens cases ILW1 and ILW4. The time of peak risk is unchanged, but the peak risk is increased by an order of magnitude as the activity emerging into the biosphere is concentrated into a smaller area and volume.

HLW

The calculated mean annual individual risk versus time for case HLW1 is shown in Figure 5.11 for key radionuclides. The major contributors to peak risk at 1,000,000 years are caesium-135 and selenium-79. Iodine-129 and, to a lesser extent, chlorine-36 contribute to early risks. Activity does not emerge from the EBS into the geosphere until 50,000 years, which is at the lower end of the probability distribution function for container failure time (C) for case HLW1. By this time, most short-lived radionuclides have decayed. Caesium-135 dominates the flux throughout the simulation period, with a significant contribution from actinium-227 and radium-226 at later times.

Figure 5.11 Calculated mean annual individual risk versus time for groundwater-borne releases for case HLW1, showing the contribution to risk from key radionuclides

Horizontal red line is the risk guidance level (RGL) from the GRA.

Figure 5.12 shows the calculated mean annual risk from all radionuclides for the 15 sensitivity cases, each case representing a different set of probability distribution functions for q, T, F, A and C, as shown in Table 5.5.

Comparison of the plots and values for cases HLW2, HLW1 and HLW3 respectively shows the effect of increasing travel time through the geosphere (T) is to decrease the value of the calculated peak risk whilst increasing the time at which the peak risk occurs.

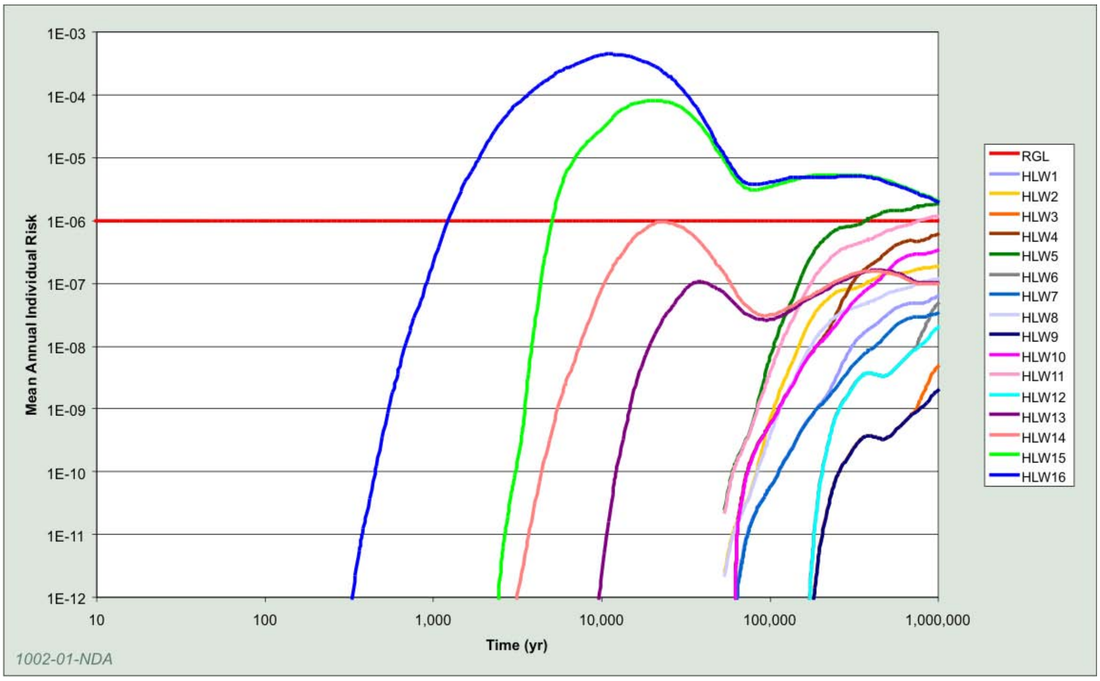
Comparison of cases HLW1 and HLW7 shows that decreasing the specific discharge velocity in the near field (q) for a long geosphere travel time (100,000 years) decreases the calculated peak risk, as less activity is released into the geosphere. The time of peak risk for both cases is the same. The same variation is seen with the comparison of cases HLW3 and HLW9, which also have a long geosphere travel time (1,000,000 years). For the cases where there is a relatively short travel time of 10,000 years (cases HLW2 and HLW8), the decrease in q results in a decrease in peak risk, but the time of peak risk has increased in case HLW8.

The effect of decreasing the groundwater flux (F) and discharge area (A) is shown in the comparison betweens cases HLW1 and HLW4. The time of peak risk is unchanged, but the peak risk is increased by an order of magnitude as the activity emerging into the biosphere is concentrated into a smaller area and volume.

The impact of varying container lifetime, with all other parameter values unchanged, is shown in the comparison of cases HLW1 and HLW13. The calculated peak risks for the shorter-lived container are about twice those of the longer-lived container, and the time of peak risk is much earlier.

Figure 5.12 Calculated mean annual risk versus time for groundwater-borne releases for HLW sensitivity studies

Horizontal red line is the risk guidance level (RGL) from the GRA.



Spent nuclear fuel

The calculated mean annual individual risk versus time for case SF1 is shown in Figure 5.13 for key radionuclides. For both AGR SF and PWR SF, the peak risks are dominated by iodine-129 at 1,000,000 years, with a significant contribution from caesium-135. Selenium-79 makes a contribution at later times, and chlorine-36 at earlier times. AGR SF is the major contributor to risk. Again, activity does not emerge from the EBS into the geosphere until 50,000 years. Nickel-59 contributes the most to the total flux at earlier times, and actinium-227 and radium-226 make the major contribution at later times.

Figure 5.14 shows the calculated mean annual risk from all radionuclides for the 15 sensitivity cases, each case representing a different set of probability distribution functions for q, T, F, A and C, as shown in Table 5.5. Overall, calculated peak risks and the times of peak risks are similar for HLW and SF. In general, these sensitivity calculations illustrate the same features as our HLW sensitivity calculations, with only minor differences as discussed in the generic PCSA [27]; these are not discussed any further here.

Figure 5.13 Calculated mean annual individual risk versus time for groundwater-borne releases for case SF1, showing the contribution to risk from key radionuclides

Horizontal red line is the risk guidance level (RGL) from the GRA.

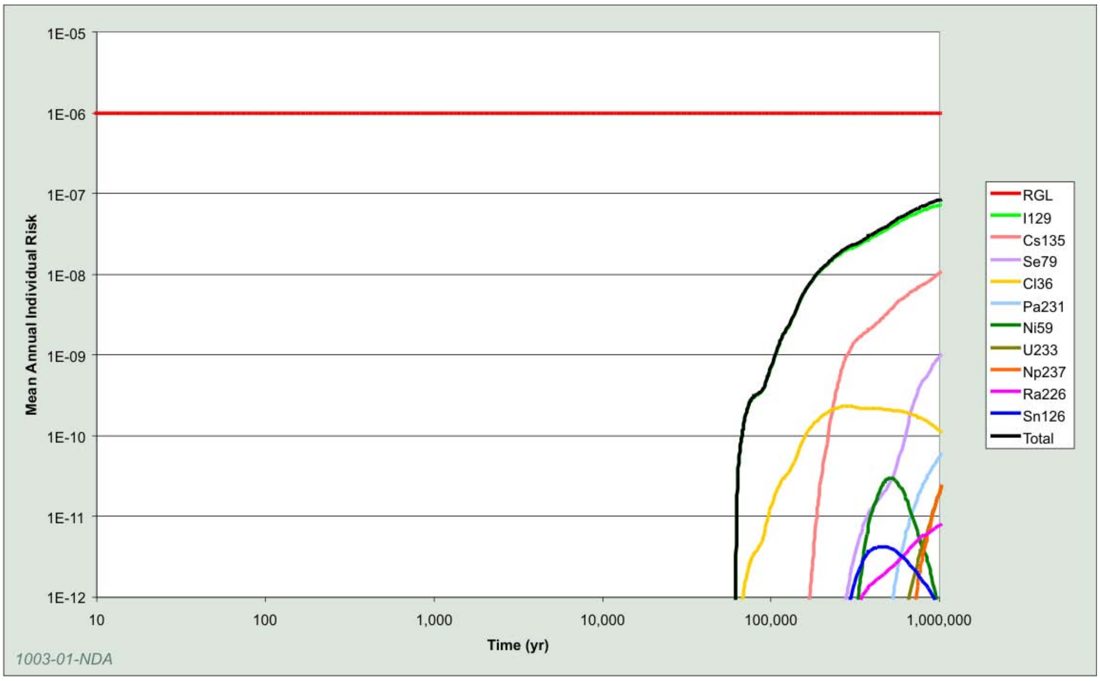
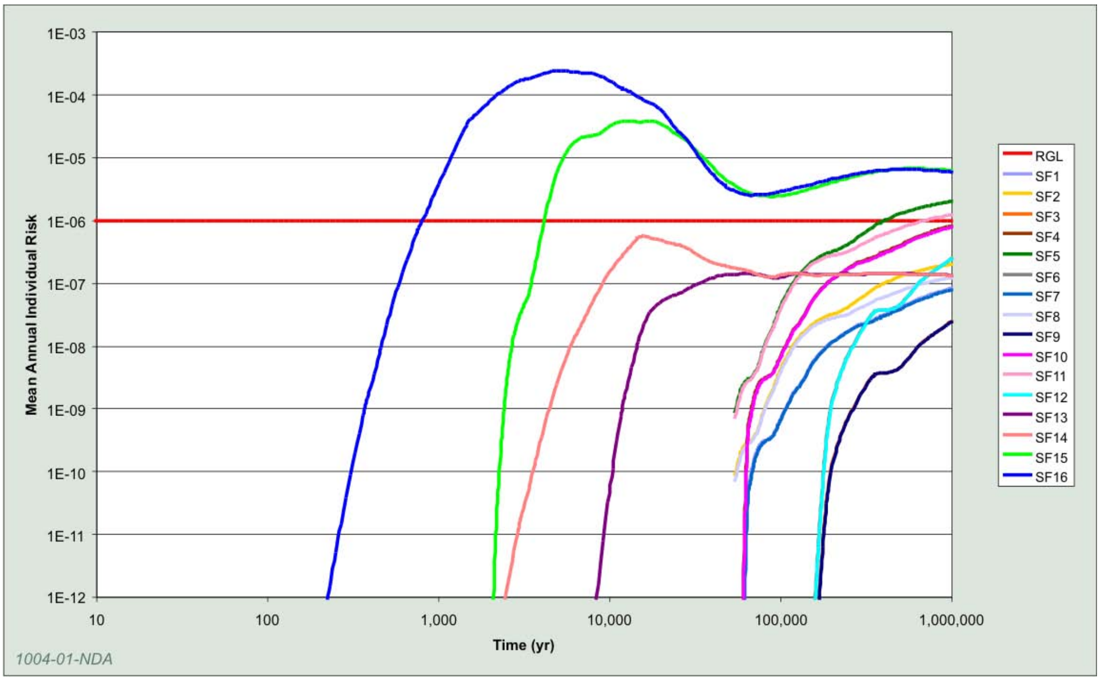


Figure 5.14 Calculated mean annual risk versus time for groundwater-borne releases for SF sensitivity studies

Horizontal red line is the risk guidance level (RGL) from the GRA.



Separated Pu and HEU

The total calculated risks from separated Pu and HEU for radionuclide transport by groundwater are negligible, being more than 1,000 times lower than the calculated risks from HLW or SF (e.g. see Figure 5.17). The calculated risk from separated plutonium is negligible because it is highly retarded by sorption in the geosphere and, hence, almost all of the plutonium-239 and plutonium-240 in its inventory would decay before reaching the biosphere. The calculated risk from HEU is negligible because its inventory is very small compared to other materials we are considering that contain isotopes of uranium. The generic PCSA [27] contains further information on the assessment of these materials.

DNLEU

If the large inventory of DNLEU in the UK were to be declared as a waste and disposed of in a GDF, we anticipate that it would occupy a dedicated disposal vault (or vaults). This would enable a greater control over the chemical environment of the DNLEU as it would not be mixed with other wastes. There is also the option of using additional physical barriers (for example, perhaps a clay lining to the vault) to improve the containment of the uranium. However, these would be the sort of issues that would be considered during optioneering for the geological disposal of this material. For this generic assessment, it is assumed that the DNLEU disposal vaults have the same design as the ILW vaults.

Uranium-238 is an important radionuclide from the point of view of post-closure safety, and the inventory of uranium-238 in the DNLEU is approximately ten times larger than the uranium-238 inventory in the rest of the materials considered. The calculated peak risk from the daughters of uranium-238 depends particularly on the following parameters:

- the inventory of uranium;
- the solubility of uranium in a GDF;
- the sorption of the daughters thorium-230, radium-226 and lead-210 in the top tens of metres of the geosphere; and
- the parameters representing the hydrogeological characteristics of the geosphere, in particular the value of the specific discharge into the host rock, q.

Four sensitivity cases were carried out for DNLEU to illustrate the importance of the uncertainties associated with these parameters. All assumptions and parameter values were as for our ILW/LLW cases, except as noted below:

- DNLEU1 – DNLEU inventory, all other parameter values as in case ILW1. Uncertainty in solubility limits and sorption coefficients considered using probability distribution functions as for the ILW cases (given in [27]).
- DNLEU2 – DNLEU inventory, all other parameter values as in case ILW1, except uranium solubility, which was given its best estimate value.
- DNLEU3 – DNLEU inventory, all other parameter values as in case ILW1, except sorption coefficients in the geosphere for radium, thorium and lead, which were given best estimate values.
- DNLEU4 – DNLEU inventory, geosphere parameter values as in case ILW7 (value of q decreased by factor 100 compared to case ILW1). Uncertainty in solubility limits and sorption coefficients considered using probability distribution functions as for the ILW cases.

The calculated mean annual individual risk versus time for case DNLEU1 is shown in Figure 5.15 for key radionuclides, and the calculated mean annual individual risks for the four sensitivity cases are shown in Figure 5.16.

Figure 5.15 Calculated mean annual individual risk versus time for groundwater-borne releases for case DNLEU1, showing the contribution to risk from key radionuclides

Horizontal red line is the risk guidance level (RGL) from the GRA.

Figure 5.16 Calculated mean annual risk versus time for groundwater-borne releases for DNLEU sensitivity studies

Horizontal red line is the risk guidance level from the GRA.

There is no significant risk from DNLEU until around 80,000 years. After this time contributions arise predominately from the uranium isotopes and the daughter products from the uranium-238 decay chain. As is the case for the calculated radiological risk for ILW, the greatest radiological impact of uranium-238 arises from its decay products (particularly thorium-230, radium-226 and lead-210). Each of the variant assumptions in cases DNLEU2, DNLEU3 and DNLEU4 gives rise to generally lower calculated risks, with the values of uranium solubility and specific discharge having the most significant impact on calculated risk.

Owing to the much larger inventory of uranium-238 in the DNLEU in comparison to the ILW, the calculated mean risk from the daughters of uranium-238 slightly exceeds the GRA risk guidance level millions of years in the future. We therefore extended the calculation of mean annual individual risk to 100 million years. The calculated risks from individual radionuclides at these very long times in the future are highly uncertain and fluctuate in a manner indicating that these results are not fully converged (i.e. that we have run insufficient calculations to obtain a meaningful result). The calculated mean risk is dominated by the contribution of just a few realisations that sample particularly pessimistic values for certain chemical parameters.

Summary of radionuclide transport by groundwater for all waste types

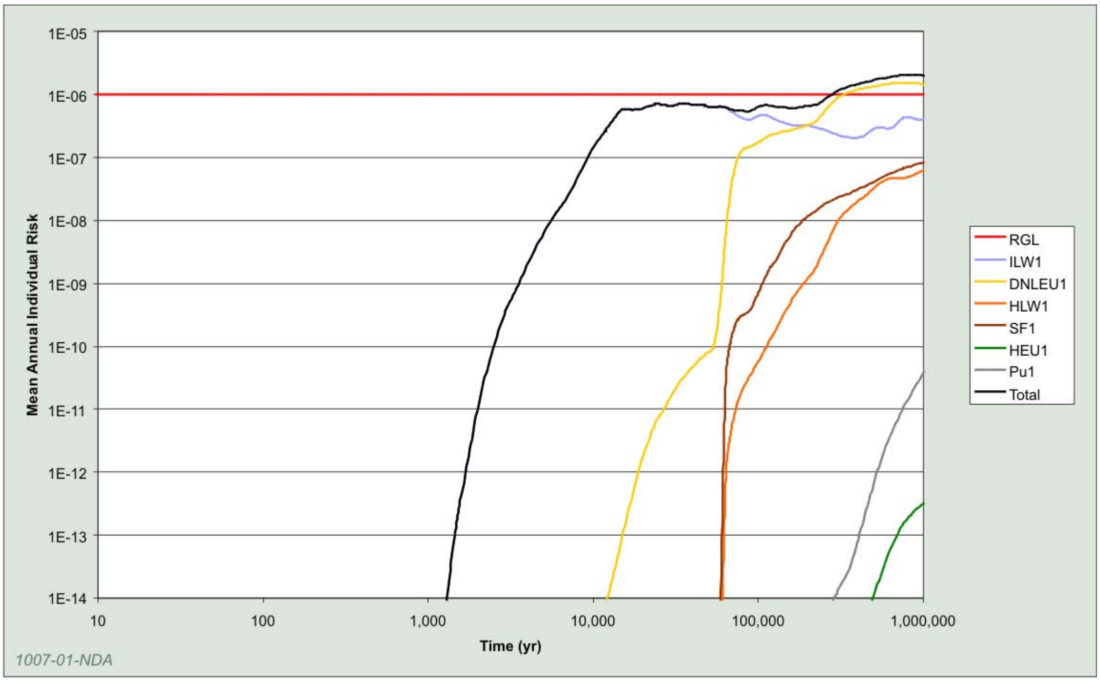
We have illustrated an approach to undertaking calculations of risk for radionuclide transport by groundwater. However, there is a limit on the extent to which meaningful calculations can be carried out for the groundwater pathway at a generic stage, when the candidate site(s) and disposal concepts are not known. In our illustrative calculations, it is the *relative* contributions to risk from different components of a GDF and from different radionuclides in the inventory that are of most interest at this stage, rather than the *precise* values of risk, which are not very meaningful.

The calculated peak annual individual risk from radionuclide transport by groundwater occurs thousands of years or more in the future across the full range of sensitivity cases we have analysed. For these cases, the calculated (mean) peak risk ranges from several orders of magnitude below the GRA risk guidance level of one in a million (10^{-6}) per year, to two orders of magnitude greater, comparable to the radiological risk from natural background radiation in the UK. Some of the most important radionuclides in terms of their contribution to the total calculated risk are the soluble and mobile chlorine-36 and iodine-129 and, at later times, the daughters of uranium-238 and neptunium-237 and its daughters.

Figure 5.17 shows the mean annual individual risk for cases ILW1, HLW1, SF1, Pu1, HEU1 and DNLEU1, and the summed total for all waste types.

Figure 5.17 Calculated total mean annual individual risk for groundwater-borne releases, showing the contribution from cases ILW1, HLW1, SF1, Pu1, HEU1 and DNLEU1

Horizontal red line is the risk guidance level (RGL) from the GRA. In this set of cases, the contribution from DNLEU exceeds the risk guidance level at about 300,000 years post-closure. Note that the assumptions on which this set of calculations is based are consistent with the assumptions considered in the TSC [5] and OSC [6]. The generic PCSA [27] includes detailed analysis of this case.



Way forward for PCSA

Once candidate sites are known, more information about the local geological environment will become available as part of our desk-based studies in Stage 4 of the MRWS Site Selection Process. This will enable us to refine the PCSA methodology and focus our research and site characterisation programmes on reducing areas of uncertainty. There will still be significant uncertainties in the high-level q, T, F and A parameters representing the geological environment at the candidate site(s), but we should at least be able to narrow this uncertainty to some extent, allowing ranges for these parameters to be estimated. As a consequence, the emphasis of the presentation of calculation results will shift from *indicative* and *relative* values of risk arising from different components of the inventory and different radionuclides, towards presenting *absolute* values of risk and associated confidence limits for comparison with the risk guidance level.

Some candidate sites could be outside the envelope of applicability of the calculations for the groundwater pathway presented in the generic PCSA. For such sites, different models may need to be developed that represent processes not included in the total system models at this generic stage. As noted in Section 4.3.2.1, depending on the geological environment at the candidate site(s), the representation of the geosphere may need to include processes such as transport dominated by diffusion rather than advection, fracture flow, and diffusion into the rock matrix, none of which have been explicitly treated in the generic PCSA calculations.

As yet more information about the sites becomes available in Stages 5 and 6 of the MRWS Site Selection Process, we will be able to build greater understanding of the likely behaviour of the geological disposal system and, in turn, improve our confidence in the assessment of long-term safety. Once we have data from a site investigation programme, we will not continue with the approach of directly estimating uncertainty ranges for q, T, F and A. Rather, we will develop and construct conceptual and mathematical models for groundwater flow based on our understanding of the candidate site(s). The results of these models may determine the uncertainty ranges for parameters such as q, T, F and A in the total system model. If so, we will continue to discuss q, T, F, and A to provide a link to previous work, but these will be system characteristics resulting from detailed modelling, rather than assumed parameter value ranges.

We will also need to develop more detailed models of the EBS. Emphasis will need to be placed on investigating uncertainties associated with proposed engineered barrier systems, and with variant scenarios that could alter the groundwater flow regime, such as major climate change. An appropriate treatment of correlations between parameters that are uncertain will also be required.

5.2.2.2 Consequences of gas

We summarise here our understanding of the key issues relating to gas generation in a GDF, and the consequences of GDF-derived gas in a range of geological environments. This section draws on the Gas status report [19] and the generic PCSA [27].

Gas generation

The key gas generation processes applicable to a GDF in the post-closure period are the corrosion of metals giving hydrogen, microbial degradation of organic material giving mainly methane and carbon dioxide, and radiolysis of water and organic materials giving mainly hydrogen. The relative importance of each gas generation process and, hence, the rates of gas formation and the gas composition depend on the wasteform, the waste package, the chemical environment provided by any buffer or backfill, and the host rock environment, particularly the availability of water.

Corrosion and radiolysis processes relevant to waste packages in likely GDF environments are relatively well understood and allow the rate at which bulk gas is generated to be calculated. Calculation of the rates of gas production from microbial processes is difficult because the wastes have a varying susceptibility to microbial attack, and there are a multitude of micro-organisms with different metabolisms, habitats and adaptation capabilities. However, conditions within in a GDF after closure are much less favourable to microbial growth than those in a conventional landfill, although microbial populations would be expected to adapt with time. The rate of gas generation from microbial action on radioactive wastes in a GDF would be expected to be much less than those from a similar volume of domestic waste in a landfill. Therefore, it is possible to understand the key issues well enough to be confident of the relative magnitude of the constituents of the bulk gas. The bulk gas would be comprised largely of hydrogen, but there would be smaller amounts of carbon dioxide and methane. The hydrogen is generated mainly from corrosion of metals in a GDF.

The main radioactive gases produced would be tritium, carbon-14 labelled species (e.g. methane and carbon dioxide) and radon-222. There is uncertainty about the rates at which these gases will be formed because the rates depend, first, on the release of a radionuclide from the waste matrix and, second, the incorporation of the radionuclide into a gas. In developing our understanding of the consequences of gas generated in a GDF, we have made some assumptions that may lead to larger calculated rates of radioactive gas generation than might actually be the case in a GDF.

Carbon-14 labelled species will be the most important with regard to the post-closure release of gas because this radionuclide has a relatively long half life (~5,730 years). In contrast, tritium has a short half life (12.3 years), and so will have decayed significantly within a few hundred years of packaging. Although radon-222 will be formed continually as part of the radioactive decay of uranium, it also has a short half life (3.82 days) and will decay significantly before it can migrate from a GDF. Therefore, our quantitative assessment of the radiological impacts of gas-phase releases of radioactivity focuses on carbon-14.

Gas migration

The migration of GDF-derived gases and, hence, the radiological impacts of gas generation, would be specific to both a site and the design of a GDF. On the basis of our work to date [19], we conclude the following with regard to gas migration in a range of geological environments:

- In the case of a higher strength host rock, a free gas phase may form and migrate away from a GDF (rates of gas generation are unlikely to be limited by the availability of groundwater). In order to determine where the gas would migrate and when it might be released at the surface, the properties of the host rock and overlying geological formations are important. In particular, geological features (e.g. ‘cap rocks’) may act as barriers to the migration of the gas, while fault zones may or may not act as conduits depending on their ability to maintain a free gas pathway. The volume of water that is available for the gas to dissolve in is also important and would be determined by flow rate and porosity in the overlying rocks. Possible consequences of gas in this geological environment include entrainment of contaminated water by gas and the release of flammable (i.e., hydrogen and methane) and radiotoxic (i.e. carbon-14 labelled methane) gases to the biosphere. Assessments have shown that the flammable gases are of little concern [19]. In contrast, an important area of our research work is to investigate the rate at which carbon-14 labelled methane might be produced from the waste, and the potential for this gas to migrate through the geosphere. Our current understanding of these issues is set out in some detail in the Gas status report [19].
- In the case of a lower strength sedimentary host rock (e.g. clays), the rates of gas generation may be limited by the supply of water from the host rock to a GDF. It would be difficult for any free gas phase formed to migrate from a GDF because the clay minerals typical of such geological environments have small inter-granular pores and the gas entry pressure is high [19]. Depending on the combination of gas generation rate, water inflow and gas migration in solution, the gas may be released through a combination of dilation and microfissuring in the clay. These pathways are then expected to close after the gas pressure has fallen, depending on the properties of the host rock. Possible consequences of gas in this geological environment include over-pressurisation of a GDF, with potential damage to the engineered barriers and the host rock, and the displacement of contaminated water from the disposal areas. However, this could be prevented by suitable design of the EBS and associated seals. Our assessment of such a geological environment would require investigation of the coupled processes of gas generation and multiphase flow in the vicinity of a GDF, and we are undertaking R&D work on this topic within the UK and as part of wider international research in this area [109]. The results of this work will also be relevant to developing our approaches for modelling of such coupled processes in other geological environments.
- In the case of an evaporite host rock, a GDF environment would likely be even more isolated from infiltrating water than for a lower strength sedimentary host rock. Many of the issues that arise are similar to the case of lower strength sedimentary rocks, with the one difference that an evaporite host rock will creep (i.e., move slowly under the influence of the lithostatic pressure) to a greater extent than a lower strength sedimentary host rock and fill voidage.

In summary, the potential consequences associated with GDF-derived gas would be site specific, and would need to be addressed in detail as part of site evaluation. However, there is a lot of international experience with different concepts and sites that we can draw upon to develop our programme in the absence of a possible site or sites. Information from the UK and abroad shows that the two primary gas issues to be considered for a GDF are:

- the need for the system pressure to remain below a value at which irreversible damage to the EBS and host geology occurs; and
- the need to ensure that the flux of gas (in particular gaseous radionuclides) to the biosphere does not result in unacceptable risk.

A recent regulatory review of our gas programme [41] noted that we have appropriate tools for modelling gas migration mechanisms, but would be unable to take our understanding further before we have specific sites to consider. As noted in Section 2.3, the recommendations of this review have been considered in developing the Gas status report [19].

We have considered the consequences associated with GDF-derived gas for a range of geological environments [166, 167]. These studies demonstrate our understanding of the relevant features, events and processes affecting GDF-derived gas; we can deploy this understanding at a site-specific level as the MRWS Site Selection Process proceeds. Understanding of the potential consequences of gas during the post-closure period has moved on significantly in the last several years, and we are confident that management strategies can be adopted to ensure that GDF-derived gas will not present a challenge to the post-closure safety of a GDF.

5.2.2.3 Human intrusion

Over the timescale for assessments of the post-closure safety of a GDF, the form of human society will change in a manner that cannot be predicted. Our approach is therefore to develop scenarios based on current technology and patterns of behaviour in the locality of the site or, in the absence of suitable information, at similar locations. In line with the GRA [1], the modes of intrusion we consider are those that might occur given present economic needs and technology and the current pattern of resource exploitation. This section illustrates how a particular example could be assessed.

In our example, we assume that people inadvertently drill through a GDF while exploring for natural resources in the far future, after knowledge of the GDF and/or its purpose have been lost to society. We consider such an event extremely unlikely, because the purpose of Stage 2 of the MRWS Site Selection Process is to screen out sites having natural resources (see Section 2.1). Nonetheless, we are evaluating the consequence of events that could bypass or damage the multiple barrier system, as an input to optimisation studies. Although we will not have a preferred site and disposal concept until the end of Stage 5 of the MRWS Site Selection Process, our intention here is to illustrate how the consequences of human intrusion could be assessed for a particular scenario.

To evaluate the radiological consequences of this mode of human intrusion, we assume that, following penetration of a GDF, radioactive material is brought to the surface and its radioactive nature is not recognised. We can identify several groups that would potentially be exposed as a result of this mode of intrusion, for example workers exposed to drill core and individuals who later reside in the area where the drilling occurred. For the purposes of illustration, the focus here is on a potentially exposed group that consists of geotechnical workers who examine drill core in the laboratory. We can already make a reasonable evaluation of the radiological doses such a group would receive because they are largely independent of site-specific factors.

Several activities may occur during laboratory analysis of core material that give rise to exposure. These exposure pathways are:

1. external exposure from short-term working in close proximity to core samples, and from longer-term, more distant irradiation while materials are stored in the laboratory;
2. ingestion as a result of handling samples leading to activity being taken into the body orally from contamination of the hands; and
3. inhalation of dusts generated as a result of laboratory analysis techniques, and of radon, associated with the decay of radium-226 that may be present in core samples stored in the laboratory.

The following GDF-specific factors are those considered to be of greatest importance in the calculation of radiological dose to a geotechnical worker in this scenario:

- the radionuclide inventory in a GDF, which in our illustrative example calculations is assumed to be distributed homogeneously within each waste type (separate calculations could be undertaken for intrusion into different waste types); and
- the design of a GDF, particularly its layout (i.e., the amount of waste material that will be encountered if it is penetrated by a borehole).

As for the calculations of radionuclide transport by groundwater, the inventories have been taken from the Derived Inventory reference case [10, 11, 12] and the GDF layout from [24]. Other parameter values are as provided in the generic PCSA [27]. We conservatively assume that the inventory remains in a GDF with no radionuclides transported away from it prior to the intrusion event occurring, and we assume that the intrusion can take place at any time in the post-closure period. Therefore, the calculated doses for intrusion into a particular waste type scale to the decay of the inventory for that waste type (shown in Figure 4.1). The actual calculated doses for ILW and HLW are provided in the generic PCSA [27].

Clearly, in this postulated highly unlikely scenario, it is possible to calculate high radiological doses, particularly for HLW (or SF), if workers handle raw wastes (of the order of Sieverts). That is an important reason why geological disposal is the UK Government's policy for such waste.

When we are able to make site-specific assessments of human intrusion, we will use information from the potential GDF location to derive additional scenarios.

