Geological Disposal

Study of recent post-closure safety cases

March 2014
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Abstract

This report has been prepared by the NDA Radioactive Waste Management Directorate Post-closure Safety Group, which has responsibility for producing the long-term Environmental Safety Case for a geological disposal facility for radioactive wastes.

Geological disposal of radioactive wastes is also planned or being implemented by many different overseas organisations and these organisations have also produced Environmental Safety Cases, or their equivalents, appropriate to the stage of facility development. This report studies selected examples of international safety cases of particular relevance to the UK, to gain knowledge and understanding of different safety case approaches and to identify what may be learned to assist the on-going development of the UK Environmental Safety Case for deep geological disposal.

In addition we have also studied the Environmental Safety Case produced for the near surface facility for Low Level Waste by Low Level Waste Repository Limited. This is a currently operating facility in the UK licensed under the UK regulatory regime, and therefore, we believe, also of relevance to assist the on-going development of the UK Environmental Safety Case for deep geological disposal.
Preface

This report is part of an ongoing programme of work conducted by the Nuclear Decommissioning Authority (NDA) and its contractors. It is a component of the research into the implementation of geological disposal for radioactive wastes in the UK.

Geological disposal is the UK Government’s policy for the long-term management of higher-activity radioactive wastes. The principle of geological disposal is to isolate the waste deep inside a suitable rock formation to ensure that no harmful quantities of radioactivity reach the surface environment. To achieve this, the waste will be placed in an engineered underground containment facility – the geological disposal facility. The facility will be designed so that natural and man-made barriers work together to minimise the escape of radionuclides. The NDA has developed a multi-barrier concept for geological disposal of higher-activity radioactive wastes. These wastes include high-level waste (HLW), spent nuclear fuel, intermediate-level (ILW) and certain low-level (LLW) radioactive wastes.

A geological disposal facility would be carefully designed and engineered to provide excavated vaults for ILW and LLW, together with the necessary access ways. Typically, the ILW and LLW wastes would be packaged in steel or concrete containers, usually with a cement grout, and subsequently placed in the vaults. Some time later, the vaults would be backfilled with a cement-based material, completely surrounding the waste packages. Engineered barriers would be provided by the cement grout, the containers and the backfill. Natural barriers would be provided by the geological formations that surround the repository and that lie between the repository and the accessible human environment. The concept for dealing with HLW and spent nuclear fuel is different in that these materials are more likely to be placed in very long-lived canisters and these placed directly into deposition tunnels, again using both engineered and natural barriers. The actual disposal concepts to be used for each waste type will be selected and refined once the location and geological setting of a geological disposal facility is known.

The multi-barrier concept is common to the majority of international disposal facilities that are being planned or implemented and the safety cases produced to support them. Therefore there is value in studying such safety cases, particularly those produced for facilities at a more advanced stage of development than that in the UK, in order to identify any lessons that can be learned which may benefit the development of the UK safety case.
Executive Summary

This report has been prepared by the Radioactive Waste Management Directorate (RWMD) of the Nuclear Decommissioning Authority to gain knowledge and understanding of a range of recently published safety cases for radioactive waste disposal facilities and to identify any lessons that may be learned to assist the on-going development of the UK deep geological disposal facility post-closure safety case. The safety cases selected for this study were chosen to represent a range of advanced safety cases for different geological settings and different waste types that are of interest and relevance to the UK radioactive waste disposal programme.

The UK radioactive waste disposal programme for higher activity wastes is addressing the disposal of intermediate level waste (ILW), some low level waste (LLW), high level waste (HLW), spent fuel and other nuclear materials, including separated uranium and plutonium. As no site has yet been identified in the UK to host a geological disposal facility for these materials, RWMD’s current post-closure safety case is generic and discusses the potential disposal issues in three broad, generic host rock environments; namely a higher strength (crystalline) rock, a lower strength (sedimentary or clay) rock and a salt host rock.

The five safety cases selected for this study cover these three host rock types and also the range of wastes requiring disposal in the UK, they are:

1. The Swedish SR-Site safety case – a licence submission for a geological disposal facility for spent fuel in crystalline granitic rock.
4. The US Yucca Mountain Repository Licence Application – an application to dispose of spent nuclear fuel and high-level waste in the unsaturated zone of volcanic tuff.