5.2.2.4 Criticality safety

We need to understand criticality safety issues for all materials in the Baseline Inventory containing significant quantities of fissile materials – in this regard, ILW and separated Pu and HEU are of particular concern. HLW and SF are of less concern for the following reasons:

- HLW arises from the reprocessing of SF, and contains little fissile material because this has been removed during reprocessing and is managed separately.
- Most SF is removed from nuclear reactors because a proportion of the fissile content has been used up and neutron-absorbing fission products have built up during irradiation, such that the fuel becomes less able to sustain a nuclear chain reaction. Also, SF would be packaged and placed in a GDF according to a design that is capable of maintaining sub-critical conditions over timescales sufficient to ensure that most of the plutonium-239 in the SF would decay to uranium-235 within the disposal package; this uranium-235 would be diluted by neutron-absorbing uranium-238 within the SF.

The potential for, and possible implications of, a criticality event in a GDF are discussed in the Criticality safety status report [21]. In the PCSA [27] we summarise arguments to explain why we believe a post-closure criticality event to be unlikely, and, if such an event did occur, why the consequences would not be significant in terms of environmental safety.

To date most criticality safety work in the UK related to disposal requirements has focused on ILW because of the many different types of ILW and the need to provide packaging advice to waste producers over the past 15 or so years, and we focus on that work here. We summarise the work on ILW first, and then consider criticality safety for separated plutonium and HEU.

Criticality safety for ILW

The main radionuclides in ILW that can undergo fission are uranium-235 and plutonium-239. The inventories of fissile radionuclides such as uranium-233 and plutonium-241 are small relative to uranium-235 and plutonium-239, and do not present a significant criticality safety concern.

In ILW the fissile material is nearly always mixed with a large excess of non-fissile material such that the concentrations of fissile materials are well below critical values. Small amounts of ILW will contain plutonium or HEU, but these are not present as pure materials and are mixed with other non-fissile material. Our disposability assessment process [14] sets a limit on the allowed level of fissile material for ILW packages. This limit ensures that a criticality is not possible for any credible configuration of packages under conditions that might occur during waste storage, transport or emplacement within a GDF.

For ILW, we consider those criticality events that may arise in a GDF through the following mechanisms under post-closure conditions [168]:

1. reconfiguration of fissile and other materials within the confines of their original packaging as a GDF evolves;
2. accumulation of fissile material outside the waste package at some location in a GDF or its immediate locality following transport of the fissile material in groundwater; and
3. concentration of fissile material by slumping as other materials in a small region of a stack of waste packages are dissolved.

The second mechanism considers transport in groundwater of dissolved fissile material, colloidal fissile material and particulate fissile material and their accumulation. We consider potential accumulation regions within a GDF, within the host rock in the immediate vicinity of a GDF, and within the geosphere at some distance from a GDF. However, we would expect that flow through a GDF would spread as it moved away from a GDF, which would tend to mean that accumulation would be less likely with increasing distance from a GDF. Furthermore, we anticipate that the pH of the groundwater would tend to decrease with increasing distance from a GDF, and that conditions would tend to become more oxidising if flow was upwards into modern shallow waters. In these circumstances, the solubility of radionuclides would tend to increase and sorption decrease. Again, this would tend to make accumulation less likely.

For all three mechanisms, it would take a long time, probably at least thousands of years, to accumulate a critical mass under GDF conditions. As noted in the Criticality safety status report [21], the likelihood of any of these mechanisms taking place is low, but we are continuing to examine this issue.

Were a critical accumulation of fissile material to occur, two different types of criticality could occur depending on the concentration of fissile nuclides in the critical mass, namely “quasi-steady state” and “rapid transient” criticalities. For the quasi-steady state criticalities, system reactivity decreases with an increase in temperature. This might lead to criticalities lasting for long periods under GDF conditions. Conversely, for a rapid transient criticality, system reactivity increases with temperature and energetic events might result. Further information on these two types of criticality event and the physical conditions that could produce them is given in the Criticality safety status report [21, Section 6.1.1].

In 2008 we carried out a generic post-closure criticality consequence assessment, aimed at developing an understanding of criticality safety for ILW under GDF conditions [168]. The study was based on information from the Nirex 2003 Generic Performance Assessment for ILW in a higher strength rock [80]. The 2008 assessment considered the likelihood of criticalities in a range of possible geological environments. Criticality calculations were undertaken to identify critical systems under GDF conditions and to evaluate whether they would lead to criticalities. Transient models of criticality were also applied to calculate the magnitude of the effects of different types and sizes of criticality.

The Criticality status report [21] conclude that the effects on post-closure safety of even the largest criticalities considered would be limited for the following reasons:

- direct radiation from a criticality event would be shielded by the surrounding rocks and materials;
- even if criticality events do occur, they are likely to affect only a limited part of a GDF;
- criticality events involving large amounts of fissile material might have an impact on the near-field environment, but these events are unlikely and can only occur a long time after closure, when the inventory will have decayed significantly; and
- the backfill and geological environment will still act to isolate the radioactive waste from the surface environment even in the event of a criticality.

In summary, our work to date on post-closure criticality safety for ILW shows that, in the unlikely event of a post-closure criticality occurring, the consequences would be limited, affecting only a small part of a GDF. Even if some local barriers to the release of radioactive material were impaired, the major part of the geological barrier would remain intact and would continue to isolate and contain the wastes.

Criticality safety for separated plutonium and HEU

As noted above, of particular relevance to criticality safety would be the presence of significant quantities of separated plutonium (~100 tonnes) and, to a much lesser extent, HEU (~1 tonne) [12], if the current stocks of these materials are declared as a waste and disposed of in a GDF. These materials contain an order of magnitude more fissile material than the ILW in the Baseline Inventory. If these materials need to be managed in a GDF, we would determine conditioning and packaging requirements and emplacement strategies that would ensure that post-closure criticality involving these materials is not a significant concern. Conditioning and packaging requirements of separated plutonium and HEU would be an important part of the waste acceptance criteria [14] that would need to be established for these materials should they be disposed of as wastes.

As noted in Section 3.2.6.1, we understand the criticality safety issues, and consider that although the total amount of fissile material would increase significantly, the likelihood of a large accumulation may be vanishingly small because of the emplacement strategy [21]. As for ILW, the main radionuclides in separated plutonium and HEU of concern, from the viewpoint of criticality safety, are plutonium-239 and uranium-235.

- For separated plutonium, the main concern is the potential for criticality associated with the presence of plutonium-239 in the waste. Plutonium-239 decays to uranium-235, and there is no process operating in a GDF that could separate uranium-235 from uranium-238. Therefore, the mixing of plutonium-239 with criticality poisons and/or dilution in the waste package with uranium-238 (and possibly other radionuclides) would significantly reduce the potential for criticality.

- For HEU, the main concern is the potential for criticality associated with the presence of uranium-235 in the waste. The mixing of uranium-235 with criticality poisons and/or uranium-238 in the waste package (termed ‘downblending’) would significantly reduce the potential for criticality.

As discussed in Section 4.1.2, the form and packaging of separated plutonium and HEU is still under consideration. Work on variant wasteforms, including consideration of criticality safety, is being undertaken by the NDA as part of its wider R&D work on the management of nuclear materials in the Baseline Inventory [135], and we have begun to consider them in our disposability assessment process. For example, one possibility that has been investigated in some detail is their immobilisation in a ceramic matrix that would be stable for long times and would only slowly release fissile material. Use of a high-integrity container would further support long-term containment of these materials.

5.2.2.5 Upper inventory

As discussed in Section 4.1.1, the upper inventory assumes that all SF from currently operating Advance Gas-cooled Reactors (AGR) and from the Sizewell-B Pressurised Water Reactor (PWR) is reprocessed, thereby maximising the volume from legacy facilities of higher activity radioactive waste requiring disposal. The effect of this assumption is to decrease the volume of SF from legacy facilities requiring disposal, and increase the volumes of ILW and HLW. The wastes to be managed would have similar characteristics and total radioactivity to those we are already considering, but the distribution of the radioactivity between waste types and the size of the required disposal areas for ILW, HLW and SF would be different in this scenario. Overall we do not consider there would be any challenges to the disposability of these components of the upper inventory.

The upper inventory also includes ILW and SF from a proposed new generation of nuclear power stations; we assume that new build SF is not reprocessed. Assuming a fleet of six to nine new stations (depending on the design(s) built) significantly increases the amount of SF requiring management, compared to the Baseline Inventory, and leads to a small increase in the amount of ILW. The result is that the disposal area required for ILW would increase by about 1% of the area required for legacy ILW, per reactor, or less than 10% for the fleet of reactors. However, the additional area required for disposal of SF would be 6-8% of that required for legacy HLW and SF, per reactor, or about 50-55% for the fleet of reactors.

We have recently undertaken initial disposability assessments for the ILW and SF expected to arise from the operation of new nuclear power stations [133, 134]. This assessment has been based on information on the nature of operational and decommissioning ILW, and SF, and proposals for the packaging of these wastes, provided to us by the potential reactor suppliers. We have used this information to assess the implications of the disposal of the proposed ILW and SF waste packages against the generic waste package specifications in place for existing ILW and HLW/SF [78, 79]. We considered the safety of transport operations, waste handling and emplacement at a GDF, and the long-term performance of a GDF, together with the implications for the size and design of a GDF. We concluded that ILW and SF from operation and decommissioning of new build reactors should be compatible with plans for transport and geological disposal of the Baseline Inventory. Compared with existing ILW and SF, no new issues arise that challenge the disposability of the ILW and SF from the operation of a new fleet of nuclear power stations in the UK. This conclusion is supported by the similarity of the ILW and SF from the proposed new build power stations to that arising from the existing PWR at Sizewell B.

One technical issue that would need to be addressed in design and operation of a GDF is that new reactors would have a maximum fuel assembly radioactivity (“burn up”) after 60 years of operation greater than that of legacy SF. This means that the new build SF would be more radioactive and thermally hotter than legacy SF. Our assessments conservatively assume that all SF from new build reactors would achieve the maximum average burn-up of 65 giga-Watt days per tonne uranium. In practice, this value will represent the maximum of a range of burn-up values for individual fuel assemblies. Our post-closure assessments considered such issues as the implication of hotter wastes on the instant release fraction (see Section 5.2.1.2), and on any buffer used as an engineered barrier around the waste package.

In summary, our qualitative assessment of the potential impacts of disposing of the upper inventory in a GDF indicate that there are no fundamental reasons why separate disposal facilities would be needed for the different types of higher activity radioactive wastes in this inventory. However, the final decision would need to be made in the light of the latest technical and scientific information concerning the inventory of wastes we will actually be required to manage in a GDF, international best practice, and site-specific environmental, safety and security assessments at candidate sites.

5.3 Register of significant uncertainties

The environment agencies require us in the GRA (paragraph 7.3.10) [1] to demonstrate that the ESC takes adequate account of uncertainties by establishing and maintaining a "register of significant uncertainties", and a clear forward strategy for managing each significant uncertainty based on avoidance, mitigation, reduction and/or quantification of the uncertainty.

The ESC is currently at an early stage of development, because the site and design have not yet been chosen. However, our work builds on more than 30 years of site-specific and generic experience studying geological disposal and undertaking safety assessments in the UK, as well as learning from more than 40 years of such experience in other countries. Therefore, although we are at the generic stage, we have a high degree of confidence in our ability to design, build, and operate and close a GDF for which a strong safety case can be made, providing a suitable site comes forward through the MRWS Site Selection Process. These studies also give us a head start on identifying where significant uncertainties are likely to be found as the MRWS Site Selection Process moves forward.

At this generic stage in the programme, our most significant uncertainties are programmatic:

1. We do not yet have a candidate site(s) to consider, and therefore we do not know the geological environment in which we will be developing a GDF or the disposal concept we will be implementing.

   We have dealt with this uncertainty in the ESC by considering a wide range of geological environments and illustrative geological disposal concept examples in the ESC. Once candidate sites become available through the MRWS Site Selection Process and site investigation proceeds, this uncertainty will be much reduced.

2. We do not know the inventory we will actually be disposing of in a GDF. We are currently planning a GDF with sufficient capacity to accommodate the upper inventory. However, some of the materials defined in this inventory have not yet been declared as waste, and our assumptions on waste volumes from a proposed new generation of nuclear power stations is predicated on these reactors actually being built and operated for their design lifetime. The inventory has a major impact on the volume of rock required for a GDF (i.e. a site selection consideration) and on the choice of disposal concept and EBS.

   We have dealt with this uncertainty in the ESC by considering the upper inventory, including wastes from a possible new programme of nuclear power plant construction, and by considering a range of illustrative geological disposal concept examples. We will continue to consider the impacts on GDF design and the ESC of variant inventory scenarios as the MRWS Site Selection Process moves forward.

3. We do not know the wasteform for some of the key waste streams that have been declared as waste (e.g. graphite).

   We have dealt with this uncertainty in the ESC by considering a wide range of parameter values in our generic post-closure assessment calculations potentially representative of the impacts of a wide range of wasteforms. However, these calculations are only illustrative, and we will continue to use our disposability assessment process to assess individual waste packaging proposals.

4. We do not yet know candidate communities' views on retrievability. Maintaining the option of retrievability potentially has an impact on GDF design, on costs and on confidence in the safety case [4].

   We have discussed this uncertainty in the ESC, and are taking steps to address it, but we note that the potential impacts are strongly linked to the type of geological environment in the candidate community/ies. We will consider the potential technical, safety and financial impacts of maintaining an option for retrievability once we have more information on the available geological environments and preliminary views of the candidate community/ies.

5. There are several issues where we would value dialogue with regulators, prior to defining approaches and expending significant effort. These issues include the approaches for treating non-radiological hazards and evaluation of impacts on non-human biota (see Sections 3.2.4 and 3.2.5).

   We expect this regulatory dialogue to inform the next update to the ESC. However, we do not consider that any of these issues are likely to impact the main conclusions of this generic ESC.

As we move forward through the MRWS Site Selection Process, our key uncertainties will become more focused on site-specific and concept-specific scientific and technical issues. Significantly more detail on this kind of uncertainty can be provided when we have specific sites and geological disposal concepts to consider. We can then develop a site-specific register of key technical uncertainties – at a more detailed level – that we keep under review as our information base increases. In the meantime, our research status reports and our R&D programme provide an indication of the kinds of more detailed uncertainties we expect to be facing for particular types of geological environment and disposal concept. We cannot set out how we would manage such uncertainties until we have specific sites to consider and have understood the importance of the uncertainties with respect to the ESC. However, the general methods we intend to use for the management of scientific and technical uncertainties in our assessments are set out in Section 3.2.2.3. Our forward programme of R&D, outlined in Section 6.3, identifies the main topic areas we are intending to address.

Similarly, as decisions are made on whether particular nuclear materials are classified as wastes, and on waste conditioning, we will be able to provide further detail on what we regard as key uncertainties in the ESC.

Finally, as identified in points 4 and 5 above, we have already identified several areas of uncertainty that can only be resolved via discussion with the potential host community(ies) and the environmental regulators.

6 Synthesis and forward programme

This section contains a synthesis of our main safety arguments with respect to the environmental safety principles contained in the GRA (Section 6.1), a review of the extent to which we have met the objectives of this generic ESC (Section 6.2), and a summary of our forward programme (Section 6.3).

6.1 Synthesis

Principle 1: Level of protection against radiological hazards at the time of disposal and in the future

Solid radioactive waste shall be disposed of in such a way that the level of protection provided to people and the environment against the radiological hazards of the waste both at the time of disposal and in the future is consistent with the national standard at the time of disposal.

Principle 2: Optimisation (as low as reasonably achievable)

Solid radioactive waste shall be disposed of in such a way that the radiological risks to individual members of the public and the population as a whole shall be as low as reasonably achievable under the circumstances prevailing at the time of disposal, taking into account economic and societal factors and the need to manage radiological risks to other living organisms and any non-radiological hazards.

Principle 3: Level of protection against non-radiological hazards at the time of disposal and in the future

Solid radioactive waste shall be disposed of in such a way that the level of protection provided to people and the environment against any non-radiological hazards of the waste both at the time of disposal and in the future is consistent with that provided by the national standard at the time of disposal for wastes that present a non-radiological but not a radiological hazard.

Principle 4: Reliance on human action

Solid radioactive waste shall be disposed of in such a way that unreasonable reliance on human action to protect the public and the environment against radiological and any non-radiological hazards is avoided both at the time of disposal and in the future.

This section summarises the main arguments we have presented in this ESC to demonstrate why we believe that we can develop a GDF that can provide the required level of environmental safety for any or all higher activity radioactive wastes that may arise in the UK. It summarises our current position and will therefore evolve significantly for future versions of the ESC as we acquire site-specific information about the candidate site(s) and develop site-specific disposal concepts and designs.

6.1.1 What we have done already

Although we are currently at an early stage in the development of a GDF and do not yet have a candidate site or a site-specific disposal concept, the knowledge base we have now is sufficient to progress from the generic stage to desk-based studies of candidate sites (MRWS Site Selection Process Stage 4, see Section 2.1). We have a great deal of experience and knowledge of the processes that are likely to occur in a geological disposal system and the features that will ensure environmental safety. This is based on work carried out by ourselves and previously by Nirex, as well as work carried out by overseas waste management organisations, collaborative international projects, international bodies such as the EC, the NEA and the IAEA, and experience gained in other related industries such as mining. We also have an active R&D programme [86] and a developing site characterisation strategy [85].

Our work to date has to a large extent involved ensuring that we have in place the strategies, tools and competence we will require to design a GDF and build confidence in its performance. These must all be in place before we commence the desk-based studies at candidate sites within the MRWS Site Selection Process. They are described in Section 3 of this generic ESC. This generic ESC is one part of the process of developing these strategies, tools and competence, and obtaining feedback from the regulators and stakeholders.

Although we do not yet have a candidate site, it is important to indicate why we have confidence that we would be able to demonstrate the required level of environmental safety for a wide range of potential geological environments and disposal concepts that we may have to consider in the future. Therefore, we have considered in this generic ESC a set of illustrative geological disposal concept examples that between them illustrate a range of issues that we currently anticipate we may need to address during the development of a GDF. The site-specific ESC we would eventually make for a candidate site is likely to draw to differing extents on safety arguments presented in this generic ESC for one or more of these illustrative examples.

6.1.2 Why we have confidence in the environmental safety of a GDF

Guidance on geological disposal has been developed over many years through international organisations such as the EC, the NEA, and the IAEA. Most recently, in 2008 the NEA Radioactive Waste Management Committee issued a Collective Statement on geological disposal that notes the following [169]:

> *"The overwhelming scientific consensus worldwide is that geological disposal is technically feasible. This is supported by extensive experimental data accumulated for different geological formations and engineered materials from surface investigations, underground research facilities and demonstration equipment and facilities; by the current state of the art in modelling techniques; by the experience in operating underground repositories for other classes of waste; and by the advances in best practice for performing safety assessments of potential disposal systems.*
>
> *"Disposal can be accommodated in a broad range of geological settings, as long as these settings are carefully selected and matched with appropriate facility design and configuration and engineered barriers."*

There are more than 40 years of international precedent in designing GDFs and demonstrating environmental safety for wastes similar to those in the UK for the range of geological environments we may need to consider. This experience, combined with existing UK experience, gives us a high degree of confidence in our ability to design, construct, operate and close a GDF that meets all environmental safety requirements.

The UK Government's 2008 policy statement that geological disposal is the best approach for long-term management of the UK's higher activity radioactive waste [4] derives from extensive reviews and assessments of the available options conducted by the Committee on Radioactive Waste Management (CoRWM), an independent body established by the UK Government to advise on policy [170]. The work by CoRWM helped to build wider confidence in the safety of geological disposal.

This previous work illustrates how geological disposal can be implemented safely in many different kinds of geological environment and for many different types of radioactive waste. 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 the multiple barriers that would be present within a geological disposal system can work together to ensure safety for a wide range of geological environments. As described in Sections 4 and 5, in Appendices A, B and C, and in the Tier 2 supporting reports, we have a good understanding of the key features of the various barriers and the way they interact. We therefore have confidence that once we have a preferred site and disposal concept, we can develop an optimised design that meets all environmental safety requirements.

With regard to ***GRA Principle 1 (protection against radiological hazards)***:

- The geological environment will provide isolation from both surface influences (for example, climate change) and inadvertent human intrusion. The screening criteria applied at Stage 2 of the MRWS Site Selection Process help ensure that inherently unsuitable sites are not carried forward.
- Many of the existing and proposed wasteforms are intrinsically stable and provide a considerable degree of physical containment. Our disposability assessment process, which will evolve into waste acceptance criteria, ensures that wastes are packaged in a way that makes them 'disposable'.
- We understand the properties of the various materials that we would use in the EBS and how they provide containment and retard contaminant migration. We therefore have confidence that we can design a sufficiently robust EBS to complement the properties of the host rock and the wasteforms.
- The geological environment will provide a significant degree of containment and retardation - the geological environment will ensure that radionuclide travel time from a GDF to the surface is sufficiently long and dilution/dispersion sufficiently great that radionuclides will not reach the accessible environment in concentrations that are unacceptable.
- Our qualitative environmental safety arguments and illustrative generic assessment calculations indicate that a GDF can be developed safely in a wide range of potential geological environments.

With regard to ***GRA Principle 2 (optimisation)***, we have already put in place key strategies and systems we will require to develop and optimise GDF design. Perhaps the most critical of these strategies is our management strategy (Section 3.3), which is designed to ensure that we can deliver our design and assessment strategies in a coherent and integrated way that will ensure an appropriate level of quality and accountability over the very long timescales associated with GDF implementation and operation.

With regard to ***GRA Principle 3 (protection against non-radiological hazards)***, we consider that a GDF designed to provide the required level of radiological performance would also provide a sufficient level of protection for non-radiological hazards present in a GDF. The multiple barrier system present in a GDF would be significantly more robust than that normally used in engineered facilities for the disposal of hazardous wastes.

With regard to ***GRA Principle 4 (reliance on human actions)***, we would design a GDF such that safety in the long term would not require any reliance on ongoing monitoring or other human action. During the operation of a GDF, there would be some need for human intervention to ensure safety, for example to monitor discharges and to ensure standard underground operating systems (e.g. ventilation systems) continue to function. However, we would rely as much as possible on passive safety features, such as the stability of the wasteforms and their engineered containment.

6.2 Meeting generic ESC objectives

The main objectives of this generic ESC were identified in Section 1.1, and we summarise below how they have been met:

- *To set out our understanding of the requirements of an ESC, explaining how we will use the ESC at various hold points in the implementation process for a GDF.*

Our approach to progressive development of an ESC has been set out in Section 2, and the structure of this document and Figure 1.2 illustrate the overall structure of an ESC.

- *To explain our safety strategy for a GDF and the way in which we will build confidence in environmental safety through a range of qualitative and quantitative lines of reasoning.*

Our design and siting strategy, assessment strategy and management strategy for a GDF have been set out in Section 3. Appendices A, B and C contain more specific design strategies relevant to particular potential geological environments and disposal concepts.

- *To provide arguments on the environmental safety of a GDF with reference to the principles and top-level requirements of the GRA and, consistent with being at a generic stage, show that safety could be provided by a combination of engineered and natural barriers in different geological environments and illustrate how a GDF could be implemented in these environments.*

Our safety arguments are summarised in Section 5 and in Section 6.1 and in the Tier 2 safety assessments and research status reports, as far as we can develop them at this generic stage. Appendices A, B and C provide arguments specific to particular potential geological environments and disposal concepts. Our Generic disposal facility designs report [24] illustrates how the illustrative geological disposal concept examples could be implemented in the UK.

- *To provide a continuing basis for our assessment of waste packaging proposals and the issue of Letters of Compliance to UK waste producers.*

Section 3.1.4 discusses our disposability assessment process, and how that will eventually lead to the development of waste acceptance criteria. Our quantitative safety assessment methodology for this generic ESC, summarised in Section 4.3, and our assessment results summarised in Section 5 and described in more detail in the Tier 2 safety assessment reports, provide an underpinning for disposability assessments that we will need to make until the ESC is next updated. The assessment methodology used here can be applied to new waste packaging proposals until we have site-specific and concept-specific information on which to base such assessments.

- *To help provide an appropriate basis for undertaking assessments of candidate sites as part of the MRWS Site Selection Process.*

Section 2 discusses the role of the ESC in the MRWS Site Selection Process, and how the ESC will be updated. Section 3 highlights aspects of the safety strategy that will become more relevant as we move forward to desk-based studies and beyond. Sections 4-5 and Appendices A, B and C illustrate the kinds of information we will be assembling and safety arguments we might be presenting concerning candidate sites. However, we note that there is a wider framework for assessment of candidate sites that extends well beyond the DSSC. In fact, safety is only one aspect of decision-making on site selection. From this perspective, the purpose of the DSSC is to illustrate how we could make a safety case at candidate sites having different geological environments, and to understand the features of particular environments and disposal concepts that would need to be investigated.

- *To identify the R&D work needed to provide relevant evidence and develop confidence in the qualitative and quantitative environmental safety arguments presented in future updates of the ESC.*

We set out key uncertainties in Section 5.3, and our view of the further R&D needed at this stage of the programme in Section 6.3 (follows below). Our R&D programme draws on work reported in our 2009 R&D strategy [86], the research status reports [e.g. 15, 16, 17, 18, 19, 20, 21], and other project inputs. We are also mindful of Nirex's 2005 Viability

Assessment [99] and the Environment Agency's review of that report [35], noting that Nirex work was focused on a GDF in higher strength rock only and did not consider all of the radioactive materials that we need to consider (see Section 2.3 and Appendix F). Since we are at the generic stage, safety assessment calculations only provide a broad-brush tool for prioritisation of R&D. In future site-specific updates of the ESC, we will describe in more detail how sensitivity analyses have been used to investigate the importance of outstanding uncertainties, and to help in prioritisation of site characterisation, R&D and design work.

- *To help demonstrate that we are developing the capability to perform the functions of a Site Licence Company in due course.*

We consider that we have in place the main elements of a management strategy that will ensure we can deliver a GDF that meets all applicable requirements. This management strategy is summarised in Section 3.3, and we recognise it to be a key part of our overall safety strategy. However, we agree with the regulators that further development of our management strategy is needed to meet principles we have agreed with regulators concerning our governance, structure and operation [119].

6.3 Forward programme to support updates of the ESC

As noted above, our R&D strategy is presented in [86] and details are described in the relevant research status reports. Our R&D, design and site characterisation programmes are designed to provide the information we require to identify preferred disposal concepts, and subsequently to optimise the design of a GDF and build the required level of confidence in its performance. The Tier 2 supporting reports form part of this ESC, and in them we identify gaps in our current knowledge and present a way forward to address these future information needs.

As discussed in Section 3.1.5.2, our R&D strategy [86] identifies six key areas in which we currently plan to carry out work to fill gaps in our knowledge and to build confidence in the implementation of geological disposal at candidate sites:

1. develop and expand our HLW and SF R&D programme;
2. support the development of future management strategies for materials such as separated plutonium and uranium that are considered in the MRWS White Paper, but are not currently declared as wastes;
3. continue R&D into ILW disposal;
4. address GDF implementation issues;
5. prepare for site characterisation; and
6. investigate the social aspects of implementing geological disposal.

We consider the work being done in each of these areas in more detail in this Section of the ESC.

1. Develop and expand our HLW and SF R&D programme

We intend to develop and expand our R&D programme for HLW/SF, drawing on international expertise and commissioning our own R&D to fill knowledge gaps. So far we have taken the following steps:

- We have issued a call for proposals for experimental work on the HLW/SF topics.
- We have restructured our R&D programme to provide an increased management focus on concepts for HLW and SF.

2. Support the development of future management strategies for materials such as separated plutonium and uranium

We are continuing to investigate the disposability of such materials. This includes R&D focused on the evaluation of proposed wasteforms. This work is being integrated with wider NDA R&D programmes [e.g. 135].

3. Continue R&D into ILW disposal

There is 30 years experience in the UK of R&D into ILW disposal and we have a good understanding of the associated technical issues, documented in a number of reports (e.g. the Tier 2 supporting reports that form part of this ESC). However, additional work is still required, and we intend to carry out R&D focused on remaining issues concerning the viability of ILW disposal, including:

- undertaking R&D on generic issues to support the disposability assessment process;
- investigating the potential application of new and emerging technologies, such as the use of polymer encapsulants for some ILW; and
- building further confidence in the viability of a cementitious disposal concept by commissioning long-term experiments to demonstrate understanding of key processes that govern long-term behaviour of the near field in such a concept.

4. Address GDF implementation issues

In the preparatory phase of our programme, before we have specific sites to consider under the MRWS Site Selection Process, we will continue to address generic issues of GDF implementation. In particular, the UK Government's 2008 MRWS White Paper [4] highlights three implementation issues:

- evaluation of the implications of disposing of all higher activity radioactive wastes in a single GDF;
- the possibility of an extended period where retrievability of wastes may be required; and
- evaluation of the implications of disposing of higher activity radioactive wastes from a proposed new generation of nuclear power stations in a single GDF.

Investigation of the expected additional impacts arising from a single facility containing both ILW/LLW and HLW/SF disposal areas has highlighted the possible detrimental impacts were alkaline water derived from an ILW vault able to enter the disposal area for HLW or SF [71, 72]. We are confident that we can develop disposal concepts and GDF designs that will minimise the likelihood and consequences of any such interactions.

The possibility of an extended period where retrievability may be required places additional importance on the longevity of the waste container, on vault and tunnel stability, and on the maintenance of operating infrastructure, to ensure that wastes remain retrievable. We have some confidence from extrapolation of the data from long-term experiments on localised corrosion of stainless steel, as currently used for ILW packages, that such packages would still be intact and capable of being handled for on the order of 1,000 years.[22] However, we are commissioning mechanistic studies aimed at increasing our understanding of localised corrosion processes to develop better predictive tools and to build confidence in the extrapolation of experimental data. We also intend to further

[22] The disposal container material for HLW and SF is yet to be selected, but we would expect such containers to be more robust than ILW containers and to remain intact for considerably longer than 1,000 years [15].

investigate the operational issues associated with stabilising vaults and tunnels and maintaining operating infrastructure for long periods.

The UK Government considers that it would be technically possible, and desirable, to dispose of any waste from new nuclear power stations in the same GDF as waste from legacy nuclear facilities [4]. We are considering disposability issues associated with higher activity radioactive wastes that would be generated during the operation and decommissioning of a proposed new generation of nuclear power stations as part of the Generic Design Assessment process [171]. The Generic Design Assessment process is the vehicle by which the nuclear regulators are assessing the safety, security, and environmental impact of power station designs, including the quantities and types of waste that are likely to arise, and the ability to store and dispose of solid wastes [172]. We are involved in this work specifically to consider disposability of wastes being proposed at an early stage. Our work to date on this issue is summarised in [133, 134]. We will undertake further disposability assessment work as required as part of the Generic Design Assessment process.

Other implementation issues that we are able to consider at this early generic stage include:

- the design of waste packages, in particular the options for container materials for HLW and SF (for example, copper or steel);
- the design of waste transport containers to transport waste packages to a GDF;
- the development of methods for inspection and monitoring of waste once disposed of in a GDF, through involvement in the collaborative EC MoDeRn Project [89], which will make use of underground research laboratories in other countries (e.g., Grimsel in Switzerland);
- the development of methods for the backfilling, sealing and closure of a GDF; and
- the development of approaches to selection of preferred disposal concepts and design options [e.g. 68].

5. Prepare for site characterisation

Our plans for characterising a site are well developed and have been documented [e.g. 85]. When we characterise a site, we will use best practice, drawing on the extensive international experience available from other radioactive waste management organisations and from other industries. We have commissioned studies to consider R&D needs associated with various aspects of site characterisation [e.g. 85]. We consider that there are established methods available for almost all site characterisation data needs. Any development need for site characterisation tools will be site specific and will be factored in to our programme once specific candidate sites become available in the MRWS Site Selection Process.

6. Undertake social science research

Social science research is important to support effective, sustained engagement with stakeholders, particularly with local communities during the siting process. Social science research can also help us plan and implement effective strategies to engage with the public, which will help to build confidence in our capability to deliver a safe long-term solution for managing radioactive waste.

Some national radioactive waste management organisations, such as the Swedish Nuclear Fuel and Waste Management Company (SKB), have undertaken significant social science research programmes [173]. We will draw on the experience of such research programmes, wherever possible.

We have also participated in a range of international projects that include social science research (e.g. EC projects MoDeRn [89], PAMINA [101], and COWAM in Practice [174]). These international projects help national waste management organisations to share experience.

We will undertake additional social science research, if required, to improve our approaches for a UK-specific context [46]. For example, such research has been conducted recently within the context of the EC PAMINA Project [45, 101].

Moving forward

In moving forward with the six R&D areas discussed above, we need to further divide our programme to help with planning. We must make sure that our R&D framework is comprehensive so that we can plan for everything we are going to need. Our R&D topics and the work associated with each topic can be found in our research status reports. Many of the topics identified are directly relevant to the ESC; others are particularly relevant to other aspects of our programme. The topics directly relevant to the ESC include those related to barrier performance, modes of radionuclide release and transport, radionuclide behaviour, GDF design, site assessment, and developing the ESC:

- wasteform;
- container;
- buffer / backfill;
- geosphere;
- biosphere;
- contaminant / radionuclide behaviour during transport by groundwater, including any effects of non-aqueous phase liquids;
- contaminant / radionuclide release as a gas and transport of gases;
- site assessment methodology;
- design development; and
- safety case methodology, including the development of assessment methods and models specific to the candidate site(s) that become available through the MRWS Site Selection Process, and further use of safety functions to identify disposal system requirements as captured in the disposal system specification, to demonstrate the links between performance assessment results, geological disposal concept selection, design development, and site characterisation.

In addition, we have research topics that are focusing on the approach to stakeholder dialogue and the social and ethical aspects of geological disposal (social sciences research), and on conducting the required Strategic Environmental Assessments and Environmental Impact Assessments (see Section 2.5 and Appendix D).

Our R&D work will be scheduled to fit in with the development of updates of the ESC and the wider MRWS Site Selection Process for development of a GDF.

As we move forward through the MRWS Site Selection Process, our forward programme will also be influenced by dialogue we have on the ESC with regulators, the Community Siting Partnership(s), and other interested stakeholders (see Sections 2.4 and 2.5). We will continue to produce a hierarchy of documents to enable stakeholders to engage with our work at different levels of technical detail. We will actively engage with Community Siting Partnership(s) when they are formed, and will be proactive in seeking opportunities to engage with a wide range of stakeholders to enable them to understand and influence our work programme.

Appendix A Illustrative geological disposal concept examples for higher strength host rock as applied to the UK

A1 Introduction

This appendix illustrates the case of a GDF constructed within a geological environment characterised by a higher strength host rock (see Table 4.2 of the main report and the Geosphere status report [17] for a summary of the characteristics of this rock type). Such a host rock would typically comprise a crystalline igneous or metamorphic rock or a geologically older sedimentary rock where any fluid movement is predominantly through discontinuities (mostly fractures). By "higher strength" we mean rocks in which it would be possible to excavate stable tunnels and disposal vaults requiring only limited rock support. One property of higher strength rocks is that they tend to be fractured, with the degree and orientation of the fracturing being important in determining their geotechnical and hydrogeological characteristics. The higher strength host rock may extend all the way to the ground surface or it may be overlain by a sedimentary sequence.

As explained in the main report, the UK Government sees no case for having more than one GDF if one facility can be developed to provide suitable, safe containment for the entire inventory. We are therefore describing in this appendix a conceptual layout for a GDF in a higher strength rock that is based on two distinct disposal areas implementing different disposal concepts, one for unshielded intermediate-level waste (UILW), shielded intermediate-level waste (SILW), low-level waste (LLW) and depleted, natural and low enriched uranium (DNLEU), should it be declared as waste, and the other for high-level waste (HLW), and spent fuel (SF), highly enriched uranium (HEU) and plutonium (Pu), should these be declared as wastes. These disposal areas would share common access ways and surface facilities, but would be separated from each other to ensure that the impact of interactions between the two disposal areas would be sufficiently small that it does not compromise the overall performance of a GDF.

The assumptions that underpin the generic DSSC are set out in [32]. As discussed in [32] and shown in Table 4.3 of the main report, we are basing our generic designs for a higher strength host rock on the UK Concept for ILW/LLW and on the Swedish KBS-3V concept for SF. Should we find ourselves considering a candidate site with a higher strength host rock, we would develop site-specific concepts, which may, or may not, be based on these disposal concepts.

In this appendix, where we refer to the ILW/LLW disposal area or vaults, we mean the part of a GDF in which a disposal concept based on Nirex's Phased Geological Repository Concept is implemented, that is the disposal area for UILW, SILW, LLW and DNLEU. When we refer to the HLW/SF disposal area or tunnels, we mean the part of a GDF in which a disposal concept based on the Swedish KBS-3V disposal concept is implemented, that is the disposal area for HLW, SF, HEU and Pu. However, discussion focuses on ILW/LLW and HLW/SF because, to date, this is where our conceptual design and assessment work has focused.

One of the purposes of this appendix is to illustrate how the different components of our work programme fit together to provide knowledge about the environmental safety of a GDF. It therefore draws together information about the characteristics and likely evolution of a GDF developed in a higher strength host rock from a number of sources, most notably the Package evolution [15], Near-field evolution [16], Geosphere [17], Gas [19] and Radionuclide behaviour [20] status reports, which describe our current state of knowledge and understanding. These status reports provide details about the characteristics, safety functions and expected evolution of the different barriers in the multi-barrier system.

A2 Geological environment

A higher strength host rock could extend almost to the ground surface or could be overlain by sedimentary rocks. Overlying sedimentary rocks could be sufficiently permeable that solute transport might be dominated by advection, or they could have a very low permeability such that solute transport is dominated by diffusion, or the sequence could contain both types of strata. Whatever the overlying sequence, there is likely to be a near-surface zone of weathered rock and recent (Quaternary) deposits.

The physical properties of the higher strength host rock mean that it is likely to be faulted and fractured on a range of length scales from metres (or less) to kilometres. Groundwater flow would be predominantly through the subset of these discontinuities that are sufficiently open and connected to support flow. The unfractured matrix between the fractures has a low intrinsic permeability[23]. Therefore, on the length scale of the spacing of the discontinuities, the rock may be very heterogeneous. However, on length scales of many times the spacing of the discontinuities (probably at least several tens of metres), a homogeneous porous medium may provide an acceptable ‘upscaled’ representation of the hydrogeological behaviour of a higher strength host rock. The Geosphere and Gas status reports [17, 19] describe the properties of higher strength host rocks in more detail, and the Geosphere status report [17] discusses the concepts of upscaling and effective representations of heterogeneous media.

Sorption in a higher strength rock is likely to be dominated by the assemblages of minerals that line the discontinuity surfaces and the mineralogy of the rock immediately adjacent to the discontinuities. Most of the groundwater would flow through the subset of discontinuities that are open sufficiently open and connected to support flow, and its interaction with the rock is likely to have resulted in the precipitation of new minerals on the fracture surfaces and the alteration (precipitation and dissolution) of the rock immediately adjacent to the discontinuities. Thus, the bulk mineralogy of the rock may not be representative of the assemblage of minerals that dominates the sorption characteristics with regard to the transport of radionuclides by groundwater.

We expect that the engineering properties of the higher strength host rock at disposal depth would be reasonably predictable. By definition, it would be possible to construct large-span disposal vaults (20 metres span or more) that are intrinsically stable and require only limited rock reinforcement at typical disposal depths of 400 to 700 m, and it should be possible to construct such large-span vaults at depths of up to 1,000 metres if appropriate engineering support is provided [139]. The layout may be strongly influenced by the requirement to avoid intersecting areas that have been identified as being highly fractured (large faults), which may be zones of increased transmissivity and reduced engineering quality.

If present, the overlying sedimentary sequence would be characterised by rocks that are likely to be of lower strength than the host rock, although, in absolute strength terms, some of them might be classified as higher strength rocks. The overlying rocks would be likely to contain fewer fractures than the host rock, although they may be affected by some of the same major faults as the host rock. Groundwater flow in the sedimentary sequence is likely to be predominantly through the matrix (i.e. porous medium flow), although discontinuities may be important in some strata. The bedded nature of the sedimentary rocks means that the permeability parallel to bedding may be significantly greater than the permeability perpendicular to bedding [17]. The composition of the overlying rocks would be important in determining both their permeability and their sorption

[23] Intrinsic permeability is a property of the permeable medium. The more commonly quoted hydraulic conductivity depends on the density, viscosity and temperature of the fluid as well as the permeability. Further details can be found in [17].

properties. Clay-rich rocks tend to have lower permeabilities than rocks containing only small amounts of clay (for example limestones or ‘clean’ sandstones), and many radionuclides, for example actinides, sorb strongly to clay minerals. Iron oxides, which are often found in the natural cements that bind sedimentary rocks together, are also important substrates for sorption. Further information may be found in the Radionuclide behaviour status report [20].

The composition of the groundwater in the host rock depends strongly on the exact setting and the composition of the overlying sedimentary rocks, which in the UK could include evaporites. Most groundwaters found at disposal depths are brackish or saline as a result of water-rock interactions, and higher strength rocks that are overlain by sequences containing evaporites are likely to contain brines. If the cover sequence does not contain evaporites, the host rock groundwater may have a low to moderate salinity (total dissolved solids in the range 10,000 to 40,000 milligrams per litre) [16, 17].

The groundwater flow rate would depend on the permeability of the rock and the hydraulic gradient driving flow. This gradient could potentially be significant in this example because the strength of both the host rock and the overlying rocks means that there may be appreciable local or regional topography (hills/valleys, with elevation differences of many hundreds of metres), depending, of course, on the geographical location. If the water table is close to the ground surface, which is likely in the UK, and the overlying rocks are relatively permeable, the gradient may be transmitted to disposal depths and advection is likely to dominate solute transport. However, solute transport may be dominated by diffusion in any low-permeability strata in the overlying sequence, and the presence of such strata may isolate the host rock from the near-surface hydraulic gradients. Rock matrix diffusion may also be an important retardation process in the host rock. Solute transport processes within the geosphere are discussed in more detail in the Radionuclide behaviour status report [20].

We expect any gas generated within the disposal areas to be able to enter the higher strength host rock relatively easily. Therefore, assuming that the gas is able to escape through the engineered barriers, we do not expect a build-up of gas pressure in the disposal areas to pose a threat to the integrity of the disposal system. Any low-permeability strata in the overlying sedimentary sequence may trap any free gas that is released from the disposal areas. The migration of gas through a higher strength host rock and an overlying sequence is described in more detail in the Gas status report [19].

We expect that natural conditions in the host rock at disposal depths would evolve only slowly on timescales of tens of thousands of years. Indeed, conditions at disposal depths may still be evolving, albeit very slowly, from the effects of the last major glaciation more than ten thousand years ago. The magnitude of the impact at disposal depths of large-scale environmental change, such as would be associated with glaciation, would depend on the properties of the host rock and of the overlying sedimentary sequence, if present. If the host rock extends almost to the ground surface or if there are very few low-permeability strata in the overlying sedimentary sequence, events such as glaciation have the potential to result in changes in groundwater chemistry and flow rates and direction of sufficient magnitude that we might need to model them explicitly when considering the future evolution of the geosphere [17].

A3 Engineered barrier system

This section summarises the application of the illustrative geological disposal concept examples for a higher strength host rock presented in Table 4.3 to the UK, as set out in our Generic disposal facility designs report [24].

ILW/LLW

In the ILW/LLW disposal concept, the waste packages would be stacked in large span (16-metre by 16-metre) disposal vaults that are several hundred metres long and separated from each other by about 50 metres. During construction, the vaults would be accessible from both ends (i.e. they are not dead-end excavations), but the far end of each vault would be sealed before the start of waste emplacement. Emplacement of unshielded waste packages would be by overhead crane and emplacement of shielded waste packages would be by stacker truck, on a 'first in – last out' basis. A cementitious backfill (Nirex Reference Vault Backfill, NRVB [175]), which has a high porosity and a higher permeability than the waste packages, would be emplaced in the spaces between the waste packages as part of the closure engineering. The volume of backfill would be sufficient to maintain high-pH conditions in the vaults for many tens of thousands of years.

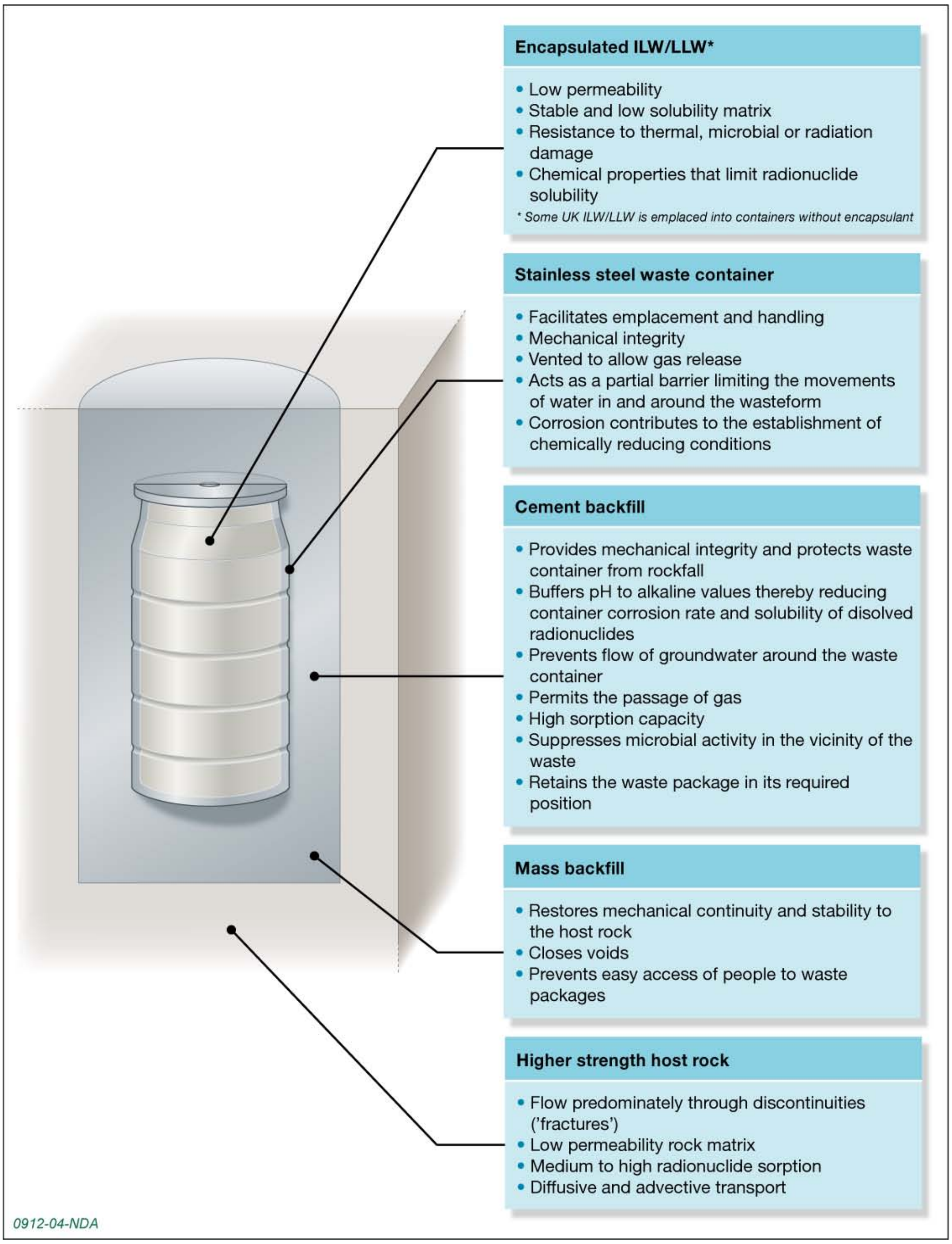
In this disposal concept, the cementitious wasteform provides an important post-closure barrier, initially by containing the contaminants and then by retarding their release into the near field. The backfill is designed to condition the groundwater chemistry and provide a long-term barrier to radionuclide release by limiting the solubility and promoting the sorption of many of the radionuclides that are of most concern during the post-closure period. Most of the ILW packages are vented to prevent the build-up of excessive gas pressures. We assume that the stainless steel waste containers would provide no containment during the post-closure period because, even at closure, the container is not fully 'intact'. The vent allows the release of gaseous radionuclides during transport and operations and the release of both gaseous and dissolved radionuclides after GDF closure, even if there is no corrosion or other degradation of the container. However, the waste containers do play an important role in containing and protecting the wasteform during the operational period and in limiting the release of short-lived radionuclides in the period immediately after the GDF is closed. The different engineered barriers and the processes and properties that contribute to them fulfilling their safety functions are illustrated in Figure A1.

HLW/SF

In the HLW/SF concept, the waste would be encapsulated in a copper canister with a cast iron insert to provide structural strength. These disposal canisters would be placed in vertical deposition holes in the floors of the disposal tunnels where they are surrounded by a compacted bentonite buffer. The disposal tunnels would then be backfilled with a crushed rock–bentonite mixture soon after disposal to provide sufficient confining pressure to allow the bentonite buffer to function correctly. The disposal tunnels are likely to have a diameter of about 5 metres and a spacing of about 25 metres. They are likely to be a few hundred metres long and to have dead ends. The vertical deposition holes in the tunnel floors would typically be about 5 metres deep (different waste packages have different lengths), 1.75 metres in diameter and spaced at 6.5 metres.

Figure A1 The barriers to radionuclide release for the ILW/LLW disposal concept in a higher strength host rock

The figure shows the barriers to radionuclide release from the ILW/LLW disposal area in a higher strength host rock and the processes and properties that contribute to their safety functions. The Near-field evolution status report [16], gives further details of the materials, processes and properties shown.

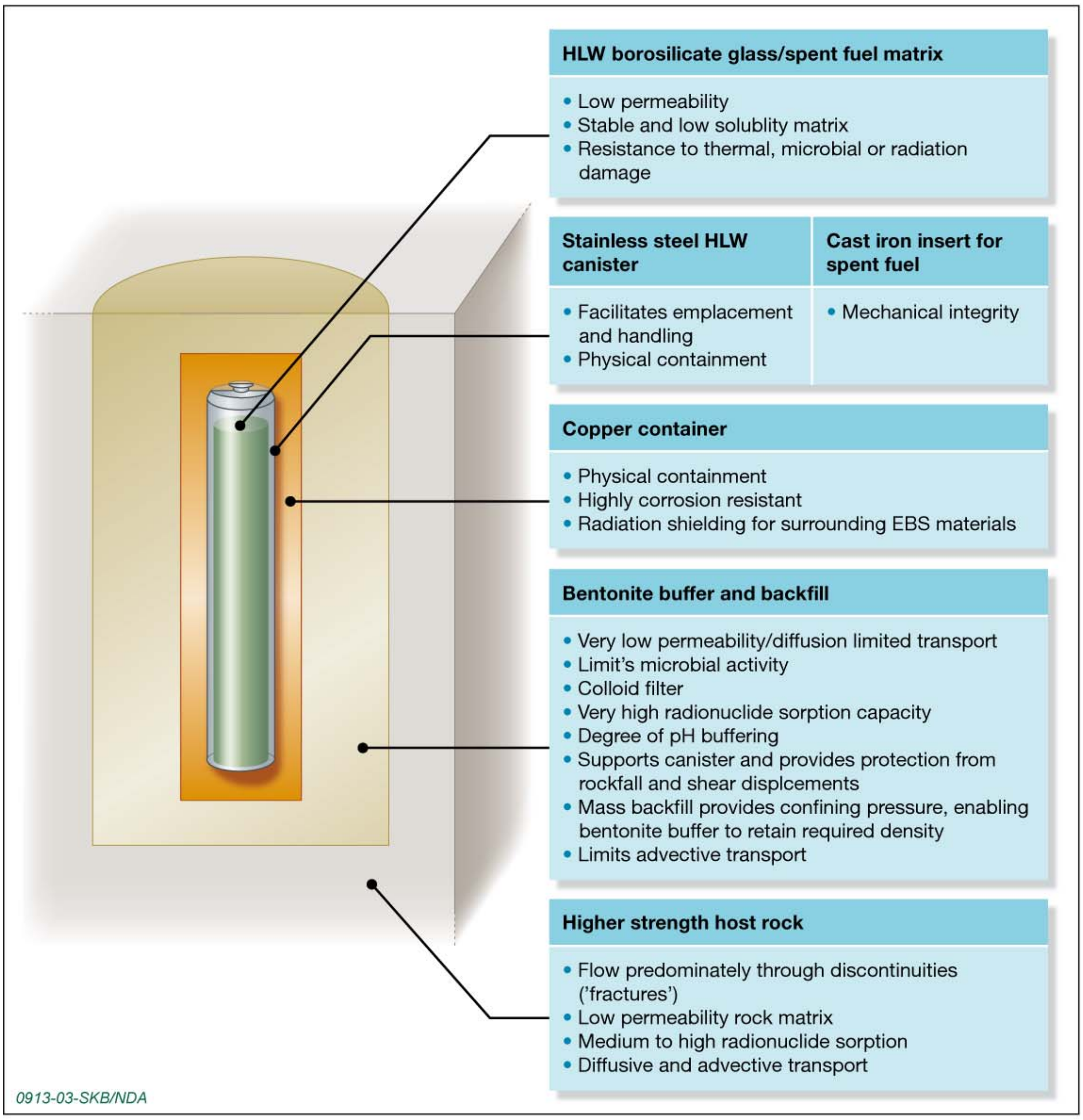


The bentonite buffer would swell as it resaturates and, therefore, would provide a low-permeability barrier that ensures that solute transport within it is by diffusion. Chemical reactions within the bentonite buffer would condition the porewater within the buffer, thus ensuring favourable geochemical conditions adjacent to the waste package [16], which would contribute to a low corrosion rate [15, 19] and provide favourable conditions for sorption [20]. The copper canister provides a very long period of absolute containment for the wastes [15]. For example, the Swedish Nuclear Fuel and Waste Management

Organisation (SKB) has concluded that the copper canister will provide at least 100,000 years of absolute containment [176]. In their base case assessment scenario, SKB assumes that, unless undetected manufacturing flaws are present, all of the canisters remain intact for at least 100,000 years and less than 1% have failed by 1,000,000 years after closure [176]. The ceramic SF and vitrified HLW wasteforms provide an additional barrier owing to their slow dissolution rates [15], which limit the rate at which radionuclides are released into the near field. The different engineered barriers and the processes and properties that contribute to them fulfilling their safety functions are illustrated in Figure A2.

Figure A2 The barriers to radionuclide release for the HLW/SF disposal concept in a higher strength host rock

The figure shows the barriers to radionuclide release from the HLW/SF disposal area in a higher strength host rock and the processes and properties that contribute to their safety functions. The Near-field evolution status report [16] gives further details of the materials, processes and properties shown.



GDF layout

We have developed a conceptual layout of the underground facilities based on the illustrative geological disposal concept examples for a higher strength host rock [24]. When considering a real candidate site, the layout would be optimised to make best use of the conditions and space available. The layout has sufficient capacity for the Derived Inventory reference case and has a footprint of approximately 5.6 square kilometres.

A4 Expected evolution

In this section we describe the expected normal evolution of the geological disposal system for the illustrative geological disposal concept examples for higher strength rock. This description draws on material in the research status reports and in our Generic disposal facility designs report [24]. Important processes and the timescales on which we expect them to be important are illustrated in Figure A3 for the ILW/LLW disposal area and in Figure A4 for the HLW/SF disposal area.

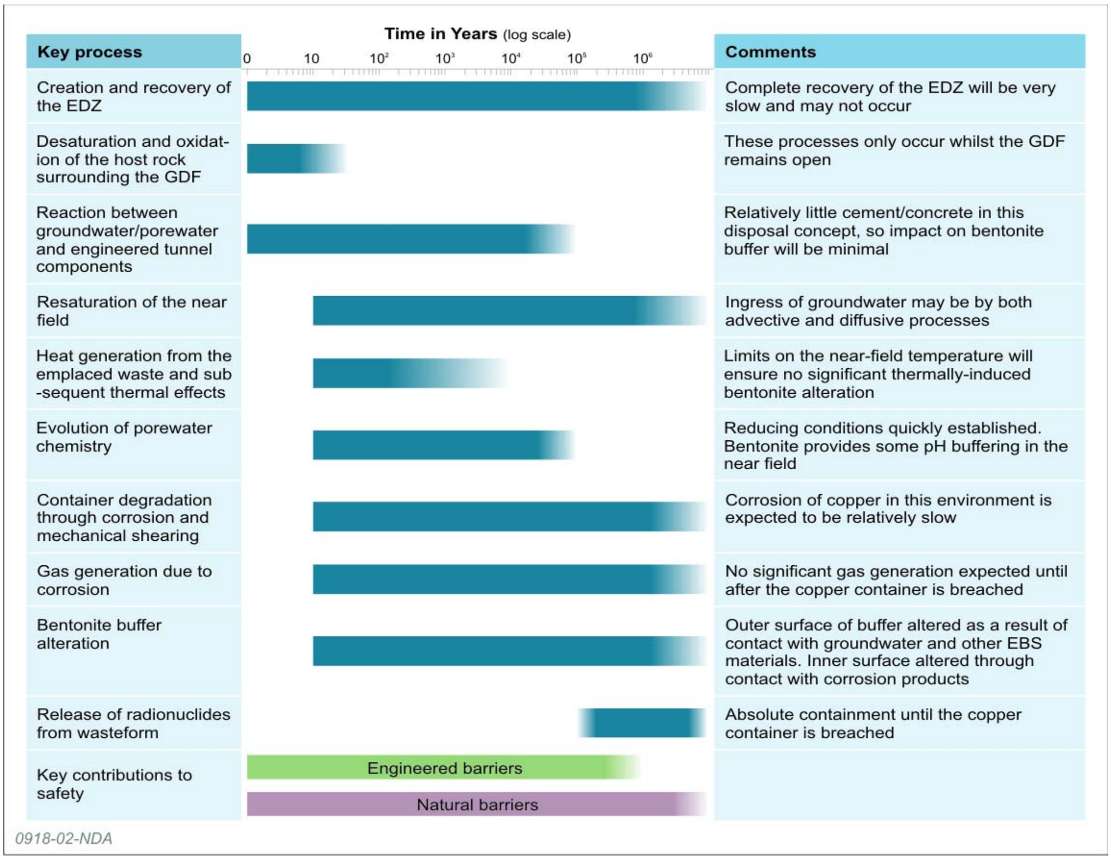
Figure A3 The evolution of the ILW/LLW disposal area in the illustrative geological disposal concept example for higher strength rock

The time axis starts from the point at which the disposal vault is excavated. Figure taken from the Near-field evolution status report [16], which gives further details of the processes and events shown.

Excavation and fitting out of each individual disposal vault or disposal tunnel could take up to several years, and we expect that a few tens of ILW/LLW disposal vaults and a few hundred HLW/SF disposal tunnels would be required to accommodate the Derived Inventory reference case. It would therefore take many decades to excavate and fit out the disposal areas were all of the disposal vaults and disposal tunnels to be constructed before waste emplacement began. Such an approach is likely to be impractical, and we expect that excavation and fitting out of new disposal areas would occur in parallel with the emplacement of waste in other areas of a GDF (i.e. disposal vaults and tunnels would be constructed on a 'just-in-time' basis). This strategy would also allow us, if necessary, to refine our designs to take account of earlier experience and changes in the characteristics and volumes of waste to be disposed. Further details of how we would ensure this can be achieved safely can be found in the OSC [6]. Therefore, at any given time, different processes would be occurring in different vaults and tunnels as they progress through excavation and operations and finally to closure.

Figure A4 The evolution of the HLW/SF disposal area in the illustrative geological disposal concept example for higher strength rock

The time axis starts from the point at which the disposal tunnel and deposition holes are excavated. Figure taken from the Near-field evolution status report [16], which gives further details of the processes and events shown.



We describe the expected evolution of a disposal system in a higher strength host rock below in terms of three time periods:

1. Construction and operational period. We assume that the operational phase of a GDF would last for about 100 years (starting from about 2040) and that a GDF would be closed as soon as waste emplacement is complete. We also assume that there would not be a subsequent period of extended retrievability when the disposal areas are left open (i.e., unbackfilled or unsealed).

2. Early post-closure period (establishment of long-term conditions and engineered barriers fulfil their safety functions, to a few tens of thousands of years, depending on disposal concept).

3. Late post-closure period (engineered barriers begin to lose their effectiveness and would no longer be able to fulfil their safety functions fully, to one million years and more).

These periods are defined primarily in terms of the events and processes that would occur in the future, rather than in terms of years, because the different characteristics of the two disposal areas mean that they would evolve at different rates.

Some of the processes described below would begin soon after the waste has been emplaced and may occur on a timescale that is shorter or comparable with the length of the operational period (about 100 years). Therefore, technically, these processes may occur, or even complete, during the operational period in some disposal vaults or tunnels and at the start of the post-closure period in others. In order to prevent repetition, such processes are only described once. For each period, processes that occur in both disposal areas are described first, then processes that are specific to the ILW/LLW disposal area, and finally processes that are specific to the HLW/SF disposal area.

The description draws heavily on material discussed in the research status reports, and further details of the processes described may be found in these reports.

A4.1 Evolution during construction and operation of a GDF

Construction activities would lead to the formation of an excavation disturbed zone (EDZ) around the tunnels and vaults, due partly to excavation-related vibrations and partly to changes in the stress field in the rock, which lead to brittle fracturing. The redistribution of stress may also cause rock spalling to occur from the walls of the excavations and/or the reactivation of pre-existing minor faults in the host rock. The extent of the EDZ and its characteristics would depend on the geotechnical characteristics of the host rock and the *in situ* stress, both of which are site-specific, and on the excavation technique and any excavation support that is installed. The EDZ may extend up to several tens of centimetres from the excavated region and have different mechanical and hydraulic properties to the intact host rock [16]; it may be more extensively fractured than the intact host rock and, consequently, could be more permeable. In the long-term, the EDZ may become sealed through processes such as mineralisation in fractures, but there is unlikely to be significant sealing through rock creep in a higher strength host rock. As a result, any recovery of the EDZ is likely to be quite slow and unlikely to occur during this period. However, construction activities may include sealing any significant fractures in the EDZ, which might otherwise provide an enhanced flow pathway, by filling them with some form of grout. The grout composition would be chosen to be compatible with the other near-field materials [16]. For example, low-pH cements might be used to seal fractures in areas where extensive use is made of bentonite as a barrier material.