Although this last safety case is for a near-surface facility, rather than a deep geological disposal facility, it is of interest to RWMD as it is a UK safety case, prepared to similar regulatory guidance as that published for a UK GDF.

The study focused on the approach to demonstrating safety, the safety arguments employed, the role and use of modelling and the structure and presentation of the safety cases. These aspects were all related to the context of each safety case and are discussed in relation to the RWMD safety case context, as summarised below.

Approach to demonstrating safety

The SR-Site approach to demonstrating safety and building confidence is based on the identification and analysis of safety functions, i.e. those properties of the disposal system that contribute to its safety. The most important safety function is the long-term integrity of the copper disposal canister. A number of other safety functions are concerned with protecting the canister’s integrity (for example this is the main role of the bentonite buffer). SR-Site focuses its safety arguments around the demonstration of the identified safety functions, with particular emphasis on the durability of the copper canister. Quantitative safety calculations are provided up to 1 million years; thereafter SR-Site provides a qualitative discussion on the expected evolution and consequences of the repository, with
links to safety arguments based on comparisons with natural analogues. SR-Site adopts an 11-step approach to safety assessment, as follows:

1. Processing of features, events and processes (FEPs)
2. Description of the initial state (including the initial state of the engineered barriers and repository layout)
3. Description of external conditions
4. Compilation of process reports
5. Definition of safety functions and safety function indicators
6. Compilation of input data
7. Definition and analyses of reference evolution
8. Selection of scenarios
9. Analyses of selected scenarios
10. Additional analyses
11. Conclusions

Andra’s safety approach is based on developing and documenting a detailed understanding of the expected behaviour of the disposal system, based on the body of scientific knowledge acquired through research and development. The approach to understanding system safety is discussed in terms of the following:

- Definition of expected safety functions
- Acquisition of knowledge to substantiate system behaviour
- Engineering solutions
- Uncertainty analysis
- Definition of a number of possible evolutions (split into normal evolution scenarios and alternative evolution scenarios)
- Performance assessment
- Feedback from the assessments to aid design and knowledge acquisition

Andra assigns a number of safety functions to those components of the disposal facility that have a significant safety role, such as the host rock, waste packages and other engineered elements. This knowledge is embodied in a safety assessment model that provides a simplified but cautious representation of the disposal system and its evolution over time. A key safety argument is the low permeability and retention capacity of the clay host rock, which is expected to act as a barrier to radionuclide migration by ensuring that any radionuclide movement will be diffusion controlled.

In the US, the assessment methodology is more explicitly defined in the relevant regulations than is the case for most European regulations. This is demonstrated in the WIPP safety case, in which a detailed numerical performance assessment, which explicitly incorporates the impact of uncertainties within the system, is conducted. These uncertainties include parameter uncertainty and probabilistic future events, such as human intrusion. The performance assessment is constructed using a logical approach in which FEPs relevant to the system (with some aspects constrained or defined through the appropriate federal regulations) are identified. These are developed into conceptual and mathematical models which are coded into computer models. The computer model is run a large number of times and results for a normalised radionuclide release calculated. Compliance with safety limits requires demonstrating that the probabilistic outputs meet the containment requirements specified in the appropriate federal regulations.

In a similar manner to the WIPP safety case, a detailed set of regulatory requirements combined with a structured FEP analysis and probabilistic risk model were the core of the
Yucca Mountain licence application. The presentation of qualitative safety arguments is limited in the Yucca Mountain licence application, where the emphasis is on providing a quantitative demonstration that regulatory requirements would be met. A 'bottom up', structured FEP screening and analysis was used to derive a set of modelling scenarios which was then assessed to demonstrate the safety of the system.

The 2011 LLWR ESC develops its safety assessment from a thorough and scientifically based understanding of the evolution and performance of the existing disposal facility and its planned development, by considering:

- Potential radiological and non-radiological hazards;
- Safety functions; and
- FEPs that provide, promote or reduce the safety functions.