The final stage of the operational period is closure engineering, which involves the backfilling of the various access tunnels and shafts, and emplacement of the main low-permeability seals. Ideally, the permeability of the seals should be no higher than the rock in which they are emplaced.

During this early period of GDF evolution, we expect the EBS to provide complete containment of the disposed inventory, with the exception of gas released through the vents in the ILW packages.

ILW/LLW

During the operational period, the ILW/LLW vaults and the access tunnels would be maintained in a dry and ventilated state. This would result in some desaturation of the host rock around the excavations. The rock immediately adjacent to the excavations may also undergo chemical changes as a result of oxidation reactions or as a result of reactions with materials, most likely concretes, that are used to provide structural support to the excavations. For example, fracture zones might be grouted to reduce water inflows or lined with sprayed concrete to reduce the potential for spalling and/or rock falls. Some of the ensuing reactions may contribute to the sealing of fractures in the EDZ referred to above. The presence of a GDF that is maintained in an open and ventilated state for many decades could, through the flow it induces, lead to changes in the composition of the groundwater in the surrounding host rock. For example, upwelling of more saline water from depth is considered to be a possibility in Sweden [176]. Minor thermal perturbations of the host rock surrounding the excavated spaces may also occur.

The ILW/LLW packages would be in an aerobic environment throughout the operational period. Under these conditions we expect that the degradation processes (primarily corrosion processes) would be different, and in some cases more rapid, than those we expect to dominate in the long term, after closure. The Package evolution status report [15] describes the various degradation processes that might affect the waste packages, their likely rates and the environmental and other parameters that control them. We would design the operational regime to ensure that waste package degradation during the operational period is kept to a minimum, for example by ensuring that groundwater cannot come into contact with the packages and ensuring that temperature and humidity are controlled within pre-defined limits. Gas would be generated within some of the waste packages [19], and any that is released through the vents would be removed by the ventilation system and discharged in accordance with the discharge authorisation [19, 26]. Similarly, water pumped from the vaults would be treated in the effluent treatment system before discharge (see [26]).

The ILW/LLW disposal vaults would be backfilled with NRVB as part of the closure engineering. Emplacement of the NRVB would transiently increase the temperature and bring water into contact with the waste packages. This is likely to result in an increase in the rate of corrosion of reactive metals within the waste packages and a transition towards general anaerobic corrosion of the waste containers and steels within the waste packages. Hence the overall rate of gas generation would be expected to increase. Delaying backfilling until closure would enhance retrievability.

HLW/SF

In the HLW/SF disposal area, the disposal tunnels would be backfilled with a mixture of crushed rock and bentonite and sealed with an operational seal[24] made from low-pH concrete as soon as all of the waste packages, and their bentonite buffers, in the tunnel have been emplaced. We envisage that individual disposal tunnels would remain open for at most a few years after emplacement of the first waste package in the tunnel. Reducing conditions are likely to develop within a few years of sealing of the disposal tunnel [16].

As soon as the bentonite buffer and the tunnel backfill have been emplaced, the bentonite would begin to swell, as it absorbs moisture from humidity in the air and any groundwater it comes into contact with. Eventually the bentonite buffer would swell to the point where it completely seals the space between the waste package and the host rock and forms a uniform low-permeability barrier encapsulating the disposal canister. Prompt backfilling and sealing of the disposal tunnels would be essential to ensuring that the saturated density of the bentonite buffer is sufficiently high that it can fulfil its safety functions [16]. The methodology for emplacement of the engineered barriers within the near field would therefore be of key importance in ensuring that the buffer attains a sufficiently high density to ensure long-term safety. The time taken to fully resaturate an individual deposition hole in a higher strength rock would vary between deposition holes, but is likely to be between a few years and a decade or so, depending on the characteristics of the fractures that intersect the hole [16]. Resaturation of the disposal tunnels would take longer (probably decades to centuries) because there would be a larger volume of pore space to be resaturated. Thus, the early deposition holes, and possibly some of the early disposal tunnels, would likely be fully resaturated before final closure. We would take the expected range of resaturation times for the candidate site into account during the design process.

The HLW/SF packages would initially give off significant quantities of heat as a result of decay of the short-lived component of the inventory. While the disposal tunnels are open, we would use the ventilation system to ensure that temperatures in the open parts of the excavations remain within specified limits. Once a disposal tunnel has been sealed, the

[24] Temporary seals that could be removed with relatively little effort if required. They would be upgraded or replaced at closure.

temperature in the buffer and in the surrounding rock may rise several tens of degrees above ambient. Thermal constraints are likely to be a significant factor determining package spacing and underground layout.

A4.2 Early post-closure evolution

Conditions (hydraulic and geochemical) in the surrounding host rock would slowly return to values close to undisturbed conditions over about 100 - 1,000 years, depending on the properties of the host rock.

ILW/LLW

Following closure, the ILW/LLW disposal area would start to resaturate. The time taken for the ILW/LLW disposal vaults to resaturate fully would depend on the properties of the host rock, but in a typical higher strength host rock we expect it to be of the order of a few decades to a few centuries [16]. The incoming groundwater would rapidly equilibrate with the NRVB resulting in the development of alkaline conditions. Corrosion reactions would result in the establishment of reducing conditions [16].

Although high-pH reducing conditions would limit the rate of corrosion [15], gas (dominantly hydrogen, but also some carbon dioxide and methane) would be generated, mostly by the corrosion of Magnox and aluminium present in the waste, but also from the corrosion of the various steel components and the degradation of organic materials. The carbon dioxide would react with the cementitious backfill [19]. Some of the gas would dissolve in the groundwater, but it is likely that a free gas phase would form. However, we expect the gas to be able to escape relatively easily into the host rock [19], although its presence may slightly reduce the rate of resaturation. Depending on the gas transport properties of the geosphere at the site, this free gas might reach the ground surface.

Once the disposal vault resaturates, groundwater could diffuse into waste containers where these have vents and diffuse into the wasteform through the low permeability capping grout and then through the encapsulation grout. Once a continuous water phase was established in this manner between wastes and the vault backfill, small amounts of radionuclides in the ILW/LLW inventory could begin to be released, in dissolved form, by diffusion back along this pathway. The alkaline conditions established by the use of cementitious backfill causes steel corrosion to occur at a very low rate and therefore metallic iron, and the hydrogen produced form its corrosion, will be present for a very long time. We expect the quantity of cementitious backfill in the ILW/LLW disposal area would be sufficient to maintain high-pH reducing conditions in the ILW/LLW disposal area for at least a hundred thousand years [16]. These conditions would reduce the rate of corrosion and the activity of microbes that degrade the organic components of the waste, and hence limit the gas generation rate. The high pH would also limit the solubility, and the minerals in the cement would promote sorption of key radionuclides [20]. The wasteform and the chemical barrier provided by the backfill would both significantly limit the rate of release of contaminants from the near field.

Some mobile radionuclides (e.g. Cl-36, I-129) may be released into the host rock and, depending on the nature of the overlying sedimentary sequence, may be transported into the biosphere during this period before they have decayed completely. Sorption and rock matrix diffusion are not particularly effective at retarding these mobile species. However, even if the travel time in the geosphere is short, the combination of the cementitious near field and the geosphere may provide sufficient retardation to ensure that more strongly sorbed species such as actinides never reach the biosphere. The majority of the inventory would remain within the near field, either within the wasteform or sorbed to the backfill.

During this period, the engineered barriers would degrade slowly as a result of interaction with the host rock groundwater and with each other. However, they would remain sufficiently intact that they are able to perform their intended safety functions fully. Throughout this period, the near-field pore-water would be conditioned to high pH by the dissolution of components of the near-field cementitious materials [16]. However, cracking of both the backfill and the wasteforms would be likely to occur [15, 16], although we do not expect this to have a significant impact on the overall performance of the cementitious barriers. The impact of organic complexing agents derived from the wastes, which can increase radionuclide mobility [20], would become progressively less significant as their concentration in the waste is reduced by chemical, radiolytic or microbial degradation, as well as by processes such as dilution or sorption.

We expect an alkaline disturbed zone (ADZ) to develop around the ILW/LLW disposal area as a result of reactions between the host rock and porewater that has been chemically conditioned by the cementitious backfill [17]. The resulting mineral dissolution and precipitation would alter the hydrogeological properties of the host rock around a GDF. We expect there to be a net decrease in porosity and permeability as fractures, especially those in the EDZ, become filled with new, relatively high-volume, minerals.

HLW/SF

In the HLW/SF disposal area, the resaturation process described above would continue and complete. The bentonite buffer would provide a low-permeability barrier around the disposal canister that would protect the canister by limiting the rate of transport to the canister surface of water and aggressive species such as sulphides and microbes that might promote corrosion, and by limiting the rate of transport of any corrosion products away from the canister. The buffer would also protect the canister from events such as minor movement on the discontinuities that intersect the deposition hole.

As noted above, the HLW/SF waste packages would give off significant quantities of heat until the short-lived component of the inventory has decayed. As a result, the temperature of the buffer and the surrounding rock may rise to several tens of degrees above ambient before slowly decreasing. The peak temperature in the EBS is likely to occur within a few decades of waste emplacement, but we expect that temperatures in the HLW/SF disposal area would remain above ambient for around 10,000 years [16]. However, we expect the copper disposal canister to remain intact for at least 100,000 years and potentially longer [15], so temperatures would have returned to ambient well before the earliest time that the container might be first penetrated by corrosion. The elevated temperatures in the HLW/SF disposal area may also result in slightly increased (up to about 10 degrees) temperatures in the ILW/LLW disposal area. We would need to take this into account when designing the layout of the disposal areas.

In addition to the swelling that occurs on resaturation, interaction with the host rock groundwater would result in some minor changes to the physical and chemical properties of the bentonite buffer. Some minerals may dissolve and others may precipitate, and ion exchange reactions would occur. These reactions result in the development of reducing conditions and buffer the bentonite pore water to a mildly alkaline composition. The reactions also have the potential to influence the swelling pressure of the bentonite, and hence its ability to fulfil its safety functions. Elevated temperatures may increase the rates of some of these reactions and may influence fluid movements, in particular transport of water vapour, which is an important process in the resaturating bentonite. Further details of the types of reactions that might occur and their potential impact can be found in the Near-field evolution status report [16].

Slow corrosion of the copper canister would occur throughout this period. However, except in the rare case of an undetected manufacturing flaw, we do not expect penetration of the canister to occur for at least a hundred thousand years. By this time, a high proportion of the radionuclide inventory would have decayed.

A4.3 Late post-closure evolution

In the very long term (many hundreds of thousands to millions of years), we do not expect the EBS to provide containment, although many of the HLW/SF copper canisters may remain intact. However, the degraded EBS barriers are likely to continue to retard the release of contaminants to some degree, although by this time the vast majority of the inventory would have decayed. On these timescales, the main processes limiting the rate of release to the biosphere are sorption and rock matrix diffusion in the geosphere. On these very long timescales we would need to consider the potential impact of various "natural" disturbances or transients arising from the effects of global climate events (e.g. glaciation) [17].

ILW/LLW

Many tens, or hundreds, of thousands of years after closure, the engineered barriers in the ILW disposal area would begin to lose their effectiveness and would no longer be able to perform their intended safety functions fully. With time, the wasteforms and the NRVB would become degraded through reaction with the host-rock groundwater [16], and eventually the pH in the ILW/LLW disposal area would fall to a level where the chemical barrier provided by the NRVB loses some of its effectiveness. However, we would include sufficient NRVB in our design to ensure that the chemical barrier would remain at least partially effective on timescales of up to 1,000,000 years. Those radionuclides that have not decayed within the waste packages and near field may be released to the host rock and, from there, eventually may be transported to the biosphere [20]. The geosphere would however continue to provide an important barrier to limit the rate of release to the surface environment. Sorption and rock matrix diffusion would retard the transport of radionuclides in the geosphere, in many cases to the extent that they are fully contained within the geosphere because the transport time is sufficiently long that they decay before reaching the biosphere.

Gas would continue to be produced from the corrosion of metals, but we expect the production rate to have significantly declined by this time [19].

HLW/SF

For HLW/SF, after about 10,000 years, a small number of the copper canisters, those with undetected manufacturing defects, may fail. However, we expect the vast majority of the copper canisters to have a lifetime of at least 100,000 years. Once a copper canister has failed, water would access the cast iron insert, which would corrode, generating hydrogen gas. If the rate of gas production exceeds the rate at which gas can be transported away by diffusion, a free gas phase may form and the gas would be released via pressure-induced microfissures that reseal once the gas has been released [19].

Once water penetrates the disposal canister, the SF and vitrified HLW would start to dissolve and release any radionuclides that have not decayed *in situ* [15, 20]. In the case of the HLW, the majority of the inventory would have decayed by this time; slow dissolution of the glass matrix would limit the release rate of any remaining inventory. For the SF, the portion of the instant-release fraction[25] that has not decayed *in situ* would be released rapidly, and then the SF would dissolve slowly and release the remaining radionuclides [15, 20]. We would expect the bentonite buffer to provide an effective barrier to the transport of radionuclides from the failed waste package to the host rock. The transport of many radionuclides that are released from the EBS would be sufficiently retarded in the geosphere that they would decay completely before reaching the biosphere.

[25] The portion of the inventory (fission and neutron activation products) that has segregated to the grain boundaries in the wasteform and may be released rapidly upon exposure to groundwater.

A5 Operational environmental safety

We describe our approach to assessing operational environmental safety in [26] and summarise it in the generic ESC main report. Once a preferred site has been identified, we will tailor the Operational Environmental Safety Assessment to that site.

A6 Post-closure environmental safety

Our confidence in the post-closure environmental safety of a GDF developed in a higher strength host rock is based on several decades of R&D. In the case of the ILW/LLW concept, we are building on the work of Nirex [80], and in the case of the HLW/SF concept, we are building on the work of the national waste management organisations in Sweden (SKB) and Finland (Posiva) (e.g. [176]), supplemented by our own work to understand any modifications that might be required to adapt the concept to UK conditions.

Both disposal concepts provide multiple barriers to the release of the inventory, and the depth of disposal ensures isolation. We have a good understanding of the major chemical and physical processes that would provide containment of the inventory and retardation of any radionuclides that are released from the waste packages. Our R&D programme will be developed to address the prioritised knowledge gaps that have been identified in the research status reports (see Section 6.3 of the main report).

ILW/LLW

The concept relies on the wasteforms and the chemical barrier provided by the cementitious backfill to provide containment in the near field. The long-term evolution of cement mineralogy is sufficiently well understood and we can predict with confidence the evolution of the pH with time for a given combination of cement and groundwater compositions and groundwater flow rate [16]. When we prepare our detailed site-specific design, we would adjust the quantity, and possibly also the composition, of the cementitious backfill to suit the groundwater composition and flow rate at the site. In this way we would be able to ensure that the chemical barrier is sufficiently long-lived that it is able to fulfil its safety functions for the required length of time. This chemical barrier would reduce the rate of gas production by limiting corrosion rates and microbial activity and promote the retention of most radionuclides within the near field.

In this concept, the geosphere can be described as having two main roles. It isolates the near field from the evolving conditions at the ground surface, meaning that we can have confidence that the host rock groundwater composition, and to a large degree flow rate, would remain within the ranges we have considered when specifying the quantity and composition of backfill. The geosphere would also have an important role as a barrier in containing and retarding radionuclide release. The EBS is not especially effective at containing mobile radionuclides such as Cl-36 and I-129, which do not sorb well to either cements or geosphere materials, although it does provide effective containment of actinides, which tend to sorb well to both cements and geosphere materials. The geosphere is therefore required to provide a travel time barrier, probably of several tens of thousands of years, as a result of a long path length and/or slow flow, and to dilute and disperse the radionuclides so that they do not reach the biosphere in concentrations that might be harmful. Therefore the properties of the geosphere are of key importance when considering the post-closure performance of the concept.

HLW/SF

The KBS-3V disposal concept relies on robust engineering to provide absolute containment of the inventory for periods of 100,000 years or more, by which time the vast majority of the disposed activity would have decayed in a GDF. The characteristics of this host rock are such that we expect resaturation to be a relatively quick process (years to decades to resaturate a deposition hole), which means that the bentonite buffer would swell and become established as a highly effective barrier soon after disposal; the density of the saturated bentonite is crucial to it performing its safety functions. This builds confidence in the longevity of the copper canister because the bentonite buffer, which is designed partly to protect the copper canister, is fully established early in the post-closure period, and for some deposition holes during the operational period.

The KBS-3V disposal concept was designed to allow safe disposal of SF in a geological environment in which a higher strength host rock extends to the ground surface. The geosphere provides a stable chemical, hydrogeological and structural environment in which the near field would evolve. In the UK, potential host rocks of this type may not extend to the ground surface, in contrast to the geological environment for which the KBS-3V disposal concept was developed and for which safety cases have been developed for sites in Sweden and Finland (for example [176]). If the host rock does not extend to the ground surface (as is possible in the UK), we expect the majority of the infiltrating groundwater to flow within the cover rocks. This partial separation of the flow systems in the cover rocks and the host rock serves to further isolate the disposal areas from the surface environment, and its associated hydrogeological and hydrochemical evolution.

The timescale on which we envisage a UK GDF would be built is such that we expect that practical operational experience of the KBS-3V concept would have been gained in Sweden and/or Finland before we finalise our detailed design. We would learn from this experience to refine our design and procedures for emplacing the EBS and waste, should we decide to use a concept based on KBS-3V in the UK.

Appendix B Illustrative geological disposal concept examples for lower strength sedimentary host rock as applied to the UK

B1 Introduction

This appendix illustrates the case of a GDF constructed within a geological environment characterised by a lower strength sedimentary host rock (see Table 4.2 of the main report and the Geosphere status report [17] for a summary of the characteristics of this rock type). In the UK, such rocks tend to be geologically younger than the higher strength rocks considered in Appendix A and they tend to be clay-rich. Such a host rock would typically comprise a sedimentary rock where any fluid movement is predominantly through the rock matrix. By "lower strength" we mean rocks in which it is possible to excavate tunnels of 10 to 15 metres in diameter at disposal depths, although these excavations are likely to require considerable excavation support, possibly including full linings. Fractures in lower strength rocks may tend to heal with time as a result of creep.

As explained in the main report, the UK Government sees no case for having more than one GDF if one facility can be developed to provide suitable, safe containment for the entire inventory. We are therefore describing in this appendix a conceptual layout for a GDF in a lower strength sedimentary rock that is based on two distinct disposal areas implementing different disposal concepts, one for unshielded intermediate-level waste (UILW), shielded intermediate-level waste (SILW), low-level waste (LLW) and depleted, natural and low enriched uranium (DNLEU), should it be declared as waste, and the other for high-level waste (HLW), and spent fuel (SF), highly enriched uranium (HEU) and plutonium (Pu), should these be declared as wastes. These disposal areas would share common access ways and surface facilities, but would be separated from each other to ensure that the impact of interactions between the two disposal areas is sufficiently small that it does not compromise the overall performance of a GDF.

The assumptions that underpin the generic DSSC are set out in [32]. As discussed in [32] and shown in Table 4.3 of the main report, we are basing our generic designs for a lower strength sedimentary host rock on the Swiss Cooperative for the Disposal of Radioactive Waste (Nagra) disposal concepts for ILW and HLW/SF in the Opalinus Clay [177]. We are also taking account of the large amount of work carried out by the French National Radioactive Waste Management Agency (Andra) to develop a GDF in this type of host rock [178]. Should we find ourselves considering a candidate site with a lower strength sedimentary host rock, we would develop site-specific concepts, which may, or may not, be based on these disposal concepts.

In this appendix, where we refer to the ILW/LLW disposal area or vaults, we mean the part of a GDF in which a disposal concept based on the Nagra ILW disposal concept is implemented, that is the disposal area for UILW, SILW, LLW and DNLEU. When we refer to the HLW/SF disposal area or tunnels, we mean the part of a GDF in which a disposal concept based on the Nagra HLW/SF disposal concept is implemented, that is, the disposal area for HLW, and SF, HEU and Pu. However, discussion focuses on ILW/LLW and HLW/SF because, to date, this is where our conceptual design and assessment work has focused.

One of the purposes of this appendix is to illustrate how the different components of our work programme fit together to provide knowledge about the environmental safety of a GDF. It therefore draws together information about the characteristics and likely evolution of a GDF developed in a lower strength sedimentary host rock from a number of sources, most notably the Package evolution [15], Near-field evolution [16], Geosphere [17], Gas [19], and Radionuclide behaviour [20] status reports, which describe our current state of knowledge and understanding. These status reports provide details about the characteristics, safety functions and expected evolution of the different barriers in the multi-barrier system.

B2 Geological environment

The geological environment for the lower strength sedimentary host rock example comprises a flat-lying indurated[26] clay-rich host rock that is overlain by other sedimentary formations and a near-surface zone of weathered rock and recent (Quaternary) deposits. In the UK, this sequence is likely to comprise a mixture of low-permeability and higher-permeability sedimentary rocks, possibly including minor aquifers and evaporites.

The host rock is assumed to be a low-permeability indurated clay formation, which has a thickness of at least 50 metres of intact host rock above the disposal areas. This host rock has similar characteristics to the host rocks being evaluated by Andra (the Callovo-Oxfordian Clay) and by Nagra (the Opalinus Clay). Faults, if present, are expected to be of low permeability, based on observations from hydrocarbon exploration and development. The permeability[27] is low enough that we can be confident that solute transport in the host rock would be dominantly by diffusion. The bedded nature of the rocks means that the permeability parallel to bedding may be significantly greater than the permeability perpendicular to bedding [17]. Many radionuclides, for example actinides, sorb strongly to clay-rich rocks [20]. The Geosphere and Gas status reports [17, 19] describe the properties of the lower strength sedimentary host rock in more detail.

As noted in the definition of this rock type, the maximum practicable excavation span at disposal depths is limited to 10 to 15 metres and extensive rock reinforcement, perhaps including a full concrete lining, is likely to be required for stability [139]. However, it should be possible to construct tunnels of more than 10 metres in diameter at depths of 300 to 500 metres if sufficiently robust rock support is installed. We would need to take into account the tendency of these types of rocks to creep when stressed (slow plastic deformation). Clays may also be susceptible to ‘weathering’ as a result of the wetting/drying cycles that could accompany GDF operations.

The composition of the groundwater in the host rock will reflect the original porewater composition at deposition, the mineralogy, subsequent water-rock interactions and the composition of the overlying rocks, which may include evaporites. Many low-permeability, lower strength sedimentary rocks in the UK were deposited in a marine environment and their present-day porewater composition still reflects this. If the overlying sequence does not contain evaporites, the host rock groundwater may be of low to moderate salinity (total dissolved solids in the range 1,500 to 35,000 milligrams per litre), but if evaporites are present the host rock groundwater is likely to be highly saline (brine) [16, 17].

The low permeability of the host rock means that diffusion is likely to dominate solute transport in the host rock. Solute transport in any low-permeability units in the overlying sequence is also likely to be dominated by diffusion. However, advection may be important in any more permeable minor aquifer units in the overlying sequence. Head gradients driving groundwater flow are likely to be low because, in the UK, these types of rocks are usually found in areas with little topographic relief. Solute transport processes are discussed in more detail in the Geosphere and Radionuclide behaviour status reports [17, 20].

We expect that it may be difficult for gas generated within the disposal areas to migrate into the host rock as a free gas phase. As a result, the disposal areas may become over-pressured if the gas generation rate is sufficiently high, although our current understanding suggests that the resulting pressures would not be so high as to compromise the integrity of the host rock [19]. We expect to be able to develop an EBS that ensures that pressures

[26] Hardened.

[27] Intrinsic permeability is a property of the permeable medium. The more commonly quoted hydraulic conductivity depends on the density, viscosity and temperature of the fluid as well as the permeability. Further details can be found in [17].

in the near field do not become too high (e.g. see [179] for the approach taken by Nagra). We also expect that the overlying sedimentary sequence may trap any gas that escapes from the disposal areas and the host rock before it reaches the ground surface. The migration of gas in a lower strength sedimentary rock is discussed in more detail in the Gas status report [19].

We expect that natural conditions in the host rock at disposal depths would evolve only slowly because low-permeability rocks such as we are considering here respond only slowly to changes in external driving forces (tectonic events, glaciation). Indeed, conditions at disposal depths may still be evolving in response to the effects of geological events that occurred many hundreds of thousands years ago, or even more than a million years ago. This property also means that we expect the disturbance associated with the development of and operations within the disposal areas to be relatively localised. Groundwater heads in the host rock may not be in equilibrium with surrounding, slightly higher permeability, rocks, meaning that, in the absence of site-specific data, it may be difficult to predict flow directions in any transmissive parts of the closed disposal areas. Large-scale environmental changes such as would be associated with glaciation are unlikely to affect conditions at disposal depths [17].

B3 Engineered barrier system

This section summarises the application of the illustrative geological disposal concept examples for a lower strength sedimentary host rock (as presented in Table 4.3) to the UK, as set out in our Generic disposal facility designs report [24].

ILW/LLW

In this concept, the disposal vaults would typically be about 10-metres wide and about 11-metres high, about 100-metres long, and separated from each other by a distance of about 30 metres. In the Nagra and Andra concepts, the basic ILW disposal package is a concrete disposal box (0.15-metre wall thickness) that typically contains between four and ten waste packages. The remaining void space within the box is backfilled with cementitious grout. The disposal boxes form the basic disposal unit. Any remaining space in the tunnels is backfilled with cement mortar that is strong enough to resist creep closure of the disposal tunnels. Nagra plans to use a specially developed high-strength, high-porosity mortar, partly in order to provide sufficient gas storage volume within the EBS to reduce the potential for gas pressure build up. In the Andra design, the tunnel linings are designed so that the disposal packages fit the disposal tunnels exactly and no backfilling is required. The volume of backfill within and between the disposal boxes would be sufficient to maintain high-pH conditions in the vaults for many tens of thousands of years.

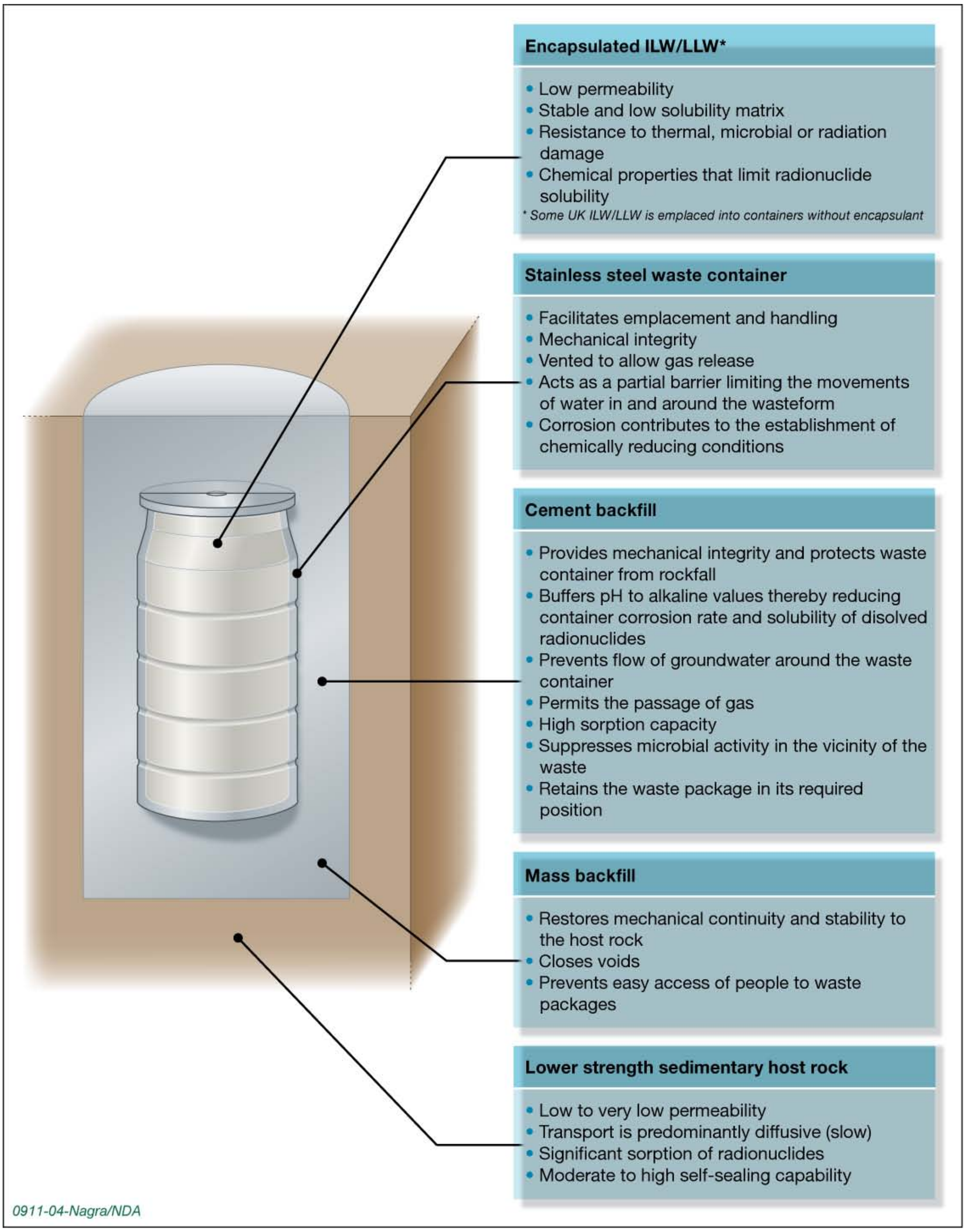
The host rock, which is of sufficiently low permeability that solute transport within it is primarily by diffusion, provides an important barrier to release in this disposal concept, and would contain many of the more strongly sorbed contaminants until they have decayed. Containment is also provided by the wasteform and the cementitious backfill around the waste packages. Most of the ILW packages are vented to prevent the build-up of excessive gas pressures. We assume that the stainless steel waste containers would provide only limited containment during the post-closure period because, even at closure, the container is not fully ‘intact’. The vent allows the release of gaseous radionuclide during transport and operations and the release of both gaseous and dissolved radionuclides after closure, even if there is no corrosion or other degradation of the container. However, the waste containers do play an important role in containing and protecting the wasteform during the operational period and in limiting the release of short-lived radionuclides in the period immediately after the GDF is closed.

The role of the concrete disposal box is primarily to facilitate emplacement of the waste, to protect the waste packages during the operational period and to facilitate the retrieval of waste, if needed.

The different engineered barriers and the processes and properties that contribute to them fulfilling their safety functions are illustrated in Figure B1.

Figure B1 The barriers to radionuclide release for the ILW/LLW disposal concept in a lower strength sedimentary host rock

The figure shows the barriers to radionuclide release from the ILW disposal area in a lower strength sedimentary host rock and the processes and properties that contribute to their safety functions. The Near-field evolution status report [16] gives further details of the materials, processes and properties shown.



HLW/SF

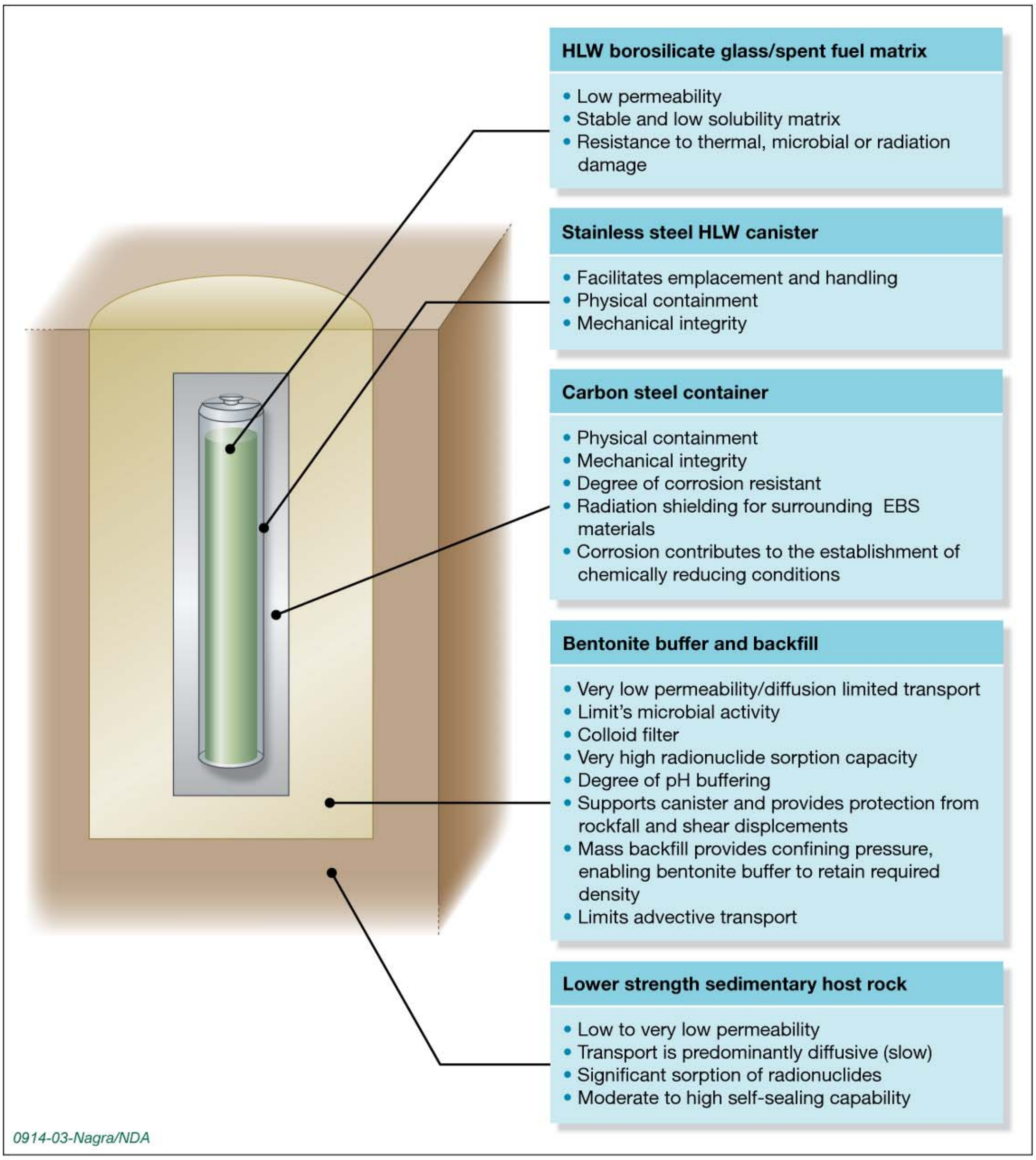
In this concept, HLW in stainless steel canisters and SF rods separated by carbon steel cladding would be placed into thick-walled carbon steel disposal canisters. These disposal canisters would then be placed in horizontal tunnels, separated from each other by a distance of 3 metres, and surrounded by a bentonite buffer (Figure B2). In the Nagra design the HLW/SF tunnels are 2.5 metres in diameter, 800-metres long and separated from each other by 40 metres. We note that Andra does not consider a bentonite buffer to be necessary for HLW.

The tunnels may need to be lined with concrete to provide strength during the operational period. If so, to avoid any adverse impacts on long-term safety, the design would incorporate breaks in the liner structure.

The low-permeability host rock would provide a barrier to the movement of radionuclides away from the disposal area. The bentonite buffer is also an important barrier both as a result of its low permeability, which ensures that solute transport within it is by diffusion, and because it buffers the porewater adjacent to the waste package [16]. While it is intact, the carbon steel disposal canister would provide containment, and many radionuclides sorb strongly to iron corrosion products, so the disposal canister would contribute to retardation even after it has failed. The wasteforms would provide further containment. The different engineered barriers and the processes and properties that contribute to them fulfilling their safety functions are illustrated in Figure B2.

Figure B2 The barriers to radionuclide release for the HLW/SF disposal concept in a lower strength sedimentary host rock

The figure shows the barriers to radionuclide release from the HLW/SF disposal area in a lower strength sedimentary host rock and the processes and properties that contribute to their safety functions. The Near-field evolution status report [16] gives further details of the materials, processes and properties shown.



GDF layout

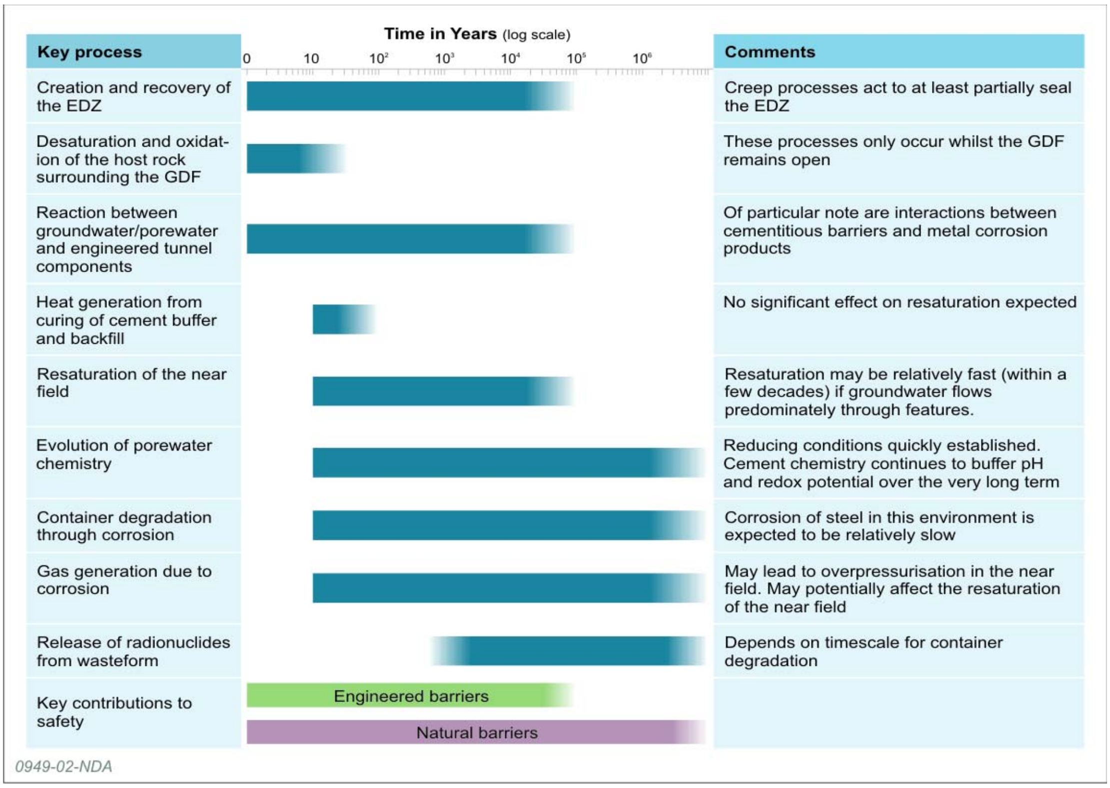
We have developed a conceptual layout of the underground facilities based on the illustrative geological disposal concept examples for a lower strength sedimentary host rock [24]. When considering a real candidate site, the layout would be optimised to make best use of the conditions and space available. This layout provides sufficient disposal capacity for the Derived Inventory reference case, and has a footprint of about 10 square kilometres.

B4 Expected evolution

In this section we describe the expected normal evolution of the geological disposal system for the illustrative geological disposal concept examples for lower strength sedimentary rock. This description draws on material in the research status reports and in our Generic disposal facility designs report [24]. The important processes and the timescales on which we expect them to be important are illustrated in Figure B3 for the ILW/LLW disposal area and in Figure B4 for the HLW/SF disposal area.

Figure B3 The evolution of the ILW/LLW disposal area in the illustrative geological disposal concept example for lower strength sedimentary rock

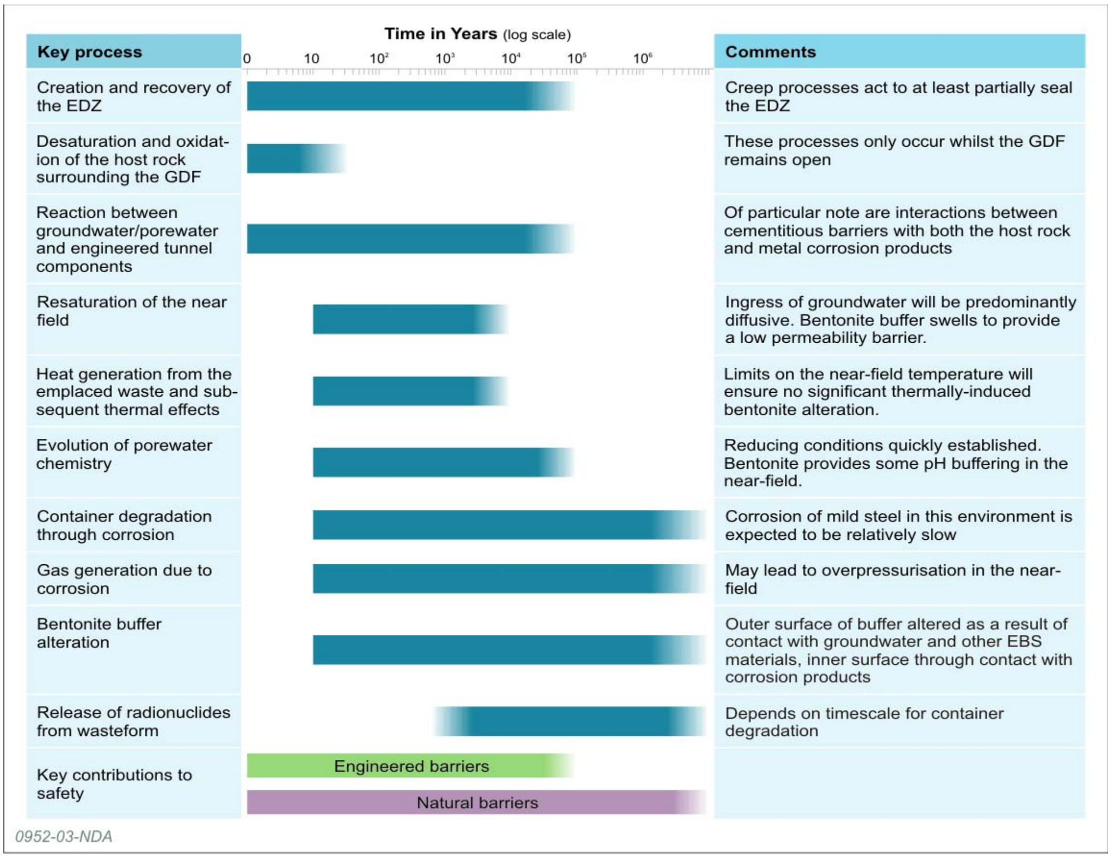
The time axis starts from the point at which the disposal vault is excavated. Figure taken from the Near-field evolution status report [16], which gives further details of the processes and events shown.



Excavation and fitting out of each individual disposal vault or disposal tunnel could take up to several years, and we expect that several hundred ILW/LLW disposal vaults and about a hundred HLF/SF disposal tunnels would be required to accommodate the Derived Inventory reference case. It would therefore take many decades to construct a GDF were all of the disposal vaults and disposal tunnels to be excavated and fitted out before waste emplacement begins. Such an approach is likely to be impractical so we expect that excavation and fitting out of new disposal areas would occur in parallel with the emplacement of waste in other areas of the facility (i.e. disposal vaults and tunnels would be constructed on a ‘just-in-time’ basis). This strategy would also allow us, if necessary, to refine our designs to take account of earlier experience and changes in the characteristics and volumes of waste to be disposed. Further details of how we would ensure this can be achieved safely can be found in the OSC [6]. Therefore, at any given time, different processes would be occurring in different vaults and tunnels as they progress through excavation and operations and finally to closure.

Figure B4 The evolution of the HLW/SF disposal area in the illustrative geological disposal concept example for lower strength sedimentary rock

The time axis starts from the point at which the disposal tunnel is excavated. Figure taken from the Near-field evolution status report [16], which gives further details of the processes and events shown.



We describe the expected evolution of a GDF in a lower strength sedimentary host rock in terms of three periods:

1. Construction and operational period. We assume that the operational phase of a GDF would last for about 100 years (starting from about 2040), and that a GDF would be closed as soon as waste emplacement is complete. We also assume that there would not be a subsequent period of extended retrievability when the disposal areas are left open (i.e., unbackfilled or unsealed).

2. Early post-closure period (establishment of long-term conditions and engineered barriers fulfil their safety functions, to a few tens of thousands of years, depending on disposal concept).

3. Late post-closure period (engineered barriers begin to lose their effectiveness and would no longer be able to fulfil their safety functions fully, to one million years and more).

These periods are defined primarily in terms of the events and processes that would occur in the future, rather than in terms of years, because the different characteristics of the two disposal areas mean that they would evolve at different rates.

Some of the processes described below would begin soon after the waste has been emplaced and may occur on a timescale that is shorter than the length of the operational period (about 100 years). Therefore, technically, these processes may occur during the operational period in some disposal vaults or tunnels and at the start of the post-closure period in others. In order to prevent repetition, such processes are only described once.

For each period, processes that occur in both disposal areas are described first, then processes that are specific to the ILW/LLW disposal area, and finally processes that are specific to the HLW/SF disposal area.

The description draws heavily on material discussed in the research status reports, and further details of the processes described may be found in these reports.

Evolution during construction and operation of a GDF

Construction activities would lead to the formation of an excavation disturbed zone (EDZ) around the tunnels and vaults. The properties of the EDZ would evolve, partly in response to the clay-dominated host rock drying out during the operational phase, and partly because lower strength sedimentary rocks can exhibit elasto-plastic behaviour (creep). The extent of the EDZ and its characteristics would depend on the geotechnical characteristics of the host rock and the *in situ* stress, both of which are site-specific, and on the excavation technique and any excavation support that is installed. We expect the damage associated with the EDZ to result in localised changes to hydrogeological properties. Creep, resulting in the self-sealing of fractures, and to some degree the excavations themselves, is likely to be important in a lower strength sedimentary host rock [16, 17]. This property is important for performance because it means that the ability of the EDZ to conduct water and contaminants would tend to decrease with time following closure of a disposal vault or tunnel. The timescale on which self-sealing occurs would depend on the physical properties of the host rock, the stress regime and the type of engineering support provided. In the case of a plastic rock and high stress regime, the process might take only a few years, but for more indurated clay rocks, such as the host rock considered here, in a lower stress regime, it could take many centuries or thousands of years, if it occurs at all. The rate and extent of self-sealing would also be influenced by the degree to which the rock has become dehydrated [16].

As soon as they have been sealed, both ILW/LLW disposal vaults and HLW/SF disposal tunnels would start to resaturate, albeit slowly, while waste is emplaced in adjacent vaults and tunnels. Given that filling a disposal vault or tunnel may take only a few months, some disposal vaults and tunnels would be sealed very early in the ~100-year operational phase. However, the low permeability of the host rock would limit the degree to which the disposal areas are able to resaturate during the operational period.

ILW/LLW

The Nagra ILW disposal concept incorporates backfilling of the voids around the disposal boxes in each ILW/LLW disposal vault and sealing the vault with an "operational" seal[28] as soon as it has been filled with waste. One of the roles of this backfill is to resist some of the rock creep that would inevitably occur and provide a space in which gas can accumulate without damaging the rest of the engineered barriers or the host rock. Rock creep in the unlined disposal vaults may make access difficult or hazardous if the vaults are left open for an extended period, and could potentially compromise the backfilling process, which is crucial to ensuring that the near-field barriers function correctly, were this left until closure. Access tunnels are left open until the disposal areas are closed, so if required, the disposal vaults could be re-opened and the backfill removed and the disposal boxes filled with waste retrieved. The presence of the concrete disposal box would simplify the process of removing the *in situ* backfill without damaging the waste packages.

[28] Temporary seals that could be removed with relatively little effort if required. They would be upgraded or replaced at closure.

Backfilling of the disposal vault would transiently increase the temperature and bring additional water into contact with the waste packages; some water would be introduced when the backfill is placed in the disposal box. This is likely to result in an increase in the rate of corrosion of reactive metals within the waste packages and a transition towards general anaerobic corrosion of the waste containers and steels within the waste packages. Hence, the overall rate of gas generation would be expected to increase. Subsequently, as the disposal vault resaturates, any oxygen would be consumed by corrosion and other, probably gas generating, processes. Alkaline conditions would develop as the incoming groundwater equilibrates with the cementitious backfill.

HLW/SF

Emplacement of the waste and bentonite buffer would completely fill the HLW/SF disposal tunnels and these would also be sealed by operational seals as soon as they are full. As water slowly enters the HLW/SF disposal tunnels, the bentonite buffer would swell and seal any gaps between both the waste package and the buffer, and the buffer and the host rock. Prompt sealing of the disposal tunnels would be required to ensure that the swelling of the bentonite buffer is resisted and that it achieves its target saturated density, which is required in order to ensure it fulfils its safety functions.

Early post-closure evolution

Following closure, the resaturation process described above would continue, and eventually complete. We expect it would take thousands of years, perhaps more than ten thousand years for the disposal areas to resaturate fully and long-term conditions to become established throughout the disposal areas [16].

During this period, the engineered barriers would degrade slowly as a result of interaction with the host rock groundwater and with each other. However, they would remain sufficiently intact that they are able to perform their intended safety functions fully. The low-permeability host rock would contribute to the durability of the EBS by providing a stable chemical environment and limiting the rate of water flow through the EBS. The majority of the degradation reactions that affect the engineered barriers are water mediated, so limiting the volume of water flowing through the near field would limit the rate of barrier degradation and the rate at which dissolved contaminants can be transported away from the waste packages. However, it is important to note that the buffers and backfills must be fully resaturated before they can function properly, and a lower than expected resaturation rate (i.e. unexpectedly low groundwater flow) may prevent the barrier functions from becoming established on the expected timescales.

ILW/LLW

High-pH conditions that limit solubility and promote sorption of key radionuclides [16] would develop in the ILW/LLW disposal vaults as the backfill interacts with the incoming groundwater. We expect the EDZ to largely self-heal during the resaturation period. The ILW/LLW wasteform would be an important barrier to the release of radionuclides during the resaturation period while the chemical barrier provided by the cementitious backfill is becoming established.

Significant volumes of gas (mostly hydrogen, but also some methane and carbon dioxide) may be generated in the ILW disposal vaults mostly by the corrosion of Magnox and aluminium present in the waste, but also from the corrosion of the various steel components and the degradation of organic materials [19]. We expect that the carbon dioxide would either dissolve or react with the cementitious backfill. The gas generation rate may be limited by the availability of water. However, we do not expect that this gas would be able to escape easily into the host rock, and some pressurisation of the disposal areas would be likely as a free gas phase develops. Any build-up of gas pressure may slow the resaturation rate and hence the gas generation rate may be self-limiting. Some free gas would be likely to enter the host rock as a result of dilation of micro-fractures [19].

The disposal vault and tunnel design would ensure that gas pressures cannot build up to the extent that they might damage the integrity of the engineered barriers or the host rock. This design would be site-specific. The high porosity backfill in the ILW/LLW vaults is intended to mitigate the localised build-up of excessive gas pressures.

During this period, mobile radionuclides as dissolved species would be released through the vents in the ILW packages, and would enter the host rock where they would migrate under diffusion. However, we would expect the majority of the radionuclide inventory to remain in the near field, either in the wasteform or sorbed to the cementitious engineered barriers. Any sorbing contaminants that are released from the near field would be retarded, and probably contained, within the host rock. The impact of organic complexing agents derived from the wastes, which can increase radionuclide mobility [20], would become progressively less significant as their concentration is reduced by physical removal and probably by microbial degradation.

An alkaline disturbed zone would form around the ILW/LLW disposal area, although its extent is likely to be limited. Reactions between cementitious water and the clay minerals of the host rock are expected to result in the precipitation of new minerals that would block the host rock porosity and result in a decrease in permeability and the sealing of any EDZ fractures that had not self-sealed as a result of creep.

HLW/SF

In the HLW/SF disposal area, swelling of the bentonite buffer would be complete, resulting in a low-permeability barrier around the waste canisters that inhibits advective flow. In addition to the swelling that occurs on resaturation, interaction with the host rock groundwater would result in some minor changes to the physical and chemical properties of the bentonite buffer. Some minerals may dissolve and others may precipitate, and ion exchange reactions would occur. Once saturated with water, chemically reducing conditions are established and the bentonite pore water is buffered to a mildly alkaline composition. They also have the potential to influence the swelling pressure of the bentonite, and hence its ability to fulfil its safety functions. Elevated temperatures may increase the rates of some of these reactions and may influence fluid movements, in particular transport of water vapour, which is an important process in the resaturating bentonite. Further details of the types of reactions that might occur and their potential impact can be found in the Near-field evolution status report [16]. In the HLW/SF disposal area, the carbon steel disposal canister would provide containment throughout the resaturation period, while the bentonite buffer becomes fully established as a low-permeability barrier, and for a considerable period thereafter.

The HLW/SF packages would give off significant quantities of heat until the short-lived component of the inventory has decayed. As a result, the temperature of the buffer and the surrounding rock may rise to several tens of degrees above ambient before slowly decreasing. The peak temperature in the near field is likely to occur within a few decades of waste emplacement, but we expect temperatures in the HLW/SF disposal area to remain above ambient for around 10,000 years [16]. However, we expect the carbon steel disposal canister to remain intact for several tens of thousands of years [15] (see below), so temperatures should have returned to ambient, or close to ambient, before the canister stops providing containment. The elevated temperatures in the HLW/SF disposal area may also result in slightly increased (up to about 10 degrees) temperatures in the ILW/LLW disposal area. We would need to take this into account when designing the layout of the disposal areas.

Despite the relatively benign chemical environment and diffusion-dominated solute transport within the bentonite buffer, corrosion of the carbon steel HLW/SF disposal canister would occur [19]. Hydrogen gas would be generated. This gas is expected to be able to escape through the bentonite without compromising its properties as a hydraulic barrier [19]. Dissolved gas is only expected to be removed by diffusion into the host rock.

The Gas status report [19] describes the processes by which free gas may be able to migrate into a lower strength sedimentary host rock once sufficient over-pressures have developed within the disposal areas.

We do not expect any release from the HLW/SF disposal tunnels while the carbon steel canisters remain intact because the waste packages would continue to provide complete containment. Once the canisters fail, the iron corrosion products and the bentonite buffer provide good substrates for sorption and would retard release from the near field [20]. The low-permeability host rock, in which diffusion dominates solute transport, would provide sufficient retardation of the transport of radionuclides that are released from the EBS, that many of these radionuclides will have decayed completely before reaching the biosphere.

Late post-closure evolution

In the very long term (many hundreds of thousands to millions of years), we do not expect the EBS to provide containment. However, the degraded EBS barriers are likely to continue to retard the release of contaminants to some degree, although by this time the vast majority of the inventory would have decayed. For example, actinides sorb strongly to iron corrosion products [20]. Retardation in the geosphere, through sorption, would now provide the main barrier to limit the rate of release to the biosphere, and the majority of radionuclides would decay within the geosphere. On these very long timescales we would need to consider the potential impact of various “natural” disturbances or transients arising from the effects of global climate events (e.g. glaciation) [17].

ILW/LLW

Many tens, or more likely hundreds, of thousands of years after closure, the engineered barriers in the ILW disposal area would begin to lose their effectiveness and would no longer be able to perform their intended safety functions fully. With time, the wasteforms and the cementitious backfill would become degraded through reaction with the host-rock groundwater, and disrupted as a result of rock creep and associated compaction [16]. Eventually the pH in the ILW/LLW disposal area would fall to a level where the chemical barrier provided by the cementitious backfill loses some of its effectiveness. However, we would include sufficient cementitious backfill in our design to ensure that the chemical barrier would remain at least partially effective on timescales of up to 1,000,000 years. Those radionuclides that have not decayed within the waste packages and near field may be released to the host rock, where their transport would be retarded by sorption. If transport is sufficiently slow, they may decay completely within the host rock. Gas would continue to be produced from the corrosion of metals, but we expect the production rate to have significantly declined by this time.

HLW/SF

Based on our understanding of likely corrosion rates under *in situ* conditions and material thicknesses, we expect the carbon steel canisters, which are protected by the bentonite buffer, to have a lifetime of several tens of thousands of years [15]. However, we note that many performance assessments that consider carbon steel canisters conservatively assume complete failure of the canister at the minimum lifetime given in the disposal system specification. In the Nagra concept, this is 10,000 years; in other concepts for lower strength sedimentary rock, the minimum design lifetime may be considerably shorter than this [28]. Should any of the waste packages fail early (i.e. at or soon after the minimum lifetime), the properties of the HLW and SF wasteforms and the bentonite buffer would limit the rate of release of the inventory [16, 20].

Once water penetrates the disposal canister, the SF and vitrified HLW would start to dissolve and release radionuclides [15]. In the case of the HLW, the majority of the inventory would have decayed by this time; slow dissolution of the glass matrix would limit the release rate of any remaining inventory. For the SF, the portion of the instant-release fraction[29] that has not decayed *in situ* would be released rapidly, and then the SF would slowly dissolve and release the remaining radionuclides [15]. We would expect the bentonite buffer to provide an effective retardation barrier to the transport of radionuclides from the failed waste package to the host rock. Any radionuclides that are released from the near field would be retarded in the geosphere. Solute transport in the host rock would be dominated by diffusion.

B5 Operational environmental safety

We describe our approach to assessing operational environmental safety in [26] and summarise it in the generic ESC main report. Once a preferred site has been identified, we will tailor the Operational Environmental Safety Assessment to that site.

B6 Post-closure environmental safety

Our confidence in the post-closure environmental safety of a GDF developed in a lower strength sedimentary host rock is based on several decades of R&D. We are building on the work of Nagra [177] and Andra [178], supplemented by our own work to understand any modifications that might be required to adapt it to UK conditions. Both the Nagra and Andra disposal and safety concepts have received international peer reviews that endorse their viability [138, 180].

Both the ILW/LLW and the HLW/SF disposal concepts provide multiple barriers to the release of the inventory. We have a good understanding of the major chemical and physical processes that would provide containment of the inventory and retard any radionuclides that are released from the wasteforms. Our R&D programme will be developed to address the prioritised knowledge gaps that have been identified in the research status reports (see section 6.3 of the main report).

The host rock permeability would be sufficiently low that solute transport would be dominated by diffusion. Many radionuclides would be sufficiently contained that they would decay completely before reaching the biosphere. The nature of the overlying sequence would be of less importance to the overall safety case because the barrier provided by the host rock would be so effective. However, the overlying sequence is likely to include additional further low-permeability units in which solute transport would be dominated by diffusion. Thus, we can be confident that the geosphere would provide an effective barrier to radionuclide release. We would design our site characterisation programme to gather the evidence needed to test this assertion. For example, Nagra has been able to show that the movement of natural tracers in the Opalinus Clay is diffusion-controlled.

The engineered barriers provided by the wasteforms, the cementitious backfill in the ILW vaults, and the bentonite buffer in the HLW/SF disposal tunnels also limit the rate of radionuclide release. We have a good understanding of the degradation processes that would occur in the engineered materials, and the majority of these are water mediated. A key feature of the illustrative geological disposal concept examples is the low groundwater flow rate through the disposal areas, which means that solute transport would be diffusion-controlled, and hence the rate of degradation reactions would be limited by the rate at which the reactants/reaction products can diffuse to/from the reaction site. Therefore, we expect the barriers in the EBS to be extremely long-lived.

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[29] The portion of the inventory (fission and neutron activation products) that has segregated to the grain boundaries in the wasteform and may be released rapidly upon exposure to groundwater.

The timescale on which we envisage a UK GDF would be built is such that we expect that practical operational experience would have been gained overseas before we finalise our detailed design. We would learn from this experience to refine our design and procedures for emplacing the EBS and waste.

Appendix C Illustrative geological disposal concept examples for evaporite host rock as applied to the UK

C1 Introduction

This appendix illustrates the case of a GDF constructed in a geological environment characterised by an evaporite host rock (see Table 4.2 of the main report and the Geosphere status report [17] for a summary of the characteristics of this rock type). Evaporites are formed by the evaporation of water from water bodies containing dissolved salts, the most common types being anhydrite (anhydrous calcium sulphate) and halite (rock salt). The strength of evaporites varies markedly with composition and the presence or absence of impurities. However, a key characteristic of evaporites from the point of view of a GDF is their ability to creep and seal voids. This property also varies widely with factors such as composition and purity, and temperature.

For an evaporite host rock to be preserved and sufficiently stable to host a GDF, it must be isolated from formations containing flowing groundwater, most notably the ground surface, by low-permeability formations. If this condition is not met, the host rock would either not have been preserved on geological timescales or it would have been screened from further consideration on account of it not being sufficiently stable in the long term.

As explained in the main report, the UK Government sees no case for having more than one GDF if one facility can be developed to provide suitable, safe containment for the entire inventory. We are therefore describing in this appendix a conceptual layout for a GDF in an evaporite that is based on two distinct disposal areas implementing different disposal concepts, one for unshielded intermediate-level waste (UILW), shielded intermediate-level waste (SILW), low-level waste (LLW) and depleted, natural and low enriched uranium (DNLEU), should it be declared as waste, and the other for high-level waste (HLW), and spent fuel (SF), highly enriched uranium (HEU) and plutonium (Pu), should these be declared as wastes. These disposal areas would share common access ways and surface facilities, but would be separated from each other to ensure that the impact of interactions between the two disposal areas is sufficiently small that it does not compromise the overall performance of a GDF.

The assumptions that underpin the generic DSSC are set out in [32]. As discussed in [32] and shown in Table 4.3 of the main report, we are basing our generic designs for an evaporite host rock on the US Department of Energy's (DOE's) Waste Isolation Pilot Plant (WIPP) concept [181] for ILW/LLW, and on the concepts developed by the German Company for the Construction and Operation of Waste Repositories (DBE) for HLW and SF [182]. Should we find ourselves considering a candidate site with an evaporite host rock, we would develop site-specific concepts, which may, or may not, be based on these disposal concepts.