The 2011 LLWR ESC is an on-going safety case for an active, operational facility. The approach to demonstrating safety therefore involves addressing action points from previous regulatory submissions (for example, the 2008 assessment) as part of demonstrating that the requirements from the Environment Agency (the regulator) have been met and thus that the case has been made that the facility is both currently compliant with regulations and will be compliant for future disposals. The 2011 LLWR ESC sets out its safety arguments under four key headings:

1. Management
2. System characterisation and understanding
3. Optimisation and Site Development Plan
4. Assessment.

The on-going management of the LLWR facility, including demonstrating understanding of the system in order to optimise future actions, is the essential aspect of the LLWR ESC. To ensure that only wastes that are consistent with the assumptions made in the safety assessments are accepted to the facility, LLW Repository Ltd has set Waste Acceptance Criteria that are derived from the safety assessments.

Safety arguments and role of models

SR-Site uses both qualitative and quantitative reasoning. A large part of the safety case report is devoted to gaining an understanding of the evolution of the repository system, which is largely achieved through quantitative modelling, with one important aim being a quantitative determination of the containment potential of the canisters. Quantitative consequence calculations are presented up to 1 million years, and thereafter qualitative reasoning for continued safety. SR-Site uses a number of different models to represent the various processes of interest to the evolution of a repository and assessment of its consequences. These models and the information flows between them are illustrated graphically in Assessment Model Flowcharts (AMFs).

The safety reasoning in the Dossier 2005 Argile feasibility study is broadly presented as a continuous and coherent argument, which revolves around building confidence in system understanding. Both qualitative and quantitative reasoning are important aspects to Dossier 2005 Argile. Models are used to support the system and site understanding alongside the expert qualitative discussions. Andra makes extensive use of a variety of models each serving different purposes. The primary safety assessment model adopts a deterministic approach, but probabilistic calculations are undertaken to support a sensitivity analysis of parameter uncertainty not explicitly accounted for in the normal evolution scenario. The primary safety assessment model is used to provide a numerical comparison against the regulatory recommendation (known as the 'Basic safety Rule'), that the committed dose must remain lower than 0.25 mSv/year in the expected evolution for a period of at least 10,000 years. Other supporting models are used to build understanding of the disposal system evolution.
The WIPP safety case makes extensive use of quantitative lines of reasoning. The federal regulations require a relatively short assessment timescale of 10,000 years and specifically direct DOE to produce a probabilistic performance assessment over this period. Qualitative lines of reasoning are therefore less important for the WIPP safety case than many other international safety cases. Qualitative arguments are primarily used to suggest that selected ranges of parameters bound the expected values and are conservative and not optimistic. The WIPP safety case utilises a computer modelling system to identify possible cumulative radionuclide releases from the facility. The model system contains a number of modules that calculate flow and transport in the culebra and salado dolomite rock formations; in addition to releases due to cuttings, cavings, spallings and through direct brine releases in the event that a future human intrusion event occurs. The model system samples stochastic input parameters and calculates probabilistic outputs that are presented as complementary cumulative distribution functions (CCDFs) for comparison with the regulatory criteria.

Quantitative performance assessment modelling was also a significant part of the Yucca Mountain licence application, with qualitative arguments supporting selected parameter ranges as reflecting the evidence at hand and being reasonably conservative. A hierarchical modelling system was used, with detailed process models abstracted into the performance assessment model. Natural analogue results were used in the validation of the process models.

LLW Repository Limited used both qualitative and quantitative reasoning in the 2011 ESC. The Long Term Radiological Assessment presents a detailed quantitative assessment of the radiological performance of the site, but it also sets out a qualitative description of the functions and expected evolution of the facility in its environment. The 2011 LLWR ESC uses a standard, internationally recognised approach to modelling and a hierarchical modelling approach is used. In the 2011 ESC a probabilistic risk analysis was undertaken for the groundwater pathway for radionuclides released via a water abstraction well. This probabilistic analysis is preceded by a suite of deterministic calculations to investigate separately the impacts of alternative near-field models, assumed future inventories, different biosphere release paths, and alternative data assumptions, e.g. for solubility limits and sorption parameters.

**Regulatory context and assessment timescales**

Swedish regulations indicate that the timescale of a safety assessment for spent nuclear fuel should be one million years after closure – with a detailed risk analysis required for the first 1,000 years and a quantitative risk analysis up to approximately 100,000 years. Beyond 100,000 years quantitative risk calculations are not considered meaningful, but the safety case should still demonstrate that releases from the engineered and geological barriers are limited and delayed as far as reasonably possible, using calculated risk as one of several indicators.

SR-Site actually presents numerical safety calculations up to one million years and thereafter provides qualitative reasoning (including discussion of natural analogues) to provide confidence in continued safety.