In this appendix, where we refer to the ILW/LLW disposal area or vaults, we mean the part of a GDF in which the disposal concept based on the WIPP concept is implemented, that is the disposal area for UILW, SILW, LLW and DNLEU. When we refer to the HLW/SF disposal area or tunnels, we mean the part of a GDF in which the disposal concept based on the DBE disposal concept is implemented, that is the disposal area for HLW, and SF, HEU and Pu. However, discussion focuses on ILW/LLW and HLW/SF because, to date, this is where our conceptual design and assessment work has focused.

One of the purposes of this appendix is to illustrate how the different components of our work programme fit together to provide knowledge about the environmental safety of a GDF. It therefore draws together information about the characteristics and likely evolution of a GDF developed in an evaporite host rock from a number of sources, most notably the Package evolution [15], Near-field evolution [16], Geosphere [17], Gas [19] and Radionuclide behaviour [20] status reports, which describe our current state of knowledge and understanding. These status reports provide details about the characteristics, safety functions and expected evolution of the different barriers in the multi-barrier system.

C2 Geological environment

In this example, the host rock is a massive bedded evaporite formation[30] that is overlain by a sedimentary sequence and near-surface zone of weathered rock and recent (Quaternary) deposits. The presence of a massive evaporite that is stable enough to be considered as a potential host rock implies that there is at least one significant low-permeability unit between the evaporite and the ground surface to protect the evaporite from dissolution by recharging groundwater.

Evidence from hydrocarbon investigations suggests that faults are likely to be of low permeability (sealing) where they intersect evaporites or low-permeability formations. The evaporite permeability is likely to be so low that it is essentially impermeable and the porosity is expected to be only a few percent. Such groundwater as is present is likely to be held within isolated pores, so there may be little accessible water in any one location. However, shaley interbeds and other impurities could result in zones of slightly higher, although still very low, permeability. The bedded nature of the rocks means that the permeability parallel to bedding may be significantly greater than the permeability perpendicular to bedding [17]. The Geosphere and Gas status reports [17, 19] describe the properties of the evaporite host rock and the overlying sedimentary sequence in more detail. The low clay content, except perhaps in shaley interbeds, means that most evaporites are non-sorbing [20].

The engineering properties of evaporite formations are crucially dependent on factors such as mineralogy, level of impurities and temperature. We know that the strength and the creep behaviour of this host rock are likely to be difficult to predict in advance and this uncertainty would be factored into our design and our site characterisation programme. Evaporite does have the advantage of sealing faults and other discontinuities, although this sealing property (creep) has the potential to lead to challenges when we try to maintain underground openings for significant periods.

Such groundwater as is present in the host rock would have a high salinity (brine with total dissolved solids in the range 180,000 to 300,000 milligrams per litre) and a composition that depends on the exact composition of the evaporite concerned. Groundwaters in overlying rocks are likely to be less saline.[17].

An important property of evaporites is that they typically have a higher thermal conductivity than most other rocks. This property means that evaporites are more effective than most other rocks at conducting heat away from a source of heat, such as a waste package, but any thermal perturbation is likely to be seen at greater distances from the heat source than in other rock types [17].

Head gradients driving groundwater flow are likely to be low because, in the UK, evaporites are usually found in basin environments with little topographic relief. As a result of the combination of extremely low permeability and low head gradient, diffusion is likely to dominate solute transport in the host rock and in the low-permeability rocks in the overlying sequence. However, advection may dominate in any more permeable minor aquifer units in the overlying sequence. Transport processes are discussed in more detail in the Geosphere and Radionuclide behaviour status reports [17, 20].

The waste packages would contain some free water, for example in the cement grouts, and therefore some gas would be generated in a GDF constructed in an evaporite host rock even though little or no groundwater would enter the disposal vaults. It is likely to be difficult for any gas generated within the disposal areas to migrate into the host rock. For this

[30] Large salt domes, the preferred host rock in Germany, are not present onshore in the UK.

reason, evaporite formations are a common choice for underground gas storage [183] and are often found as cap rocks in hydrocarbon reservoirs. We would need to design the disposal areas to ensure that the gas pressure cannot damage the integrity of the host rock. Should any gas escape from the host rock, we expect the overlying sedimentary sequence to trap any gas that escapes from the disposal areas and the host rock before it reaches the surface environment. The migration of gas through an evaporite is discussed in more detail in the Gas status report [19].

We expect that natural conditions in the host rock at disposal depths would evolve only slowly because very low-permeability rocks such as evaporites respond only slowly to changes in external driving forces (tectonic events, glaciation). Indeed, conditions at disposal depths may still be evolving in response to the effects of geological events that occurred many hundreds of thousands years ago, or even more than a million years ago. This property also means that we expect the disturbance associated with the development of and operations within the disposal areas to be relatively localised. Groundwater heads in the host rock and over/underlying strata may not be in equilibrium with surrounding rocks, meaning that, in the absence of site-specific data, it may be difficult to predict flow directions in any transmissive parts of the closed disposal areas. Large-scale environmental changes such as would be associated with glaciation are unlikely to affect conditions at disposal depths [17].

C3 Engineered barrier system

The evaporite host rock provides an important barrier to the release of the inventory for both disposal areas. Evaporite deposits typically have the mechanical strength to support relatively large underground excavations (i.e. vault-type structures) without engineered support (although rock creep would cause void spaces to close over time). This reduces the range and quantity of materials that would need to be introduced into a GDF for structural support.

A key feature of disposal concepts that have been developed for implementation in evaporites is that the host rock essentially provides complete containment for a disposal system that evolves ‘normally’. Therefore, the role of the EBS in an evaporite host rock is largely to ensure that the wastes can be emplaced safely and, with time, we expect the backfills and buffers, which are often based on crushed host rock, to become indistinguishable from the host rock. This reduced emphasis on the EBS to contain and retard contaminants increases the importance of the engineered tunnel and shaft seals as primary engineered barriers, because the access tunnels and shafts are potential pathways through the containment barrier provided by the host rock. These seals are usually multi-barrier systems in their own right and are generally more complex than the seals proposed for use in other host rock types. The creep behaviour of evaporites also significantly affects the ease of waste retrievability in this geological environment.

This section summarises the application of the illustrative geological disposal concept examples for a higher strength host rock presented in Table 4.3 to the UK, as set out in our Generic disposal facility designs report [24].

ILW/LLW

At the WIPP, the US equivalent of SILW packages are stacked in the disposal vaults and the US equivalent of UILW packages are inserted into horizontal boreholes drilled into the sides of the disposal rooms [16]. A sack of magnesium oxide is placed on the top of each waste stack. The relative numbers of SILW and UILW waste packages in the UK inventory would probably preclude this emplacement arrangement because, compared to the WIPP, we would need to dispose of relatively more UILW packages and relatively fewer SILW packages. Therefore, in our example, all of the waste packages would be stacked in disposal vaults and sacks of magnesium oxide would be placed on top of the waste stacks. These sacks are designed to burst as a result of creep closure and swelling of the contents as they absorb water, and to fill the void space surrounding the waste packages. The magnesium oxide buffer is intended to react with any moisture during the operational period. Post-closure, the magnesium oxide would react with any carbon dioxide that is produced and condition the pH of any brines that are present for several tens of thousands of years [16].

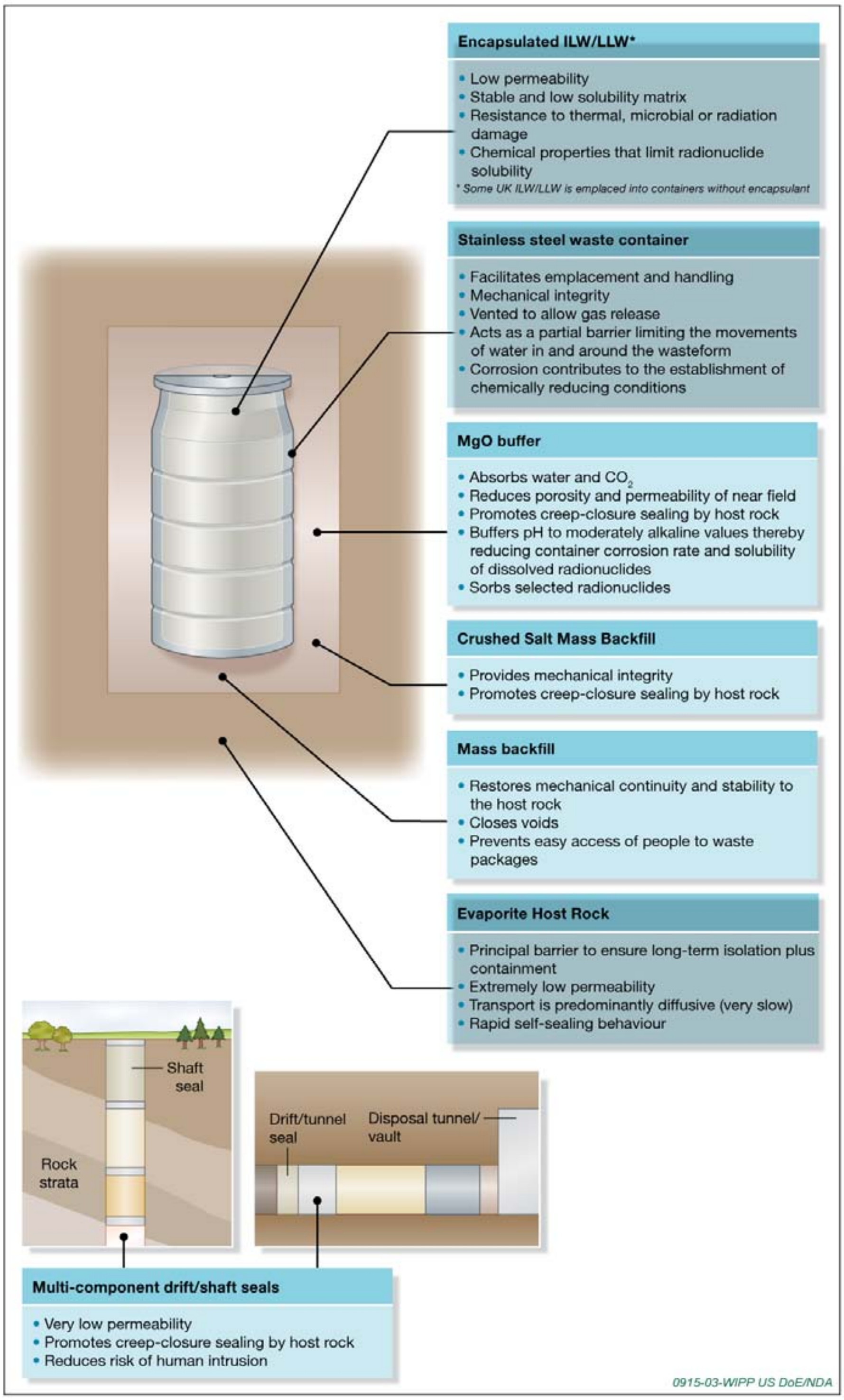
We anticipate that the disposal vaults in the ILW/LLW disposal area would be about 10 metres wide, 5 metres high, and a few hundred metres long. The separation between the vaults would be about 30 metres. Waste packages would be emplaced using a remotely operated stacker truck. We do not envisage that any backfilling would be necessary because the spaces between the waste packages and the vault walls would be small, the magnesium oxide buffer would swell, and the salt would creep and fill any voids around the packages. However, crushed host rock would be used as a backfill material if required.

In this disposal concept, the host rock would provide a long-lived containment barrier, and the overlying sedimentary rocks would retard the transport of any contaminants that are eventually released from the host rock. The cementitious wasteform retards the release of radionuclides from the waste. Most of the ILW packages are vented to prevent the build-up of excessive gas pressures. We assume that the stainless steel waste containers would provide only limited containment during the post-closure period because, even at closure, the container is not fully 'intact'. The vent allows the release of gaseous radionuclide during transport and operations and the release of both gaseous and dissolved radionuclides after closure, even if there is no corrosion or other degradation of the container. However, the waste containers do play an important role in containing and protecting the wasteform during the operational period. The magnesium oxide backfill would fulfil a number of different roles. It would buffer the pH within the disposal vault to alkaline conditions, which would limit actinide solubility and promote sorption to the magnesium oxide, as well as limiting corrosion rates. It would also fill voids and thereby provide mechanical stability and a degree of physical containment, which would accelerate the self-sealing of fractures around the disposal vault [16].

The different engineered barriers and the processes and properties that contribute to them fulfilling their safety functions are illustrated in Figure C1.

Figure C1 The barriers to radionuclide release for the ILW/LLW concept in an evaporite host rock

The figure shows the barriers to radionuclide release from the ILW/LLW disposal area in an evaporite host rock and the processes and properties that contribute to their safety functions. The Near-field evolution status report [16] gives further details of the materials, processes and properties shown.



HLW/SF

In the German disposal concept for HLW, the disposal packages are placed in boreholes that penetrate several hundred metres into a salt dome. However, there are no major salt domes onshore in the UK. In the German disposal concept for SF, fully shielded 'transport, storage and disposal' containers are placed in the disposal tunnels, which are then backfilled with crushed salt.

A variant on the German concepts that might be practicable for implementation in the UK is a disposal concept in which the disposal packages are non-shielding steel canisters of the type used for the lower strength sedimentary host rock HLW/SF concept (Appendix B). In this disposal concept, which we have considered as our illustrative geological disposal concept example for HLW and SF disposal in an evaporite, the waste packages would be placed in small-diameter (a few metres) disposal tunnels, separated from each other by a few metres. The voids around the packages would be backfilled with crushed host rock.

Engineered barriers would be provided by the carbon steel disposal canister and the wasteform, and also by the crushed rock backfill. The disposal canister would provide containment, and once it has been breached, the corrosion products would be a good substrate for sorption of key radionuclides such as actinides. The host rock would provide a long-lived containment barrier, and the overlying sedimentary sequence retards the transport of any contaminants released through the host rock. The crushed rock backfill would compact and creep as the walls around a GDF converge owing to the plasticity of evaporites, and would eventually consolidate to form an almost continuous low-permeability barrier contiguous with the surrounding host rock.

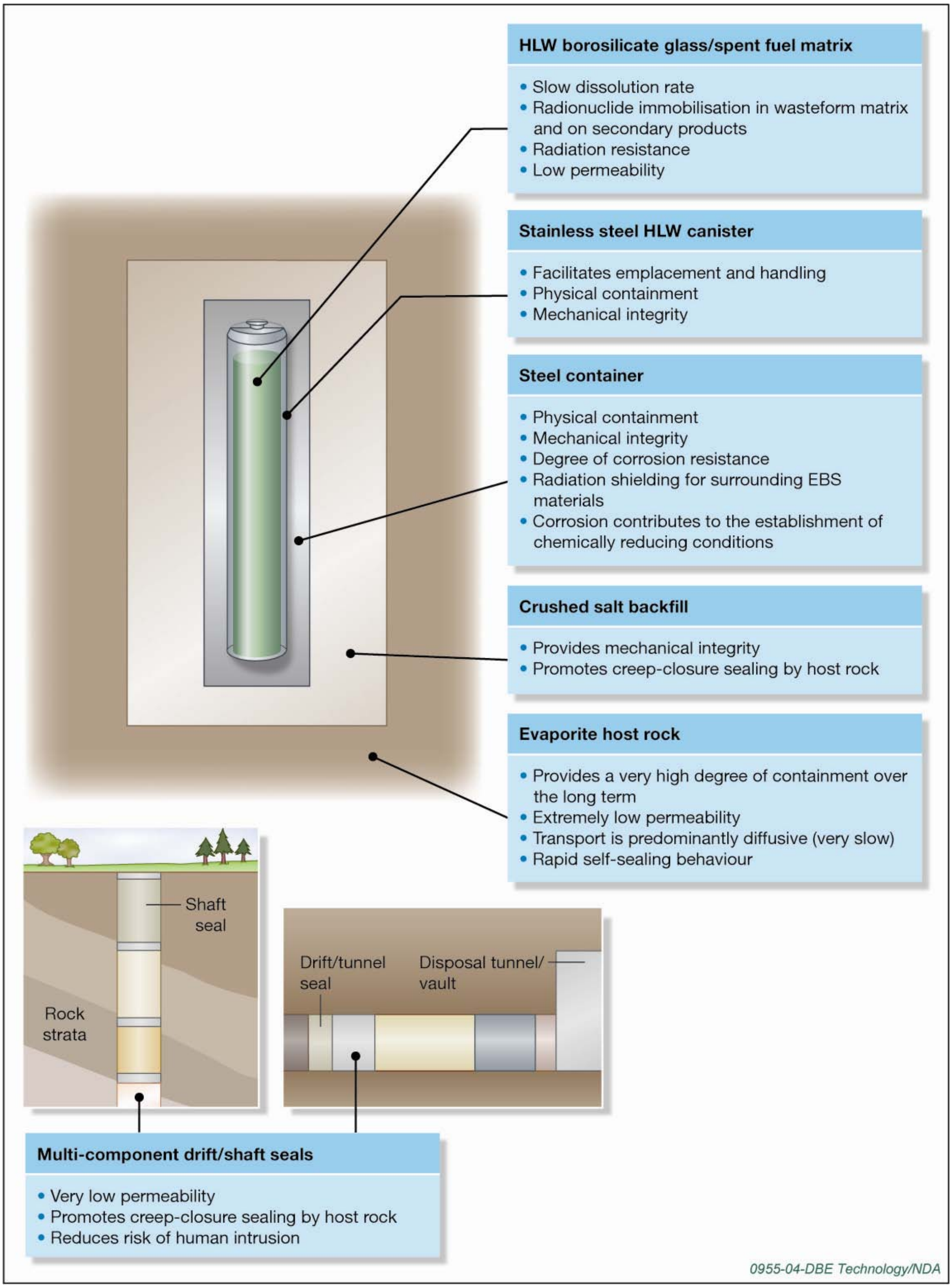
The different engineered barriers and the processes and properties that contribute to them fulfilling their safety functions are illustrated in Figure C2.

GDF layout

We have developed a conceptual layout of the underground facilities based on the illustrative geological disposal concept examples for an evaporite host rock [24]. When considering a real candidate site, the layout would be optimised to make best use of the conditions and space available at the site. The layout is able to accommodate the Derived Inventory reference case, and has a total footprint of about 9 square kilometres.

Figure C2 The barriers to radionuclide release for the HLW/SF disposal concept in an evaporite host rock

The figure shows the barriers to radionuclide release from the HLW/SF disposal area in an evaporite host rock and the processes and properties that contribute to their safety functions. The Near-field evolution status report [16] gives further details of the materials, processes and properties shown.



C4 Expected evolution

In this section we describe the expected normal evolution of the geological disposal system for the illustrative geological disposal concept examples for the evaporite host rock. This description draws on material in the research status reports and in our Generic disposal facility designs report [24]. The important processes and the timescales on which we expect them to be important are illustrated in Figure C3 for the ILW/LLW disposal area and in Figure C4 for the HLW/SF disposal area.

Excavation and fitting out of each individual disposal vault or disposal tunnel could take up to several years, and we expect that a few hundred ILW/LLW disposal vaults and about a hundred HLF/SF disposal tunnels would be required to accommodate the Derived Inventory reference case. It would therefore take many decades to construct a GDF were all of the disposal vaults and disposal tunnels to be constructed before waste emplacement begins. Such an approach is likely to be impractical so we expect that excavation and fitting out of new disposal areas would occur in parallel with the emplacement of waste in other areas of a GDF (i.e. disposal vaults and tunnels would be constructed on a 'just-in-time' basis). This strategy would also allow us, if necessary, to refine our designs to take account of earlier experience and changes in the characteristics and volumes of waste to be disposed. Further details of how we would ensure this would be achieved safely can be found in the OSC [6]. Therefore, at any given time, different processes would be occurring in different vaults and disposal tunnels as they progress through excavation and operations and finally to closure.

We describe the expected evolution of a GDF in an evaporite host rock below in terms of three periods:

1. Construction and operational period. We assume that the operational phase of a GDF would last for about 100 years (starting from about 2040), and that a GDF would be closed as soon as waste emplacement is complete. We also assume that there would not be a subsequent period of extended retrievability when the disposal areas are left open (i.e., unbackfilled or unsealed).
2. Early post-closure period (establishment of long-term conditions and engineered barriers fulfil their safety functions, to a few tens of thousands of years, depending on disposal concept).
3. Late post-closure period (engineered barriers begin to lose their effectiveness and would no longer be able to fulfil their safety functions fully, to one million years and more).

These periods are defined primarily in terms of the events and processes that would occur in the future, rather than in terms of years, because the different characteristics of the two disposal areas mean that they would evolve at different rates.

Some of the processes described below would begin soon after the waste has been emplaced and may occur on a timescale that is shorter than the length of the operational period (about 100 years). Therefore, technically, these processes may occur during the operational period in some disposal vaults or tunnels and at the start of the post-closure period in others. In order to prevent repetition, such processes are only described once. For each period, processes that occur in both disposal areas are described first, then processes that are specific to the ILW/LLW disposal area, and finally processes that are specific to the HLW/SF disposal area.

The description draws heavily on material discussed in the research status reports, and further details of the processes described may be found in these reports.

Figure C3 The evolution of the ILW disposal area in the illustrative geological disposal concept example for evaporite

The time axis starts from the point at which the disposal vault is excavated. Figure taken from the Near-field evolution status report [16], which gives further details of the processes and events shown.

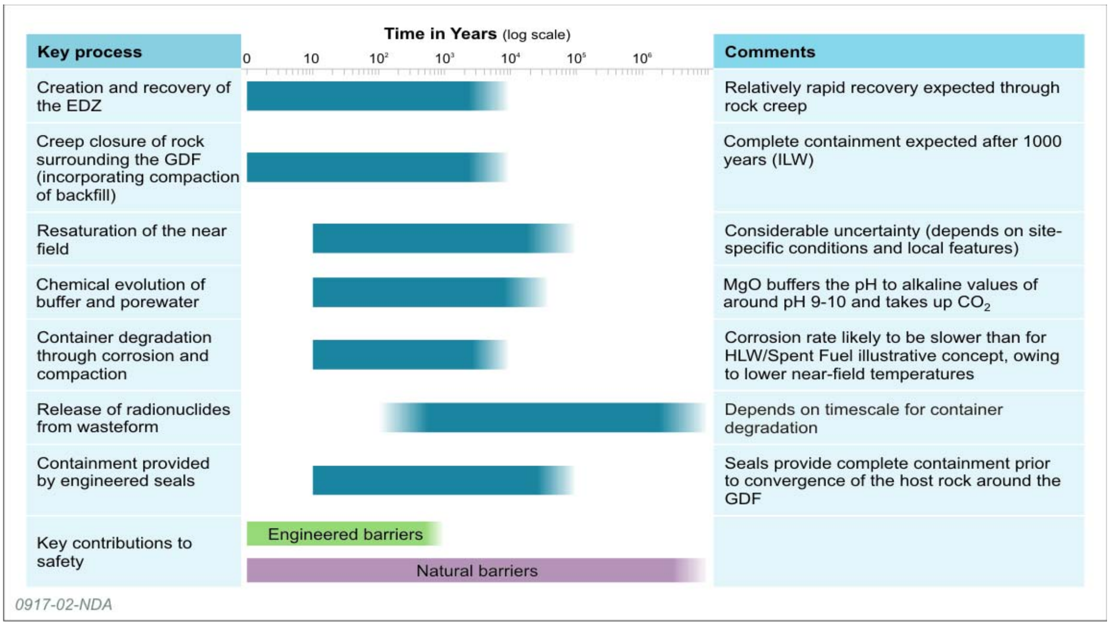
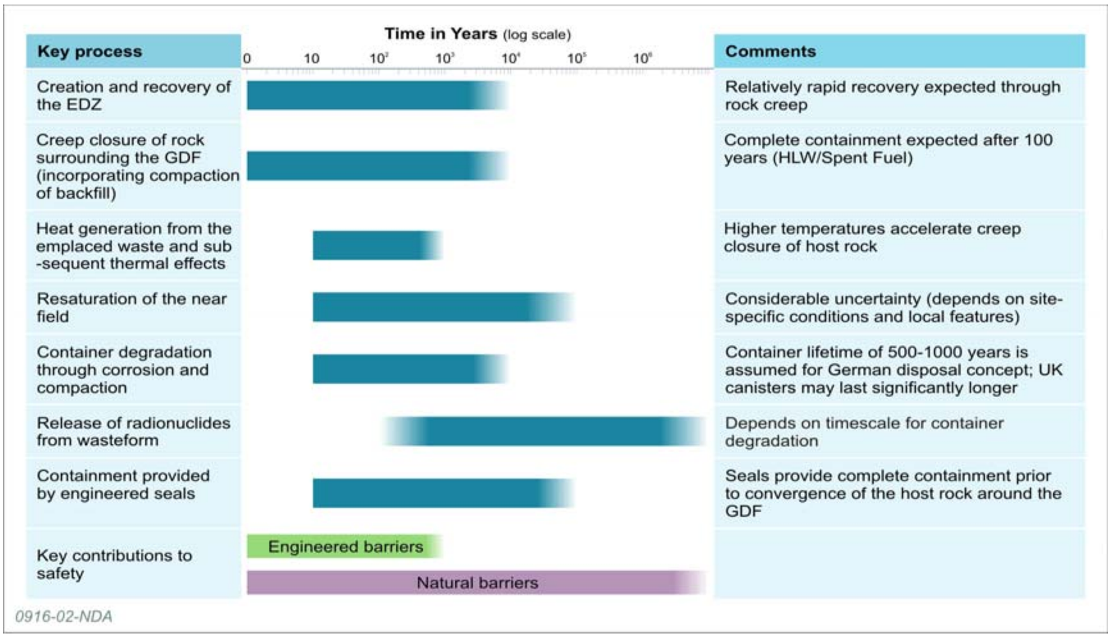


Figure C4 The evolution of the HLW/SF disposal area in the illustrative geological disposal concept example for evaporite

The time axis starts from the point at which the disposal tunnel is excavated. Figure taken from the Near-field evolution status report [16], which gives further details of the processes and events shown.



Evolution during construction and operation of a GDF

Construction activities would lead to the formation of an excavation disturbed zone (EDZ) around the excavations as a result of changes in the stress fields in the rock, which would lead to brittle fracturing. This fracturing, which is not a feature of the intact rock, is likely to result in the EDZ being a zone of enhanced permeability. Rock spalling may also occur during excavation and/or construction or during the operational period. The EDZ is likely to extend no more than a few metres from the excavated region [16], and its characteristics would depend on the geotechnical characteristics of the host rock and the *in situ* stress, both of which are site-specific, and on the excavation technique and any excavation support that is installed.

Salt creep would begin immediately following excavation, driven by the differential stress due to the creation of void spaces. In some evaporites, creep may be significant on timescales of months to a few years, and creep processes may place severe constraints on GDF layout and operational procedures. In other examples, creep may be extremely slow and we may be able to construct large vaults that would remain stable for many decades or even centuries. Scraping of vault walls, ceilings and floors would be necessary during construction and operations to maintain the excavations until closure. We would expect the host rock to start to creep as soon as the disposal vaults and disposal tunnels are excavated and to begin to seal any voids around the waste packages as soon as these are emplaced. The rate of creep, and hence the length of time for which a disposal vault or disposal tunnel can be kept open, may be an important factor in determining the operational life, and hence length, of the disposal vaults and tunnels. For this reason, it may be necessary to seal disposal vaults and disposal tunnels as soon as waste emplacement in them has been completed. Operational seals[31] would be placed at the entrances to disposal vaults and tunnels.

'Resaturation' is a potentially misleading term in an evaporite host rock where there would be minimal movement of water (brine). However, it would begin as soon as the individual disposal vaults and disposal tunnels are sealed. Such 'water movement' as occurs would probably be driven by the creep processes in the rock. Brine pockets may be important and their movement may be driven by thermal gradients [16]. The extremely low permeability of the host rock means that we would expect the 'resaturation' period to last for many tens of thousands of years or longer [17, 19].

ILW/LLW

In the ILW/LLW disposal vaults, the magnesium oxide buffer would start to absorb moisture and carbon dioxide as soon as it is emplaced, and would buffer the pH of the near field. In the WIPP, these processes are initially limited by the polypropylene 'supersacks' used to emplace the buffer/backfill material. Ventilation of the excavations during construction and operation would help to maintain the function of the buffer prior to sealing of the disposal areas, by removing some of the moisture and carbon dioxide from the disposal vault.

During construction and operation, any heat generated in the disposal areas would be dissipated by ventilation and by heat transfer through the surrounding host rock. Oxidising conditions would predominate in the near field during this phase, but these would quickly revert to reducing conditions once the disposal vault or disposal tunnel is closed [16].

The extremely low permeability of the host rock means that the amount of water that comes into contact with the waste packages is likely to be very small. However, we would expect some gas to be generated as a result of degradation of the wastes and corrosion of the waste packages by consuming water already present within the packages at the time of disposal. The gas generation rate is likely to be limited by the low water availability [19].

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[31] Temporary seals that could be removed with relatively little effort if required. They would be upgraded or replaced at closure.

Any gas that enters the unsealed parts of the excavations during the operational period would be filtered by the ventilation system.

HLW/SF

As soon as the waste packages and surrounding backfill have been emplaced, rock creep would begin to seal any voids in the HLW/SF disposal tunnels. The carbon steel canisters would start to corrode, but the low availability of water would mean that the corrosion rate would be extremely low and very little gas would be generated.

Early post-closure evolution

Rock creep around the excavations would continue, and the geological barrier would provide increased containment in its own right as time progresses and the stresses associated with excavation and waste emplacement dissipate. Relatively rapid recovery of the EDZ is expected through rock creep, which would enable fractures to heal. The recovery of the EDZ around the disposal shaft would be aided by the resistance of rigid components of the shaft sealing system. The EDZ around the shaft would therefore not be expected to provide a continuous pathway for fluid flow. Later, the rigid resistance of compacted backfill in the ILW disposal vaults and the carbon steel disposal canister in the HLW/SF disposal tunnels would also encourage healing of the fractures surrounding them [16].

The impermeable components of the shaft seals (for example, concrete, clay and asphalt) would be particularly important during the early part of the post-closure period in preventing inflow of water from the surface to the disposal areas, as well as in preventing radionuclide transport to the accessible environment. Compaction and creep of crushed host rock components of tunnel and shaft seals would render their permeability similar to that of the host rock within approximately 200 years [16]. This would prevent water circulation in the disposal areas and isolate them from the accessible environment.

ILW/LLW

Rock creep and the evolution of excavations would cause the magnesium oxide sacks in the ILW/LLW disposal vaults to burst and spread the magnesium oxide powder around the waste packages. The disposal areas would be gradually compacted as further salt creep occurs. The magnesium oxide would continue to fulfil its chemical buffering functions through the early post-closure phase, minimising free water in the near field and protecting the waste packages from chemical degradation. Some waste packages may be breached as the evolution of excavations and compaction of the magnesium oxide buffer arising from rock creep continue [16].

As the region around the disposal areas re-saturates, such fluid flow as was occurring into the near field would slow and eventually stop. Gas (dominantly hydrogen, but also some carbon dioxide and methane in the ILW/LLW disposal vaults) would be generated, mostly by the corrosion of Magnox and aluminium and the degradation of organic materials present in the ILW, but also from the corrosion of the various steel components, including the HLW/SF disposal canisters. It is likely to be difficult for locally generated gas to escape from the disposal areas and significant overpressures could develop were sufficient gas to be generated. However, we expect such low-permeability host rock to severely restrict the availability of water, meaning that the rate of gas generation should be very low. The magnesium oxide would contribute to the low gas generation rate in the ILW/LLW disposal vaults by absorbing water, and we would expect it to react with any carbon dioxide that is produced. The magnesium oxide backfill therefore helps to reduce the gas generation rate, but it does not react with hydrogen, which is the main bulk gas produced by corrosion of metallic wastes and of the waste packages.

Gas pressurisation may cause fracturing of any more brittle layers within the host rock, perhaps those containing impurities or shaley interbeds. This process may then drive the outwards flow of brine from the disposal vaults and tunnels. Gas would dissolve in such water as is present and, if the pressures are high enough, may enter the host rock through a process of microfracturing [19]. The behaviour of evaporites at high gas pressures is relatively well understood from work on underground gas storage [183].

HLW/SF

The crushed host rock backfill around the HLW/SF disposal canisters would compact under the influence of surrounding rock creep. Over time, porosity in the crushed host rock would be eliminated and eventually the backfill would assume the same properties as the surrounding rock, merging with it to provide a continuous barrier of low-permeability evaporite. The carbon steel disposal canisters may have sufficient strength to remain intact for a significant time after closure.

The HLW/SF packages would give off significant quantities of heat until the short-lived component of the inventory has decayed. As a result, the temperature of the backfill and rock around the disposal canister may rise to several tens of degrees above ambient before slowly decreasing. The peak temperature in the EBS is likely to occur within a few decades of waste emplacement, but we would expect temperatures in the HLW/SF disposal area to remain above ambient for around 10,000 years [16]. Thermal considerations would be a factor in determining the separation distance between the ILW/LLW and HLW/SF disposal areas. We note that the thermal conductivity is significantly higher for most evaporites than for most other rock types. The rate of rock creep increases with temperature and we expect it to be significantly accelerated in proximity to the heat-generating wastes for ~100 years after emplacement [16].

Late post-closure evolution

After several thousands of years, the engineered barriers are likely to be significantly degraded. Waste packages and disposal canisters may be breached owing to a combination of compaction and various chemical degradation processes, including corrosion. Radionuclides would be released by diffusion from the wasteforms [15] to the surrounding evaporite host rock. However, by this time, complete containment of the waste would be provided by the host rock. No radionuclide release from the host rock is expected for a normal evolution (undisturbed) performance scenario.

C5 Operational environmental safety

We describe our approach to assessing operational environmental safety in [26] and summarise it in the generic ESC main report. Once a preferred site has been identified, we would tailor the Operational Environmental Safety Assessment to that site.

C6 Post-closure environmental safety

Our confidence in the post-closure environmental safety of a GDF developed in an evaporite host rock is based on several decades of R&D. We are building on the work of the DOE and DBE, supplemented by our own work to understand any modifications that might be required to adapt the concepts to UK conditions. The safety case for the WIPP facility has been accepted by the regulatory authorities and the facility has been accepting waste since 1999. The successful implementation of the WIPP demonstrates the feasibility of geological disposal in bedded salt deposits, and underpinning R&D provides confidence in the long-term safety provided by disposal in an evaporite host rock.

In both of the disposal concepts the wasteform and the extremely low-permeability host rock are barriers to the release of the inventory to the biosphere. We have a good understanding of the major chemical and physical processes that would provide containment of the inventory and retard any radionuclides that are released from the wasteforms. Our R&D programme will be developed to address the prioritised knowledge gaps that have been identified in the research status reports (see Section 6.3 of the main report).

The host rock is an important barrier to the release of radionuclides to the biosphere. The evaporite host rock is essentially impermeable and we can be confident that solute transport vertically through the host rock would be dominated by diffusion. We expect that there would be at least 50 metres of intact host rock above the disposal areas. For typical diffusivities, it would take more than 100,000 years for radionuclides to diffuse out of the host rock in significant quantities. Consistent with the long-term stability of the host rock, the cover sequence would contain at least one further low-permeability unit in which solute transport would be dominated by diffusion. Thus, we can be confident that the geosphere would provide an effective barrier to radionuclide release. We would design our site characterisation programme to gather the evidence needed to test this assertion.

Creep closure ensures that the waste packages are completely encapsulated within a low-permeability matrix. Creep would also assist in the sealing of the access shafts and tunnels.

The engineered barriers are provided by the wasteforms and the steel overpacks surrounding the HLW/SF; any backfill or buffer becomes indistinguishable from the host rock. We have a good understanding of the degradation processes that would occur in the engineered materials. We do not expect most cementitious materials (the ILW/LLW wasteforms) to be particularly durable when in contact with brine, although we note that specialist cement-based materials, such as saltcrete, have been developed for use in highly saline environments such as this. In any case, we would expect degradation reactions to be slow because they are likely to be controlled by the time required for water and solutes to diffuse to or from the reaction sites. However, the stresses associated with creep closure may rupture some of the waste packages [16].

Evaporites are a resource, and so a GDF developed in an evaporite host rock might be more vulnerable to human intrusion than a GDF developed in other host rock types. However, in developing the Managing Radioactive Waste Safely (MRWS) 'sub surface unsuitability' screening test [4], the UK Government decided that evaporites are too widely distributed for the intrusion risk to be significant enough to justify their exclusion as potential host rocks. We expect gas pressures in the disposal areas to potentially be higher in a facility constructed in an evaporite host rock than in a GDF constructed in other types of host rock. Penetration of a pressurised disposal vault or tunnel by a borehole could lead to a significant hazard, which we have started to address in our generic work on human intrusion [27].

The timescale on which we envisage a UK GDF would be built is such that we would be able to build on the practical experience gained from decades of operations at WIPP before we finalise our detailed design. We would learn from this experience to refine our design and procedures for emplacing the EBS and waste.

Appendix D Related international and national legislation, guidance and obligations

D1 International context

Certain international treaty obligations affect UK radioactive waste management policy, including obligations stemming from membership of the European Union, obligations under the 1993 Oslo and Paris Convention on the Protection of the Marine Environment of the North East Atlantic (OSPAR Convention), and obligations under the IAEA-sponsored Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management. Policy and regulation in the UK also takes account of guidance from the IAEA and the International Commission on Radiological Protection (ICRP). The UK Government's 2008 MRWS White Paper [4] and the GRA [1] are consistent with these international treaties and guidance. Therefore, compliance with the GRA ensures compliance with the existing international expectations for the environmental safety of a GDF. A summary is provided here.

D1.1 European legislation

The Euratom Treaty was established in 1957 to promote the peaceful use of nuclear technology in Europe. Under Article 37 of the Treaty, Member States have to give the European Commission sufficient information about any plans to dispose of radioactive waste to allow the Commission to decide whether the plans could cause radioactive contamination of the water, soil or airspace of another Member State. This information must be provided before the competent authority of the Member State concerned authorises the disposal of the waste.

Under Article 33 of the Euratom Treaty, Member States have to implement appropriate provisions to ensure compliance with the Basic Safety Standards established under Article 31. In order to meet this requirement, Basic Safety Standards for the protection of the health of workers and the general public against the dangers arising from ionising radiation have been set out in various directives since 1959. The most recent is the Basic Safety Standards Directive [47]. In the regulation of radioactive waste disposal, the environment agencies comply with the requirements of the Basic Safety Standards Directive and the implementing national regulations through the EPR 2010 environmental permitting process in England and Wales [2], and through the RSA 93 authorisation process in Scotland and Northern Ireland [3].

D1.2 OSPAR Convention

The OSPAR Convention guides international co-operation on protecting the marine environment of the northeast Atlantic. The OSPAR Radioactive Substances Strategy [48] seeks progressive and substantial reductions of discharges, emissions and losses of radioactive substances to the marine environment. By 2020, it aims to achieve concentrations near to natural background levels in the marine environment for naturally occurring radioactive substances and close to zero for releases of artificial radioactive substances.

In 2002, the Department for Environment, Food and Rural Affairs (Defra) and the devolved administrations published the "UK Strategy for Radioactive Discharges 2001–2020" [184], which set out how the UK would implement the OSPAR Radioactive Substances Strategy. In 2009, the Department of Energy and Climate Change (DECC) and the devolved administrations published an updated "UK Strategy for Radioactive Discharges" [185], which builds on the initial UK Strategy and expands its scope to include aerial, as well as liquid discharges, from decommissioning as well as operational activities, and from the non-nuclear as well as the nuclear industry sectors. The 2009 Strategy sets out the radiological, environmental and other principles that the regulatory bodies will apply when setting discharge permits or authorisations. A discharge permit is not required for the disposal of radioactive waste at a GDF, but is needed during the operational period of a GDF for activities associated with interim storage, handling and emplacement of the wastes. We are required to demonstrate the application of Best Available Techniques to manage any discharges during operations (see Section 3.1.3 and Section D2.7).

D1.3 Joint Convention and the IAEA

The IAEA is the world's main intergovernmental forum for scientific and technical co-operation in the nuclear field. Its pronouncements have no legal jurisdiction. However, Member countries commit themselves to complying with standards and recommendations. The UK is a Contracting Party to the IAEA-sponsored Joint Convention on the Safety of Spent Nuclear Fuel and the Safety of Radioactive Waste Management [49]. The Joint Convention is effectively a Treaty by which the Contracting Parties agree to be legally bound under international law to carry our their obligations as defined in the Convention. The Convention includes measures covering agreed practice for radioactive waste management and sustainable development.

The IAEA produces standards relating to radioactive waste safety (i.e. waste management, waste treatment and safety of disposal facilities). These standards, and the safety guides that support them, implement the IAEA's fundamental safety objective and safety principles for radiological protection (Table D1, [50]). Of particular relevance to the ESC is the IAEA safety standard setting out the safety requirements for geological disposal [7].

Table D1 IAEA fundamental safety objective and principles for radiological protection

Safety Objective / Principle
Fundamental Safety Objective: The fundamental safety objective is to protect people and the environment from harmful effects of ionising radiation.
Principle 1: Responsibility for safety: The prime responsibility for safety must rest with the person or organisation responsible for facilities and activities that give rise to radiation risks.
Principle 2: Role of government: An effective legal and governmental framework for safety, including an independent regulatory body, must be established and sustained.
Principle 3: Leadership and management for safety: Effective leadership and management for safety must be established and sustained in organisations concerned with, and facilities and activities that give rise to, radiation risks.
Principle 4 Justification of facilities and activities: Facilities and activities that give rise to radiation risks must yield an overall benefit.
Principle 5: Optimisation of protection: Protection must be optimised to provide the highest level of safety that can be reasonably achieved.
Principle 6: Limitation of risk to individuals: Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm.
Principle 7: Protection of present and future generations: People and the environment, present and future, must be protected against radiation risks.
Principle 8: Prevention of accidents: All practical efforts must be made to prevent and mitigate nuclear or radiation accidents.
Principle 9: Emergency preparedness and response: Arrangements must be made for emergency preparedness and response in case of nuclear or radiation incidents.
Principle 10: Protective actions to reduce existing or unregulated radiation risks: Protective actions to reduce existing or unregulated radiation risks must be justified and optimised.

D1.4 International Commission on Radiological Protection

The ICRP is an independent advisory body that provides recommendations and guidance on radiation protection. The ICRP has no formal power to impose its proposals, but most countries adhere closely to its recommendations. In March 2007, the ICRP approved a new set of fundamental recommendations on protecting people and the environment against ionising radiation [51]. These recommendations update the ICRP's previous recommendations from 1990 [186]. Both sets of recommendations are based on three principles:

- Justification – no practice shall be adopted unless it produces sufficient benefit to the exposed individuals or to society to offset the radiation detriment it causes.
- Optimisation – the magnitude of the doses, the number of people exposed, and the likelihood that potential exposures will occur shall be kept as low as reasonably achievable, economic and social factors being taken into account (ALARA).
- Limitation – limits are placed on the dose and risk to individuals so that they do not exceed a value that is considered acceptable.

The principles represent quite general aims and underlie all radiation protection activities. They are reflected in other international principles and guidance, such as those set by the IAEA and the European Commission, and in the GRA.

The ICRP has also developed guidance on the application of its principles in specific areas, including radioactive waste disposal [e.g. 187, 188]. The ICRP guidance has been considered in the setting of radiological protection targets in the GRA, and in advice given recently by the Health Protection Agency on radiological protection objectives for solid radioactive waste disposal [189].

D1.5 Nuclear Energy Agency

The NEA is a semi-autonomous body within the Organisation for Economic Co-operation and Development (OECD). The primary objective of the NEA is to promote co-operation among the governments of its member countries in furthering the development of nuclear power as a safe, environmentally acceptable and economic energy source. The mission of the NEA's Radioactive Waste Management Committee is "to assist Member Countries in developing safe management strategies and technologies for spent nuclear fuel, long-lived waste and waste from the decommissioning of nuclear facilities." We are members of the NEA's Radioactive Waste Management Committee. In developing the ESC, we have taken account of recommendations for best practice from the NEA [8,190]. Figure 1.4 is based on the NEA's recommendations for the structure and content of an ESC main report.

D2 National legislation and guidance other than the GRA

In addition to EPR 2010 [2] and the GRA [1], development of a GDF in England or Wales must comply with a range of other legislation relevant to environmental safety and the ESC [52]. A summary of extant legislation and how this is currently expected to impact the ESC is given here.

D2.1 Strategic Environmental Assessment

Under European Directive 2001/42/EC [53], certain public sector plans and programmes that are likely to have significant effects on the environment must have a Strategic Environmental Assessment (SEA) when they are being prepared. This is to make sure that environmental effects are taken into account fully before the plan or programme is adopted. The UK Government's 2008 MRWS White Paper commits us to apply SEA to the programme for developing a GDF [4]. This ESC will support the SEA by providing an understanding of the operational and long-term environmental safety of a GDF implementation programme. Our strategy for SEA is set out in [191]. We will undertake a SEA at the appropriate stage in the MRWS Site Selection Process. The SEA will contain our evaluation of the potential environmental, social and economic impacts of implementing a GDF at a particular site.

D2.2 Land-use planning and Environmental Impact Assessment

Any proposed facility for disposing of solid radioactive waste will need planning permission under the Town and Country Planning Act 1990 in England and Wales [54]. Several applications for planning permission may be needed, with applications to undertake surface investigations at candidate sites preceding an application to undertake underground investigations and construct a GDF. We will prepare the required applications to be considered by the planning authority, either the county council, the Planning Service in Northern Ireland, or the relevant UK Government minister, who has a power under planning legislation to call in some planning applications of national importance for determination.

A statutory requirement for a GDF under the planning system will be the preparation of an Environmental Impact Assessment (EIA). An EIA identifies the environmental effects (both positive and negative) of development proposals and associated mitigation measures. We will prepare an EIA when planning permission is being sought to undertake site investigations or to begin construction of a GDF. The EIA will cover radiological and non-radiological impacts, including the use of materials, construction noise, traffic volumes, and visual intrusion. The EIA will contain our assessment of the potential environmental, social and economic impacts of our proposals. The ESC will inform the EIA of the operational and long-term environmental safety associated with emplacement of wastes in a GDF. Our strategy for EIA is set out in [191].

The environment agencies are statutory consultees to the planning authorities. The dialogue we will have with the agencies on the ESC will increase their understanding of our planning proposals, so that they can make informed comments to the planning authority.

D2.3 Water Framework Directive and Groundwater Directive

The 2000 European Water Framework Directive (WFD) [55] requires the development and implementation of a strategic framework for the management of the water environment, and establishes a common approach to protecting and setting environmental objectives for groundwaters and surface waters within the European Community. The WFD sets objectives for groundwater quality, including an objective to meet "good chemical status" by 2015, an objective on pollution trends, and an objective to prevent or limit the input of pollutants to groundwater. The 2006 European Groundwater Directive [56] clarifies these objectives and sets out specific measures to prevent and control groundwater pollution. The WFD and the 2006 Groundwater Directive together make up the new groundwater regime, which will be fully in effect in the UK by 2014.

Radioactive substances are not excluded from the 2006 Groundwater Directive. The environment agencies anticipate that new regulations to transpose the 2006 Groundwater Directive into UK law will need to be taken into account in applications for the environmental permitting of radioactive waste disposal under EPR 2010. The ESC will present information about the impacts of a GDF on groundwater.

D2.4 Health and Safety at Work Act 1974 and Nuclear Installations Act 1965

Under the Health and Safety at Work Act 1974 [57], employers are responsible for ensuring the safety of their workers and the public. The health and safety responsibilities with respect to ionising radiation are set out in the Ionising Radiations Regulations 1999 [192]. For nuclear installations, employers' responsibilities for health and safety are reinforced by the Nuclear Installations Act 1965 as amended (NIA 65) [58]. A GDF is expected to require a site licence under NIA 65, issued by the Nuclear Directorate of the Health and Safety Executive (HSE). The Nuclear Directorate has published "Safety Assessment Principles for Nuclear Facilities" [62] that apply to its assessment of safety cases for nuclear facilities and cover nuclear safety and radioactive waste management.

We will prepare the regulatory submissions required for a nuclear site licence in a parallel stream to the EPR 2010 submissions; the OSC and the ESC represent two of the three components of the DSSC. The relationship between staged environmental permitting under EPR 2010, the submissions to HSE for a site licence, and submissions for the planning process is illustrated in Figure 2.3.

D2.5 Nuclear Industries Security Regulations 2003

In the UK, civil nuclear operators must have site security plans dealing with the security arrangements to protect nuclear licensed sites and the nuclear material on these sites. The Nuclear Directorate's Office for Civil Nuclear Security within the HSE is the security regulator for the UK's civil nuclear industry. It is responsible for approving security arrangements within the industry and enforcing compliance. The Office for Civil Nuclear Security conducts its regulatory activities under the Nuclear Industries Security Regulations 2003 [59]. We will ensure that security arrangements associated with our business and the development and operation of a GDF are approved, and are consistent with the ESC.

D2.6 Regulations for the Safe Transport of Radioactive Material

Radioactive waste will be transported to a GDF under strict controls and in accordance with national and international regulations applicable to the mode of transport used (i.e., road, rail, or sea). The international regulatory standards for the transport of radioactive materials are the IAEA Regulations for the Safe Transport of Radioactive Material [61]. In the UK these regulations are implemented by the Department for Transport (DfT) in the Carriage of Dangerous Goods and Transportable Pressure Equipment, Statutory Instruments 2009 No. 1348 [60] for transport by road or rail and via the Merchant Shipping Regulations for transport by sea.

Regulation of transport of waste to a GDF (outside the boundaries of the nuclear licensed site) is undertaken by the DfT. The IAEA Transport Regulations require consignors, carriers and consignees to produce a radiological protection programme in order to demonstrate that controls and measures are in place to ensure the safety of workers and members of the public during transport and these will be prepared and made available for inspection by the DfT prior to transport. In addition, package designers are required to apply to the DfT in its role as competent authority for waste package certificate of approvals.

D2.7 Radioactive Substances Regulation Environmental Principles 2010

In concert with the establishment of EPR 2010 by the UK Government, the Environment Agency of England and Wales established a set of Radioactive Substances Regulation Environmental Principles (REPs) [193]. The ten principles are intended to provide the underlying basis for the technical assessments and judgements that Environment Agency staff make when regulating radioactive substances. The fundamental protection objective and the principles in the GRA are consistent with the REPs. However, the REPs include a change from the terminology of Best Practicable Means and Best Practicable Environmental Option to that of Best Available Techniques. This is intended to deliver a regime that is more consistent with environmental protection regimes in other countries and other regimes in England and Wales. This change applies to any discharge permits that we seek for operation of a GDF in England or Wales.

D2.8 Other environmental legislation

There will be requirements for environmental regulatory consents for activities at a GDF other than radioactive waste disposal, such as conventional waste management, discharging aqueous effluent, water abstraction, and protecting conservation sites and biodiversity. On-site storage of certain substances may fall under Control of Major Accident Hazards regulations. There are regulations and/or guidance covering all of these activities. However, they are not discussed further here as they cannot be defined in any detail at this generic stage of our programme, and they will not anyway be covered by the ESC.

Appendix E GRA crosswalk

E1 Objectives

This appendix has three objectives:

1. to provide a mapping from requirements set out in the text of the environment agencies' 2009 Guidance on the Requirements for Authorisation of geological disposal facilities on land for solid radioactive wastes (the GRA) [1] to relevant sections of the ESC main report and underlying reports;
2. to demonstrate that the requirements of the GRA have been met or will be met, or provide reasoning as to why the requirements are not considered relevant to the ESC; and
3. to support the checking of the content of the ESC and supporting reports, and to support planning of future work to fill gaps, address uncertainties, and build confidence in future updates of the ESC.

This information is presented in tabular form. We refer to this table as the 'GRA Crosswalk', to emphasise the strong link of our work to regulatory requirements. In the generic ESC, this table is not yet fully developed, because few of the GRA requirements can be met at the generic stage.

E2 Scope

The GRA is split into two parts, with Part 1 made up of Chapters 4 to 7 being identified as the guidance, and Part 2 discussing the national and international context. In the guidance in Part 1, Chapter 4 sets out the fundamental protection objective and the principles to meet the objective, Chapter 5 sets out requirements on the process, and Chapter 6 sets out management, radiological and technical requirements. This arrangement is illustrated in Figure 2.2 in the main text. Chapter 7 then sets out a series of requirements on the ESC itself, many of which build on the principles and requirements in Chapters 4 to 6 of the GRA.

The requirements considered in the GRA crosswalk are extracted from Chapters 5 to 7 of the GRA. We consider it unnecessary to produce a mapping from the ESC to Chapter 4 of the GRA, which sets out the fundamental protection objective and principles for radioactive waste management. As the GRA notes [paragraph 3.2.3], if the requirements are fulfilled proportionately to the hazard presented by the waste, then this should ensure that the principles are properly applied.

A degree of subjectivity must be applied in extracting the requirements from the GRA text. The alternative of simply reproducing each paragraph of the GRA as a potential requirement would not be appropriate, because each paragraph does not necessarily provide a requirement on us. We will independently review the set of extracted requirements against the text of the GRA as an audit to ensure that the set is comprehensive.

E3 Structure of the ESC in relation to the GRA crosswalk

The documentation structure for the ESC is set out in Figure 1.2. The crosswalk provides a mapping to the Tier 1 and Tier 2 reports that make up the ESC. It also identifies other reports that contain information to support the demonstration that a requirement is met (Tier 3). In general, it is considered sufficient to map only to the section level of the Tier 1 and Tier 2 reports and to whole Tier 3 reports only.

E4 Structure of the GRA crosswalk table

An example of several rows of the crosswalk is presented in Section E5 in the form of a table with the following columns:

- ***Requirement ID***, referring to the paragraph in the GRA from which the requirement text is extracted. Where more than one requirement is extracted from the same paragraph, a letter suffix (a), (b), etc. is used. Where two paragraphs in the GRA provide essentially the same requirement, a single entry in the table refers to both requirement IDs.
- ***Requirement text***, edited from that provided in the GRA to put the wording into the form of a need placed on the provider of the ESC.
- ***Mapping to where the requirement is addressed*** in the ESC Tier 1 and Tier 2 reports and in underpinning reports (Tier 3), or a note to indicate why the requirement has not been addressed or is not considered appropriate.
- ***Status***, summarising our position on whether a requirement is:
  - Not Addressed – we consider that the requirement is not relevant or is superseded by our position;
  - Addressed – the requirement is satisfied and no further work is to be undertaken;
  - Addressed/Ongoing – the requirement has been addressed in this version of the ESC, but the requirement will need to be revisited periodically;
  - Ongoing – work to address the requirement has been reported in this version of the ESC, but further work to build confidence or address the requirement is ongoing/anticipated; or
  - Pending – work to address the requirement has yet to be started.
- ***Forward programme***, providing a reference to which part of our forward programme is addressing any requirements that we do not regard as 'Addressed'.

E5 GRA crosswalk table

Table listing example requirements extracted from the GRA, references to material relevant to meeting these requirements in the generic ESC and supporting documents, and forward programme items aimed at further addressing the requirements. The status of each requirement is also shown, as discussed in Section E4. This table is included here as an example of how we would intend to develop the GRA crosswalk table through successive updates of the ESC.

Req'ment ID (GRA paragraph)	Requirement text (edited)	Where addressed	Status	Forward programme
5.2.3	**Requirement R1: Process by agreement** The developer should follow a process by agreement for developing a disposal facility for solid radioactive waste.	Section 2.3	Addressed	
5.2.4	Enter into an agreement with the regulator to provide advice and assistance after a decision has been made to start a process to select a site for a geological disposal facility.	Section 2.3	Addressed	
5.3.1	Apply for an authorisation under RSA 93 before any disposal of radioactive waste.	No longer relevant with the introduction of EPR 2010 [2]	Not Addressed	
5.4.4	Submit an updated environmental safety case at each hold point in the 'staged environmental permitting process (important stages in the facility development programme), as agreed with the regulator.		Addressed/ Ongoing	
5.4.5	Provide a forward work programme at each hold point for review by the regulator. This should identify the proposed work during the next development phase including discussion of how any regulatory issues are to be addressed.	Section 6.3 R&D Overview	Addressed/ Ongoing	
5.4.8	Apply for an environmental permit to proceed with intrusive site investigation.		Pending	
5.4.9	Submit an 'initial site evaluation' at the hold point before an intrusive site investigation programme, giving largely qualitative views on the feasibility of constructing a geological disposal facility at the potential site.		Pending	
5.4.12	Submit a 'preliminary environmental safety evaluation' at the hold point before proceeding with underground operations to support a request for a revised environmental permit. At this hold point, the preliminary environmental safety evaluation would need to be consistent with the GRA.		Pending	

Appendix F Scrutiny of the ESC

Each update of the ESC will contain an appendix or supporting report that identifies and summarises reviews of the preceding version of the ESC and our responses to those reviews and/or where in the ESC the comment has been considered. In addition, we will identify the main changes between successive updates of the ESC. In the generic ESC, we only consider the feedback from previous regulatory scrutiny because other organisations are yet to provide comments that would need to be considered in developing the ESC. However, updates to the ESC will summarise the range of dialogue we have had with other interested parties.

F1 Regulatory scrutiny

As discussed in Section 2.3.2, the Environment Agency has been involved in regulatory scrutiny of our work since the formation of RWMD and, previously, it scrutinised the work of Nirex. A wide range of regulatory scrutiny interactions and reports have been produced, as summarised in [34]. The two regulatory scrutiny reviews most relevant overall to the generic ESC are the reviews conducted by the Environment Agency of Nirex's GDF viability assessment [35] and generic post-closure performance assessment (GPA) work [36]. In the interest of demonstrating a connection between past and current work on UK GDF implementation and regulatory dialogue, we set out below the issues identified in the viability assessment work by Nirex and the Environment Agency's review of that work (Section F1), and the issues identified in the Environment Agency's review of the GPA and our responses to them (Section F2).

The Environment Agency reviewed a storyboard we developed in 2009 to set out our intentions for the generic ESC [43]. We also summarise below the Environment Agency's comments on the storyboard and our responses to those comments and/or where they have been considered in this report (Section F3).

F1.1 Environment Agency review of Nirex viability assessment

Nirex considered in 2005 the key challenges for a GDF in higher strength rock in its viability assessment [99]. Nirex's main conclusion was that disposal of higher activity radioactive wastes in higher strength rock in the UK was technically viable and that the disposal concepts considered then (and which we have presented in Appendix A) could be implemented in the UK. The remaining challenges were considered significant, but were not considered to represent a fundamental threat to concept viability; we concur with this position.

With publication of the 2008 MRWS White Paper [4], initiation of the MRWS Site Selection Process, and establishment of ourselves as the delivery organisation for a GDF, both the context and responsibility for implementation of a GDF in the UK has changed. In particular, we – RWMD – are not focused on implementing geological disposal in higher strength rock; prior to the identification of candidate sites during the initial part of Stage 4 of the MRWS Site Selection Process, our work is giving equal weight to the range of possible geological environments in the UK. Nonetheless, we consider that the challenges set out in Nirex's 2005 Viability Assessment are still relevant and their consideration demonstrates a sensible connection between this historic work, our progress in the interim, and our current programme:

- For the ILW/LLW disposal concept, these challenges were the potential impact of carbon-14 (considered in Sections 5.1.2 and 5.2.2.2 and in our safety assessment reports [26, 27] and Gas status report [19]) and non-aqueous phase liquids (considered in the Geosphere and Radionuclide behaviour status reports [17, 20]).

- For the HLW/SF disposal concept, these challenges were retrievability (considered in Section 3.1.3 and in our Generic disposal facility designs report [24]; also see [75]), optimisation of the layout of the disposal area, such as vertical or horizontal emplacement openings (considered in Section 3.1.2 and our disposal concept option studies [28, 29]), and the technical implications of co-locating a HLW/SF disposal facility at the same site as a ILW/LLW facility (considered in [71, 72]).

In its review of Nirex's viability assessment, the Environment Agency agreed that there were no major issues that made geological disposal in higher strength rock non-viable. However, the Environment Agency considered that the list of technical challenges should have included other issues [35]. We have considered these issues in the Tier 2 research status reports and safety assessments that form part of the document suite making up the generic ESC, as shown in Table F1. The issues will need to be considered further once we have specific sites to work with and have developed site-specific disposal concepts.

Table F1 Research status reports and safety assessments that consider issued raised by the Environment Agency's review of the Nirex viability assessment

None of these issues has yet been closed out, and they will be considered further once we have specific sites to work with and have developed site-specific disposal concepts.

No.	Environment Agency issue (from [35])	Generic ESC Tier 2 report(s) addressing the issue
1	The need to better understand waste package longevity and corresponding degradation mechanisms over a long period of storage and, hence, any requirement to produce improved packages for certain waste streams or to make provision for reworking	Package evolution status report [15]
2	Developing a good understanding of groundwater flow and radionuclide transport at a specific site, including the representation of groundwater flow and contaminant transport	Geosphere status report [17] Radionuclide behaviour status report [20]
3	A fuller understanding of the impact of organic complexants and colloids as well as non-aqueous phase liquids	Geosphere status report [17] Radionuclide behaviour status report [20]
4	Understanding the potential coupling between gas and groundwater flow	Gas status report [19]
5	Developing a better understanding of the evolution of the near field and its role in limiting radionuclide release, which should be closely linked to the consideration of possible design optimisation	Near-field evolution status report [16]
6	The need for long-term experiments to demonstrate the behaviour of near-field components	Near-field evolution status report [16]
7	Building more confidence in the safety case for criticality	Criticality safety status report [21]
8	Developing a clear strategy for GDF sealing that is demonstrated to function adequately in the long term	Near-field evolution status report [16]
9	Building an understanding of time-dependent effects and their consideration in a justifiable way in assessment models	All research status reports and safety assessments
10	Demonstrating an adequate understanding of the values of key parameters	All research status reports and safety assessments

F1.2 Environment Agency review of Nirex generic performance assessment

Nirex conducted a suite of generic performance assessments (GPAs) [e.g. 80], which has been reviewed by the Environment Agency. In so far as our assessment methodologies and presentational approaches build on the work of Nirex, the Environment Agency's review of the Nirex GPA work is relevant to the generic ESC. The Environment Agency's review and recommendations to us are contained in [36], and our response to the review is contained in [37]. In our response, we set out in general terms how we intended to respond to the Environment Agency's comments in the DSSC. Table F2 sets out the Environment Agency's recommendations, and where we have considered them in the Generic ESC documentation.