When considering assessment timescales, Andra is guided by the Basic Safety Rule. In keeping with international practice, Dossier 2005 Argile extends performance assessment calculations against the Basic Safety Rule recommended dose limit to a period of one million years. Although Andra’s approach to assessment remains consistent over all timescales, it is acknowledged that the various system components and processes are more or less relevant depending on the timeframe of interest.

US federal regulations define the performance assessment timescale for assessments of the WIPP facility to be 10,000 years. This is therefore the period modelled within the numerical performance assessment which underpins the WIPP CCA.
For the Yucca Mountain licence application, initial regulations required demonstration that the mean annual dose up to 10,000 years post-closure was no more than 0.15 mSv. Subsequently, an individual protection standard applying up to one million years post-closure was introduced, during which period the mean annual dose should not exceed 1 mSv. The Yucca Mountain performance assessment therefore includes a probabilistic dose assessment up to one million years.

As LLW Repository Limited disposes of low-level waste, it has a slightly different approach to defining the assessment timescales as the site is located near to the sea in an area of coastal erosion. The ESC draws upon both qualitative and quantitative evidence including modelling studies, and concludes that the coastal site for the LLWR facility is likely to begin to be eroded on a timescale of a few hundred to a few thousand years, with consequent disruption of the repository, with erosion of the vaults and trenches being complete within one to a few thousands of years.

Based on this, the LLWR ESC presents two scenarios; the first is the expected evolution scenario which calculates releases on the completion of closure engineering, at the end of management control, 100 years after present and 500 years after present (based on the start of coastal erosion). The second scenario is a delayed coastal erosion scenario where impacts are considered for up to 10,000 years after present with the assumption (considered to be unlikely) that the site will still not be completely eroded after 10,000 years. Several other timescales are studied for assessing the contributions to dose and the rationale for choosing these timescales is also provided.

**Structure and presentation**

SR-Site consists of a main report of 900 pages. It is supported by six production reports, four process reports, a FEP report, a Data report, a Model Summary report and four other technical reports, and around 100 more specialised reports. The reports were primarily prepared for the regulatory authorities and hence are all technical in nature. Within the main safety case report there is no formal hierarchical structure although it includes a 38-page summary section in which the main safety arguments are presented.

Andra refers to Dossier 2005 Argile through a series of reports at varying levels. At the top level there is a synthesis report written in relatively succinct language. This summarises information from the three main underpinning volumes, titled:

1. Safety evaluation of a geological repository
2. Phenomenological evolution of a geological repository
3. Architecture and management of a geological repository

Supporting these volumes are a number of technical and reference documents. Graphics are used throughout Andra’s Dossier 2005 Argile documents to aid presentation and clarification of the discussion in the text. Diagrams range from graphs to illustrations of the disposal system and technical drawings of container designs. Although not explicitly part of the dossier, Andra has produced various leaflets and a high-level, 40-page summary document for a non-technical audience, covering different aspects of geological disposal in France.

The WIPP Compliance Certification Application is a single document with multiple chapters and appendices, totalling around 72,000 pages, available and searchable electronically. This structure was also used for the 2004 Compliance Recertification Application (CRA). The original 1996 CCA and 2004 CRA were structured with nine chapters on specific areas (together with appropriate appendices and appendix attachments) and a regulatory crosswalk document was provided to map the safety case to the regulatory requirements. However, the 2009 Recertification was structured according to a different scheme that reflected the federal regulations, thus a regulatory crosswalk document was no longer required. Results of a peer review are also presented as part of the submission.
The Yucca Mountain licence application is a submission to address specific regulatory acceptance criteria allowing the construction of a disposal facility. It is a detailed set of documents, focussed on systematically demonstrating that each of the regulatory criteria is satisfied. The Yucca Mountain licence application has two volumes, a general introduction and the Safety Analysis Report. The models supporting the licence application are described in an additional four volumes, totalling approximately 4,000 pages.