Table F2 Locations in the Generic ESC main report and supporting reports that consider issues raised by the Environment Agency's review of the Nirex generic performance assessment

These issues need to be considered as part of every update to the ESC. In the generic ESC some of these issues have only been dealt with at a high level because of our use of a simplified assessment approach appropriate to the generic stage of the programme.

No.	Environment Agency recommendation (from [36])	Place(s) considered in generic ESC
1	NDA RWMD should more clearly define the scope and purpose of the GPA (or its replacement)	Section 1.1 Section 2.2
2	Assessment objectives should be more specific and should be critically evaluated at the end of the assessment	Section 6.2
3	In future post-closure assessments, the assessment objectives, methodology, key assumptions and uncertainties should be summarised in the main body of the report. Detailed information necessary for evaluating or verifying the assessment should be placed or referenced in appendices	Section 4.3.2 Section 5.2.2 Section 5.3 Generic PCSA [27]
4	NDA RWMD should better define the assessment context in future post-closure assessments	Section 2 Section 4
5	In future post-closure assessments, the overall assessment methodology should be better described and justified	Section 3.2 Section 4.3.2 Section 5.2.2 Generic PCSA [27]
6	NDA RWMD should continue to use simple calculations as a means of cross-checking the results of more complex post-closure modelling	PCSA approach overall is relatively simple for generic ESC
7	Justification for the selection and use of models (including adaptation of Nirex 97 models) needs improvement and better documentation in future post-closure assessments	Section 4.3.2 Generic PCSA [27] Research status reports
8	We encourage NDA RWMD to consider the issues and arguments associated with model complexity when developing a future post-closure modelling strategy	Section 3.2.2.4 Section 4.3.2 Generic PCSA [27]

No.	Environment Agency recommendation (from [36])	Place(s) considered in generic ESC
9	NDA RWMD should clearly explain the context of the groundwater pathway assessment to minimise the likelihood of readers misinterpreting the projected risk. When post-closure risks are cited in other NDA RWMD reports, there should be clear qualifying statements to explain the context under which those risks were generated	Section 4.3.2.1 Section 5.2.2.1 Generic PCSA [27]
10	NDA RWMD should highlight the strengths and weaknesses inherent in 'working backwards' from the risk target in respect of the disposal inventory	Approach no longer being followed
11	NDA RWMD should consider the issues associated with model uncertainty and lessons learnt from model inter-comparison studies	Section 3.2 Section 5.2.2 Generic PCSA [27] Research status reports
12	NDA RWMD should clarify its position on peer review and submit any future post-closure assessments for peer review	Section 3.3.2 Section 3.3.5

F1.3 Environment Agency review of storyboard for the Generic ESC main report

Table F3 sets out the Environment Agency's comments on the storyboard for the Generic ESC main report, and our responses to those comments and/or where the comment is addressed in this report.

Table F3 Response to Environment Agency's review of our storyboard for the Generic ESC main report

Category headings are as set out by the Environment Agency in their comments [43]. Note that there was an error in the numbering of the Environment Agency's comments, which ran from 1 to 22, and then from 20 to 27. In order to give each comment a unique reference number, we have renumbered the comment set 20 – 27 so that they run instead from 23 to 30. We have also shortened the Environment Agency's comments where possible, without losing the sense of the comment, by including mainly the Environment Agency's summary statement and not the more detailed supporting text in the comment; deleted text is indicated by the use of dots (…).

No.	Environment Agency review comment (from [43])	Our response / where addressed in Generic ESC main report
Purpose of the generic ESC		
1	The purpose of the Generic ESC is not clear to us from the storyboard. …	The objectives of the generic ESC are set out in Section 1.1, and the objectives of updates to the ESC at later stages of the MRWS Site Selection process in Section 2.
2	This general ambiguity in the purpose is reflected in some ambiguities about what RWMD will actually do: • The rationale for the choice of reference cases for assessment in the Generic ESC is not explained in the storyboard. … • …We are unclear whether current packaging advice is vulnerable in the event of a GDF in a different host rock. • …We would like RWMD to justify its intention to assess groundwater migration for the different "reference cases" in the Generic ESC but not assess gas migration for the same cases.	We consider each of these comments in turn: • Sections 4.1.3 and 4.1.4 explain the rationale behind selection of the set of illustrative geological disposal examples we considered in the generic ESC. • Section 3.1.4 explains the link between the disposability assessment process and other aspects of the ESC. • The generic ESC considers both radionuclide transport by groundwater and by gas, although only the former is discussed quantitatively for the operational period and only the latter for the post-closure period; we consider these to be the most likely means of release of radionuclides to the surface environment during the two different periods. Our Gas status report [19] and generic PCSA [27] summarise previous quantitative work on gas generation and migration during the post-closure period, including calculations for a range of geological environments. However, we have not previously undertaken calculations of radionuclide transport by groundwater for a range of geological environments.
3	In describing the purpose of the Generic ESC, we will expect RWMD also to describe the inherent limitations of a generic ESC designed for that purpose.	See response to comment 1.

No.	Environment Agency review comment (from [43])	Our response / where addressed in Generic ESC main report
Future development of the ESC		
4	…We are pleased to see a recognition of the importance of a clearly defined assessment context and modelling strategy.	This is a comment only.
5	The 'storyboard' summarises what will be in the 2010 Generic ESC, and gives pointers to how subsequent iterations of the ESC might look as the site selection and design processes continue. We are encouraged that RWMD is thinking through such questions, but the picture is not yet clear. …	Section 1.3 sets out the structure of the generic ESC documentation, and Sections 2.1 and 2.2 set out how the ESC will be updated with time and summarise how its objectives will evolve.
6	We understand that future iterations of the ESC will serve a dual purpose: firstly demonstrating the viability of the proposed GDF in light of the information available at that stage; and secondly illustrating how the proposed activities will address, in the next iteration of the ESC, the identified technical challenges. …We recognise that RWMD needs to retain the flexibility to respond to the actual circumstances when future stages are reached, but we would expect to see a basic strategy for this in place.	The "basic strategy" is set out in Sections 2.1 and 2.2, which summarise how the ESC will be updated with time and how its objectives will evolve. Section 3.1 discusses how new information from our design, site assessment, and research work will feed into the next update of the ESC.
7	…We would like RWMD to provide a clear outline of what level of understanding of the site and disposal system it expects to have reached by each stage, and what it expects to demonstrate at each iteration of the ESC (and the wider DSSC). …	See response to comment 6.
8	…While the general concept of progressively building the knowledge and understanding to underpin an ESC is appropriate, we would like RWMD to consider also possible situations in which more significant changes of direction might be necessary at particular stages.	Sections 3.1.3 - 3.1.5 discuss how our disposability assessment, design, and research and site investigation processes are responsive to new information. Section 3.1.2 discusses how optioneering is an integral part of our work programme.
9	…Given each stage (development, operation and closure) will span many decades we suggest that within these stages there will be a need to maintain and revisit the safety cases for other reasons, e.g. to meet statutory obligations.	Sections 2.1 and 2.2 summarise how the ESC will be updated with time; the discussion in these sections is in accord with this comment.

No.	Environment Agency review comment (from [43])	Our response / where addressed in Generic ESC main report
10	…We expect that RWMD will provide a clear and simple audit trail of changes in successive iterations of safety cases and the reasons for those changes, which will include not only reviewers' comments but also new information, new data, new requirements etc.	Section 1.4 sets out our intentions in this regard. Section 3.3.4 discusses documentation and record keeping with regard to our management systems. Appendix F outlines where in the ESC comments have been addressed. We will agree with the Environment Agency the form of the audit trail to document changes between updates to the ESC. One possibility would be to summarise where significant changes have occurred in an appendix, and via the use of redlining in the margin of the text.
11	The Preface indicates that RWMD intends to develop the DSSC (including the ESC) in a modular way. We recognise that this may have advantages, but we stress that all modules and the DSSC as a whole will need to be kept fully internally consistent, at least at each stage when a regulatory permission is sought.	This language has been deleted from the report. It was intended to convey the point that the ESC comprises a suite of documents, and that some of the Tier 2 documents could be updated and discussed with the Environment Agency without updating the entire set of ESC documents, that is between stages at which further regulatory permission is sought.
12	RWMD needs to include an unequivocal statement about where ownership of and responsibility for the ESC sits.	Ownership of the ESC and DSSC resides with RWMD. Section 3.3.1.1 summarises how we ensure we are a "capable organisation", and how we act as an "intelligent customer" in using input from contractors (e.g. to assist with report development and peer review). We use "we" and "our" terminology throughout to demonstrate our ownership of the reports.
Hierarchy of DSSC-related documents		
13	The 'storyboard' refers to an "overview report" heading the hierarchy of DSSC documents in which the ESC will sit, and indicates that for the 2010 Generic DSSC this overview report will be the document NDA/RWMD/010. We will provide separately our comments on a draft of that document.	This is a comment only.
14	There are few references in the 'storyboard' to other supporting or underpinning RWMD (or Nirex) strategy or position documents. …We are unclear about the status of previously issued RWMD documents.	This Generic ESC main report references Nirex, Environment Agency, national and international literature, as well as the Tier 2 ESC reports and other relevant RWMD reports as appropriate. Also, in the interest of demonstrating a connection between past and current work on UK GDF implementation and regulatory dialogue, Section 2.3.2 provides selected examples of regulatory scrutiny of previous Nirex and RWMD work, and Appendix F summarises where in the ESC we have dealt with previous Environment Agency comments that are of most relevance to the generic ESC.

No.	Environment Agency review comment (from [43])	Our response / where addressed in Generic ESC main report
Relationship between the ESC and disposability assessment process		
15	We understand that the Generic ESC will be used to support waste packaging advice (given via the LoC process) in the same way that assessments of the PGRC are used. The 'storyboard' does not explain how or whether this will happen (though we accept this may be described elsewhere). For example, a range of geological environments and concepts will be considered, but it is not yet clear how RWMD intends to address packaging advice for a GDF located in geologies other than hard rock. Maintaining an appropriate degree of stability in packaging advice while shifting from a single generic repository concept, through uncertainty about the geological environment and concept for disposal, to a site-specific GDF design, will be challenging. Because of this, we expect RWMD to present evidence that it is developing a carefully planned course through this transition.	This is really two comments. As noted in Section 3.1.4, the assessments summarised in Section 5 of this Generic ESC main report provide an updated basis for our conduct of disposability assessments. As discussed in Section 5.2.2, the generic PCSA [27] is not specific to any particular geological environment. Our aim has been to illustrate that a GDF could be safely developed in a wide range of geological environments. The assessments therefore provide a basis for confidence that decisions on the packaging of waste are robust to a wide range of geological environments that could be present at the candidate site(s). Section 3.1.4 summarises the role of the generic ESC in the disposability assessment process. Our approach to updating the ESC is summarised in Section 2.1.
16	When we reviewed the GPA [36] we questioned the extent to which a generic assessment can demonstrate the feasibility of a concept. We noted that the GPA assessment appeared to be focussed on one particular approach i.e. 'working backwards' from a predefined risk end-point, and that this aspect and its associated limitations were not well explained. …We would encourage the use of a range of approaches, with clearly defined success criteria and full explanation of the intent and limitations.	In the Generic ESC main report we make use of a wide range of qualitative arguments in discussing environmental safety (e.g. Section 5.1.1 and 5.2.1). Our post-closure quantitative assessment in the generic ESC is not based on working backwards from a predefined risk end-point. Rather, our aim has been to illustrate that a GDF could be safely developed in a wide range of geological environments. The risk guidance level in the GRA [1] does, however, provide a metric for presenting post-closure assessment results in Section 5.2.2.1 and in the generic PCSA report [27]. We also recognise that a GDF would have to be optimised once a preferred site and disposal concept are identified, as discussed in Section 3.1.2.

No.	Environment Agency review comment (from [43])	Our response / where addressed in Generic ESC main report
Scope of the generic ESC		
17	….We are unclear how ‘common’ issues of concern will be managed between the SEA and Generic ESC. How far will generic considerations be taken? Will the Generic ESC take into account how staged development might proceed and what information would be needed/available for each geological environment at each stage?	This is really two comments. Section D2.1 discusses the relationship between the Strategic Environmental Assessment (SEA) and the ESC, and this refers the reader to [191] for further detail on the SEA. The ESC is seen as supporting the SEA. The ESC is based on staged implementation of a GDF, consistent with the UK Government’s 2008 MRWS White Paper [4], and staged environmental permitting by the Environment Agency, consistent with the Environmental Permitting Regulations 2010 [2], as discussed in detail in Sections 2.1 and 2.2.
18	….The ‘storyboard’ indicates that the Generic ESC will describe how the ESC, site characterisation and concept/design will develop iteratively to produce an optimised solution. We are particularly interested in this at this stage and we look forward to seeing these descriptions.	Section 2.1 discusses our approach to updating of the ESC, Section 3.2 considers the roles of optioneering and optimisation as the ESC is updated, and Sections 3.4 and 3.5 discuss how design and site characterisation plans will be successively updated in concert with updates to the ESC.

F2 Scrutiny by other organisations

For the generic ESC, this section is here as a placeholder for future updates of the ESC, as discussed above.

Glossary

activation product

A radionuclide created by the neutron irradiation of a previous non-radioactive material. Typically this occurs in the structural and moderator materials (e.g. steel and graphite) of nuclear reactors.

active institutional control

Control of a disposal site for solid radioactive waste by an authority or institution holding a permit under the EPR 10, involving monitoring, surveillance and remedial work as necessary, as well as control of land use.

activity

The number of atoms of a radioactive substance which decay by nuclear disintegration each second. The SI unit of activity is the becquerel (Bq).

Advanced Gas-cooled Reactor (AGR)

The reactor type used in the UK's second generation nuclear power plants. (MRWS)

alpha activity

Alpha activity takes the form of particles (helium nuclei) ejected from a decaying (radioactive) atom. Alpha particles cause ionisation in biological tissue which may lead to damage. The particles have a very short range in air (typically about 5 cm) and alpha particles present in materials that are outside of the body are prevented from doing biological damage by the superficial dead skin cells, but become significant if inhaled or swallowed. (MRWS)

alkaline disturbed zone (ADZ)

A volume of rock in the vicinity of a GDF containing water with an elevated pH as a result of the presence of cementitious materials (e.g. in waste packages, backfill or structural items).

anion

A negatively charged atom or molecule.

Assessment basis

Information that underpins the qualitative and quantitative safety assessments provided in a performance assessment or an environmental safety case.

backfill

A material used to fill voids in a GDF. 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 or chemical means of preventing or inhibiting the movement of radionuclides.

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. (MRWS)

becquerel (Bq)

The standard international unit of radioactivity equal to one radioactive decay per second. Multiples of becquerels commonly used to define radioactive waste activity are:

- kilobecquerels (kBq) equal to 1 thousand (10^3) Bq
- megabecquerels (MBq) equal to 1 million (10^6) Bq
- gigabecquerels (GBq) equal to 1 billion (10^9) Bq
- terabecquerels (TBq) equal to 1 trillion (10^{12}) Bq

bentonite

A clay material that swells when saturated with water which is used as a backfill and buffer material in some disposal concepts.

beta activity

Beta activity takes the form of particles (electrons) emitted during radioactive decay from the nucleus of an atom. Beta particles cause ionisation in biological tissue which may lead to damage. Most beta particles can pass through the skin and penetrate the body, but a few millimetres of light materials, such as aluminium, will generally shield against them. (MRWS)

biosphere

Regions of the earth's surface and atmosphere normally inhabited by living organisms.

borehole disposal

The concept of disposing of some forms of radioactive waste in extremely deep boreholes, a number of kilometres down in the Earth's crust. (MRWS)

British Geological Survey (BGS)

The BGS provides expert services and impartial advice in all areas of geoscience.

buffer

An engineered barrier that protects the waste package and limits the migration of radionuclides following their release from a waste package.

canister

A term used in specific concepts to describe the waste container into which a wasteform is placed.

closure

The administrative and technical actions that have to be taken to put a disposal facility in its intended final state after the completion of waste emplacement.

community siting partnership

A partnership of local community interests that will work with the NDA's delivery organisation and with other relevant interested parties to ensure questions and concerns of potential host communities and its Wider Local Interests are addressed and resolved as far as reasonably practicable and to advise Decision Making Bodies at each stage of the process.

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.

container

The vessel into which a wasteform is placed to form a waste package suitable for handling, transport, storage and disposal.

criticality

A state in which a quantity of fissile material can maintain a self-sustaining neutron chain reaction. Criticality requires that a sufficiently large quantity of fissile material (a critical mass) be assembled into a geometry that can sustain a chain reaction; unless both of these requirements are met, no chain reaction can take place and the system is said to be sub-critical.

criticality safety

The protection against the consequences of an inadvertent nuclear chain reaction, preferably by prevention of the chain reaction.

decommissioning

The process whereby a nuclear facility, at the end of its economic life, is taken permanently out of service and its site made available for other purposes. The term ‘site clean-up’ is sometimes used to describe the work undertaken to make the site available for other purposes. (MRWS)

depleted uranium

Uranium containing a lesser mass percentage of uranium-235 than in natural uranium. (IAEA)

deposition hole

Vertical hole in which a waste package is placed in certain disposal concepts.

devolved administrations

Collective term for the Scottish Executive, Welsh Assembly Government and in Northern Ireland, the Department of the Environment.

disposability

The ability of a waste package to satisfy the defined requirement for disposal.

disposability assessment

The process by which the *disposability* of proposed waste packages is assessed. The outcome of a disposability assessment may be a *Letter of Compliance* endorsing the disposability of the proposed waste packages.

disposal

In the context of solid waste, disposal is the emplacement of waste in a suitable facility without intent to retrieve it at a later date; retrieval may be possible but, if intended, the appropriate term is storage. (MRWS)

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 deposal tunnels. The number of deposition tunnels in a module could vary for different rock types.

disposal system

All the aspects of the waste, the disposal facility and its surroundings that affect the radiological impact.

disposal tunnel

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

disposal vault

Underground opening where ILW or LLW waste packages are emplaced.

dose

A measure of the energy deposited by radiation in a target. (IAEA)

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).

dose rate

The effective dose equivalent per unit time. Typical units of effective dose are sievert/hour (Sv hr^{-1}), millisieverts/hour (mSv hr^{-1}) and sievert/year (Sv yr^{-1}).

drift

A sloping underground tunnel.

effective dose equivalent

In addition to dose equivalent taking into account the biological effectiveness of various forms of radiation, effective dose equivalent takes into account the differing sensitivities of various body tissues. Effective dose equivalent thus aims to reflect the risk to health for the irradiated person, regardless of the widely different dose equivalents that might be received by the various organs. This is a useful concept for comparisons between the risks from various radiation exposure pathways.

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 uranium-235 in uranium.

Environmental Impact Assessment (EIA)

A legal requirement under EU Directive 85/337/EEC (as amended) for certain types of project, including various categories of radioactive waste management project. It requires information on the environmental impacts of a project proposal to be submitted by the developer and evaluated by the relevant competent authority (the planning authority, HSE or other regulators concerned). (MRWS)

environmental safety

The safety of people and the environment both at the time of disposal and in the future. (Definition taken from the GRA.)

Environmental Safety Case (ESC)

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.

Euratom Treaty

The legislative basis for the activities of European Union countries in the nuclear energy field. (MRWS)

European Commission (EC)

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. (MRWS)

European Union (EU)

The European Union of countries of which the United Kingdom is a member. The EU issues its own legislation which the UK, as a member state, is obliged to follow. (MRWS)

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 (EDZ)

A region of the geosphere surrounding the engineered barrier system which has been affected (i.e. physically altered) 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 surrounding a GDF, comprising the surrounding geological strata, at a distance such that the GDF can be, for modelling purposes, considered a single entity.

fissile material

Fissile material is that which undergoes fission under neutron irradiation. For regulatory purposes material containing any of the following nuclides is considered to be ‘fissile’: uranium-233, urainium-235, plutonium-239 and plutonium-241.

fission product

A radionuclide produced by nuclear fission. (IAEA)

footprint

The area of host rock required to accept the inventory which is to be disposed of. The footprint will also be determined by the properties of the host rock, the geometry of the features within it and whether the disposal tunnels or vaults are built on a single or multiple levels within a GDF.

gamma activity

An electromagnetic radiation similar in some respects to visible light, but with higher energy. Gamma rays cause ionisations in biological tissue which may lead to damage. Gamma rays are very penetrating and are attenuated only by shields of dense metal or concrete, perhaps some metres thick, depending on their energy. Their emission during radioactive decay is usually accompanied by particle emission (beta or alpha activity). (MRWS)

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. (MRWS)

geological disposal facility (GDF)

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

geological environment

The structure, composition and physical and chemical characteristics of the rocks that make up the geosphere.

geological unit

A discrete subdivision of rock mass within which the certain defined characteristics (e.g. hydrogeological properties, age range or rock type) are essentially similar and can be treated as a coherent body for modelling purposes. Each geological unit will have characteristics that distinguish it from adjacent geological units.

geosphere

The rock surrounding a GDF that is located below the depth affected by normal human activities and is therefore not considered to be part of the biosphere.

groundwater

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

half-life

The time taken for the activity of a given amount of a radioactive substance to decay to half of its initial value. Each radionuclide has a unique half-life. (MRWS)

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: low level waste not suitable for near surface disposal, intermediate level waste and high level waste.

higher strength rock

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

highly enriched uranium (HEU)

Uranium containing 20% or more by mass of the isotope uranium-235.

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. (MRWS)

host community

The community in which any facility will be built. This 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. (MRWS)

immobilisation

A process by which the potential for the migration or dispersion of the radioactivity present in a material is reduced. This is often achieved by converting the material to a monolithic form that confers passive safety to the material.

indurated clay

Clay rocks with a dry density of greater than about 1,900 kg m^{-3}. This includes geologically older rocks such as the Mercia Mudstone along with Silurian and Ordovician mudstones. Oxford Clay, which is of Jurassic age, straddles the boundary between plastic clay rocks and indurated clay rocks. It is considered that at relatively shallow depth the Oxford Clay may behave as a plastic clay rock while at deeper depths, it may behave as an indurated clay rock.

Industrial Package (Type-IP)

A category of transport package, defined by the IAEA Transport Regulations for the transport of radioactive materials with low specific activities.

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. (MRWS)

internal irradiation

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

ionisation

When radiation (alpha, beta, and gamma activity) interacts with matter, it can cause atoms and molecules to become unstable (creating ions). This process is called ionisation. Ionisation within biological tissue from radiation is the first stage in radiation leading to possible change or damage within the tissue. (MRWS)

legacy waste

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

Letter of Compliance (LoC)

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.

lithostatic pressure

The pressure imposed on a layer of soil or rock by the weight of overlying material.

low enriched uranium (LEU)

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

lower strength sedimentary rock

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

low level waste (LLW)

Defined as “radioactive waste having a radioactive content not exceeding 4 gigabecquerels per tonne (GBq/te) of alpha or 12 GBq/te of beta/gamma activity”. MRWS

Low Level Waste Repository (LLWR)

The UK national facility for the near surface disposal of solid LLW, located near to the village of Drigg in Cumbria.

natural uranium

Uranium containing the naturally occurring distribution of uranium isotopes (approximately 99.28% uranium-238 and 0.72% uranium-235 by mass).

near field

The engineered barrier system (including the wasteform, waste containers, buffer materials, backfill, and seals), as well as the host rock within which the GDF is situated, to whatever distance the properties of the host rock have been affected by the presence of a GDF.

new build

New build of a nuclear power station.

nuclear licensed site

Any site which is the subject of a license granted by the Nuclear Installations Inspectorate (part of HSE) under the Nuclear Installations Act 1965. Nuclear licensed sites include nuclear power stations, nuclear fuel production and reprocessing sites, sites undertaking storage of and/or research into nuclear materials, and major plant producing radioisotopes.

nuclear material

Fissile material or material that can be used to produce fissile material (i.e. source material). This includes all isotopes of uranium, plutonium and thorium, together with certain isotopes of neptunium and americium.

operational period (of a disposal facility)

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

operational waste

Radioactive waste produced during the normal operations of a nuclear facility (as distinct from *decommissioning waste*).

overpack

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

package

See waste package, transport package.

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 waste 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. (MRWS)

performance assessment

Assessment of the performance of a system or sub-system and its implications for protection and safety at an authorised facility. (IAEA)

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.

planning authorities

A general term for those regional planning bodies and local authorities throughout the UK who are responsible for the preparation of planning strategies and for determining applications for construction and operation of waste treatment and disposal facilities that may be sited in their area of responsibility. (MRWS)

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. (MRWS)

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.

post-closure period (of a disposal facility)

The period following sealing and closure of a facility and the removal of active institutional controls.

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. (MRWS)

quality management system (QMS)

A quality management system is the overall system by which an organisation determines, implements and ensures quality.

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. (MRWS)

radioactive material

Material designated in national law or by a regulatory body as being subject to regulatory control because of its radioactivity. (MRWS)

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. (MRWS)

radioactivity

Atoms undergoing spontaneous random disintegration, usually accompanied by the emission of radiation. (MRWS)

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. (MRWS)

reprocessing

A physical or chemical separation operation, the purpose of which is to extract uranium or plutonium for re-use from spent nuclear fuel. (MRWS)

resaturation

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

retardation

A feature of a component of a GDF that contributes to safety. The engineered barriers and host geological environment 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.

retrievability

A feature of the design of a GDF that enables the waste to be withdrawn, even after the disposal vaults have been backfilled.

reversibility

Term used internationally to denote the ability to reverse decisions, as part of a phased decision-making process. Has been used in the UK to describe retrieval by reversing the original emplacement process.

safeguards

Measures used to verify that nation states comply with their international obligations not to use nuclear materials (plutonium, uranium and thorium) for nuclear explosives purposes. Global recognition of the need for such verification is reflected in the requirements of the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) for the application of safeguards by the International Atomic Energy Agency. Also, the Treaty Establishing the European Atomic Energy Community (the Euratom Treaty) includes requirements for the application of safeguards by the EC.

safety argument

A statement about the safety of the geological disposal system, backed up by supporting evidence and qualitative and/or quantitative reasoning.

safety cases

A 'safety case' is the written documentation demonstrating that risks associated with a site, a plant, part of a plant or a plant modification are as low as reasonably practicable and that the relevant standards have been met. Safety cases for licensable activities at nuclear sites are required as license conditions under NIA65. (MRWS)

safety concept

The set of safety functions provided by the natural and engineered barriers in a geological disposal system that provides assurance of safety for a specific site and design.

safety function

A specific purpose that must be accomplished for safety. (IAEA)

safety function indicator

A measure for the performance of a system component or several components to support the development of system understanding and to assess the quality, reliability and effectiveness of particular aspects or components of a disposal system.

safety function indicator criterion

A quantitative limit such that if the safety function indicator to which it relates fulfils the criterion, the corresponding safety function is maintained.

safety indicator

A measure of the overall performance of the geological disposal system with respect to a specific safety aspect (e.g. radiological dose or risk), in comparison with a reference value that can be shown to be safe, or is at least commonly considered to be safe.

safety strategy

An approach or course of action designed to achieve and demonstrate the safety of people and the environment both at the time of disposal and in the future. Generally consists of a design and siting strategy, an assessment strategy, and a management strategy.

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.

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. (IAEA)

sorption

The interaction of an atom, molecule or particle with the solid surface at a solid–solution or a solid–gas interface. Used in the context of radionuclide migration to describe the interaction of radionuclides in pore- or groundwater with soil or host rock, and of radionuclides in surface water bodies with suspended and bed sediments. (IAEA)

spent fuel

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. (IAEA)

stakeholders

People or organisations, having a particular knowledge of, interest in, or be affected by, radioactive waste, examples being the waste producers and owners, waste regulators, non-Governmental organisations and local communities and authorities. (MRWS)

Strategic Environmental Assessment (SEA)

The type of environmental assessment legally required by EC Directive 2001/42/EC in the preparation of certain plans and programmes. The authority responsible for the plan or programme must prepare an environmental report on its likely significant effects, consult the public on the report and the plan or programme proposals, take the findings into account, and provide information on the plan or programme as finally adopted. (MRWS)

stillage

A metal frame designed to hold four 500 litre drum waste packages so that they can be handled, stacked and transported as a single *disposal unit.*

stylised approach

A stylised approach involves the use of a set of assumptions that are either generally reasonable or clearly conservative, and is accepted by the environmental regulators in the absence of specific information about what will actually happen in the far future. Used in modelling the biosphere and in considering possible future human intrusion in post-closure safety assessments.

Sustainability Appraisal (SA)

A form of assessment used in England, particularly in regional and local planning, covering the social, environmental and economic effects of proposed plans and appraising them in relation to the aims of sustainable development. SAs fully incorporating the requirements of the SEA Directive (2001/42/EC) are mandatory for a range of regional and local planning documents under the Planning and Compulsory Purchase Act 2004. (MRWS)

transport container

A reusable container into which waste packages are placed for transport, the whole assembly then being referred to as a *transport package.*

transport package

The complete assembly of the radioactive material and its outer packaging, as presented for transport. (IAEA)

transport system

The transport system covers the transport modes, infrastructure, design and operations. It can be divided in two main areas– the transport of construction materials, spoil and personnel associated with building a GDF and the more specialised transport of the radioactive waste to a GDF by inland waterway, sea, rail and/or road.

UK Radioactive Waste Inventory (UK RWI)

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 Defra 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. (MRWS)

uncertainty

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

unconditioned waste

Radioactive waste in its initially generated state, prior to its preparation and packaging for longer term storage and/or disposal in a solid and stable form.

underground rock laboratory (URL)

An underground facility developed for research and testing purposes at a site that may eventually be used for waste disposal and may be a precursor to the development of a GDF at the site

unshielded waste package

A waste package which, owing either to radiation levels or containment requirements, requires remote handling and must be transported in a reusable transport container.

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. (MRWS)

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

voluntarism

An approach in which communities “express an interest” in participating in the process that would ultimately provide the site for a geological disposal facility. Initially a community would be expressing an interest in finding out more about what hosting such a facility would involve. In the latter stages there would be more detailed discussion of plans and potential impacts. (MRWS)

waste acceptance criteria (WAC)

Quantitative and/or qualitative criteria, specified by the operator of a *disposal facility* and approved by the regulator, for solid radioactive waste to be accepted for disposal.

Quantitative or qualitative criteria specified by the regulatory body, or specified by an operator and approved by the regulatory body, for radioactive waste to be accepted by the operator of a repository for disposal, or by the operator of a storage facility for storage.

waste container

Any vessel used to contain a wasteform for disposal.

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. (MRWS)

waste package

The product of conditioning that includes the waste form and any container(s) and internal barriers (e.g. absorbing materials and liner), as prepared in accordance with requirements for handling, transport, storage and/ or disposal. (IAEA)

waste producer

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

WVP canister

A stainless steel vessel containing vitrified HLW, as manufactured in the Waste Vitrification

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