The LLWR 2011 ESC consists of a non-technical summary and a main report (or Level 1 report) supported by six technical assessment (Level 2) reports, as follows:

1. Environmental Safety During the Period of Authorisation
2. Assessment of Long-term Radiological Impacts
3. Assessment of Non-Radiological Impacts
4. Assessment of Impacts on Non-human Biota
5. Waste Acceptance
6. Assessment of an Extended Disposal Area

There are 16 level 2 reports in total, which are also supported by several level 3 reports in the form of LLW Repository Limited and contractor technical reports, and external references. The 2011 LLWR ESC has a hierarchical structure in terms of its reporting and appears to mirror the requirements in the regulatory guidance. This submission was developed in the form of an ESC, which includes considerations both during the period of authorisation and in the future and is much wider than just a performance assessment.
# Main acronyms used in this report

## National organisations

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<tr>
<th>Acronym</th>
<th>Full Name</th>
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<tbody>
<tr>
<td>CoRWM</td>
<td>Committee on Radioactive Waste Management</td>
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<td>NDA</td>
<td>Nuclear Decommissioning Authority</td>
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<td>RWMD</td>
<td>NDA Radioactive Waste Management Directorate</td>
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## International and overseas organisations

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<td>Andra</td>
<td>French national radioactive waste management agency</td>
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<tr>
<td>IAEA</td>
<td>International Atomic Energy Agency</td>
</tr>
<tr>
<td>NEA/OECD</td>
<td>Nuclear Energy Agency / Organisation for Economic Co-operation and Development</td>
</tr>
<tr>
<td>SKB</td>
<td>Swedish nuclear fuel and waste management company</td>
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<tr>
<td>US DOE</td>
<td>United States Department of Energy</td>
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## Legislation, Regulation, Guidance

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<td>EPR 2010</td>
<td>Environmental Permitting (England and Wales) Regulations 2010</td>
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<td>GRA</td>
<td>Guidance on Requirements for Authorisation</td>
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<td>RSA 93</td>
<td>Radioactive Substances Act 1993</td>
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## RWMD safety cases and safety assessments

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<tr>
<td>DSSC</td>
<td>Disposal System Safety Case</td>
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<td>ESC</td>
<td>Environmental Safety Case</td>
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<tr>
<td>OESA</td>
<td>Operational Environmental Safety Assessment</td>
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<tr>
<td>PCSA</td>
<td>Post-closure Safety Assessment</td>
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## Radioactive material types

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<tr>
<td>HLW</td>
<td>high level waste</td>
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<td>ILW</td>
<td>intermediate level waste</td>
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<tr>
<td>LLW</td>
<td>low level waste</td>
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<td>vLLW</td>
<td>very low level waste</td>
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<td>SF</td>
<td>spent fuel</td>
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## Other acronyms

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<tbody>
<tr>
<td>BAT</td>
<td>best available techniques</td>
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<tr>
<td>EBS</td>
<td>engineered barrier system</td>
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<tr>
<td>FEPs</td>
<td>features, events and processes</td>
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<tr>
<td>GDF</td>
<td>geological disposal facility</td>
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<tr>
<td>PA</td>
<td>performance assessment</td>
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<tr>
<td>PDF</td>
<td>probability density function</td>
</tr>
<tr>
<td>PWR</td>
<td>Pressurised Water Reactor</td>
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<tr>
<td>R&amp;D</td>
<td>research and development</td>
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UKRWI United Kingdom Radioactive Waste Inventory

**SR-Site specific**
- AMF: assessment model flowchart
- BWR: boiling water reactor
- UOX: uranium oxide fuel
- MOX: mixed oxide fuel
- SSM: the Swedish Radiation Safety Authority
- SSMFS: the Swedish Radiation Safety Authority’s regulations
- DR: data report

**Dossier 2005 Argile specific**
- MHM: Meuse/Haute-Marne
- THMC: thermal, hydrological, mechanical and chemical
- SEN: normal evolution scenario
- CEA: Commissariat à l’Energie Atomique
- EDF: Electricité de France
- QSA: Quantitative Safety Analysis
- PARS: Phenomenological Analysis of Repository Situation

**Waste Isolation Pilot Plant Compliance Certificate Application specific**
- WIPP: Waste Isolation Pilot Plant
- CRA: compliance recertification application
- CCA: compliance certification application
- CCDF: complementary cumulative distribution functions
- USCFR: United States code of federal regulations
- TRU: transuranic waste
- USEPA: United States Environmental Protection Agency
- PCM: plutonium contaminated material
- CH: contact handled
- RH: remote handled
- TDOP: ten drum overpack
- HERE: horizontal emplacement and retrieval equipment
- CARDs: compliance application review documents
- LHS: Latin hypercube sampling
- PAVT: performance assessment verification test

**Yucca Mountain Total System Performance Assessment specific**
- USNRC: United States Nuclear Regulatory Commission
- MTHM: metric tons of heavy metal
- TSPA: total system performance assessment
- TSPA-VA: viability assessment
- LADS: licence application design selection
TSPA-LA  licence application
RMEI  reasonably maximally exposed individual
PMA  performance margin analysis
GI  general information
SAR  safety analysis report
EPRI  Electric Power Research Institute

Low Level Waste Repository Environmental Safety Case
LLWR  low level waste repository
PCSC  post closure safety case
OESC  operational environmental safety case
SDP  site development plan
LTRA  long term radiological risk assessment
GRM  generic repository model
RDA  reference disposal area
EA  Environment Agency (England and Wales)
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1 Introduction

This report has been prepared by the Radioactive Waste Management Directorate (RWMD) of the Nuclear Decommissioning Authority to gain knowledge and understanding of a range of recently published safety cases and to identify what lessons can be learned from these safety cases to influence the development of RWMD’s post-closure safety case for deep geological disposal.

The safety cases selected for this review were chosen because of their diversity, maturity and their interest and relevance to the UK higher activity radioactive waste disposal programme. In identifying any lessons for RWMD it is important to take note of the context in which each safety case was produced (for example, the inventory for disposal, the geological setting, the relevant regulatory framework and the stage of disposal facility development) and consider how that relates to RWMD’s current context.

In RWMD’s generic Disposal System Safety Case [1, 2, 3], three generic geological environments were considered to illustrate the potential performance characteristics of a geological disposal facility (GDF) in the types of geological host rock likely to be considered in the UK. These three generic host rocks are:

- Higher strength rock (such as granite);
- Lower strength, sedimentary rock (such as clay);
- Evaporite (salt).

Examples of relatively advanced international safety cases developed for each of these three host rocks were selected for inclusion in this study. In addition, a further advanced safety case based on geological disposal in unsaturated rock was also studied; and also the recently submitted Environmental Safety Case for the Low Level Waste Repository (LLWR) near Drigg in West Cumbria. Although this latter safety case is for a near-surface facility, rather than a deep geological disposal facility, it is of interest to RWMD as it is a UK safety case, prepared to similar regulatory guidance as that published for a UK GDF [4, 5].

The terminology of an ‘Environmental Safety Case’ is specific to the UK regulatory requirements [4, 5] and is much wider than performance assessment in the post-closure period of a geological disposal facility. The Environmental Safety Case will demonstrate that members of the public and the environment are adequately protected (from both radiological and non-radiological hazards), both at the time of disposal (including during the period of authorisation) and in the future. An Environmental Safety Case covers more than just performance assessment calculations. For assessments covering times after the period of authorisation and extending into the very long term, it is likely that safety will be demonstrated through multiple lines of reasoning based on a variety of evidence, leading to complementary environmental safety arguments. It is the assembly of multiple safety arguments that constitutes a safety case, of which the safety (or performance) assessment is one component.

In this study, we have selected a variety of different published documents for review. However, it is important to note that due to the differing regulatory regimes internationally, some of the documents reviewed place more emphasis on the performance assessment (specifically the documents from the US) whereas the others adopt a broader ‘Safety Case’ approach.

The five safety cases that were selected for study in this report are as follows (including reference to the main report reviewed for each example):

1. The Swedish SR-Site safety case – a licence submission for a geological disposal facility for spent fuel in crystalline granitic rock [6]


4. The US Yucca Mountain Repository Licence Application – an application to dispose of spent nuclear fuel and high-level waste in the unsaturated zone of volcanic tuff [9]


The following five chapters of this report discuss each of these five safety cases in turn. For each the background and context of the safety case is described, along with the wastes for disposal and the disposal concept. The discussion then focuses on the approach to assessing safety and the main safety arguments presented to support each safety case. Consideration is also given to the presentation of the safety case and how this reflects the intended audience.

Chapter 7 then discusses what lessons can be learned from these safety cases for the ongoing development of RWMD’s post-closure safety case.

This report has been produced for the benefit of RWMD staff, to demonstrate relevant learning from international safety cases. It will also be of interest to anyone who wishes to obtain an overview of the safety case approaches of the five safety cases selected for this study. It is recognised that there are other relevant safety cases and these may be the subject of future studies. The five safety cases considered here were prioritised because of their diversity and their greatest relevance and interest to RWMD at its current generic stage, in terms of the geological host rocks addressed, the disposal inventory and the regulatory context.
2 SR-Site

2.1 Background and context

SR-Site is the post-closure safety case submitted by the Swedish nuclear fuel and waste management company, SKB, to support a licence application to construct and operate a disposal facility for spent nuclear fuel at Forsmark in the municipality of Östhammar in Sweden [6].

SKB has carried out site investigations for a disposal facility in the municipalities of Östhammar (Forsmark area) and Oskarshamn (Laxemar area) and in June 2009, the Forsmark site was selected as the site for the disposal facility.

SR-Site demonstrates long-term safety for a repository at Forsmark based on the Swedish ‘KBS-3’ disposal facility design. The KBS-3 method for the disposal of spent fuel was first introduced in 1983 and has been developed by SKB over several decades of research and development. In the KBS-3 method, copper canisters with a cast iron insert containing spent nuclear fuel are surrounded by bentonite clay and deposited at approximately 500 metres depth in groundwater saturated granitic rock (for further details see Section 2.2 below).

SR-Site builds on previous safety cases, in particular:

- SR 97 [11], an assessment of the KBS-3 concept using geological data from three different sites in Sweden, which SKB published in 1999 to demonstrate that the KBS-3 concept had good prospects of being able to meet the specified safety and radiation protection requirements; to demonstrate a methodology for safety assessment; and to derive preliminary safety functional requirements on the canisters and other barriers.

- SR-Can [12], a first assessment of the long-term safety of a KBS-3 disposal facility with vertical emplacement (KBS-3V) at the Forsmark and Laxemar sites using data from the initial site investigations, which SKB published in 2006. The results of SR-Can were used to give feedback to the design development of the KBS-3 concept, so that more detailed design premises could be formulated for the canister, bentonite buffer and deposition tunnel and on how to adapt the design and layout to the host rock. SR-Can was reviewed by the Swedish regulators and the conclusions from that review were considered in detail in the preparation of SR-Site.

SR-Site is part of SKB’s licence application to construct and operate a disposal facility for spent nuclear fuel at Forsmark. The licence application includes a Safety Report, an Environmental Impact Assessment and supporting documents on the selection of the KBS-3 disposal concept and the selection of the Forsmark site. The Safety Report has two sub-reports, SR-Operation and SR-Site. SR-Operation describes how safety will be maintained during operation and closure of the facility and the measures that will be taken to ensure that the end of operations provide the ‘initial state’ that is the starting point for analysis in SR-Site.

As the main report of the assessment of long-term safety for the licence application to build a disposal facility at the Forsmark site, SR-Site has two main stated purposes:

- To assess the safety, as defined in applicable Swedish regulations, of the proposed KBS-3V disposal facility at Forsmark; and
To provide feedback to design development, to SKB’s research and development programme, to detailed site investigations and to future safety assessment projects. As SR-Site is part of a licensing submission, the regulatory authorities (principally SSM) are the main audience. During the SR-Site project, the report was reviewed at several stages by SKB’s own independent international expert review group. After publication SR-Site has been subject to international review by an OECD-NEA expert group, as requested by the Swedish Government. The SR-Site main report includes a ~50 page summary, which may be of more interest to local community members [6].

2.2 System description

2.2.1 Disposal inventory and waste types

SR-Site addresses the disposal of spent nuclear fuel from Swedish nuclear power plants. The 12 Swedish nuclear power plants are all either boiling water reactor (BWR) or pressurised water reactor (PWR) type. The majority of fuel used in these reactors is uranium oxide fuel (UOX), but there are minor amounts of mixed oxide fuel (MOX) from Oskarshamn and minor amounts of other oxide fuel types from research and the early years of the Swedish nuclear power programme. The Swedish national policy is to dispose of spent fuel directly, without reprocessing. Around 12,000 tonnes of spent nuclear fuel, in about 6,000 canisters, are planned for disposal. No other waste types are included in SR-Site.

2.2.2 Geological environment

The proposed disposal site at Forsmark, within the municipality of Östhammar, is located about 170 km north of Stockholm. The geological setting, including the host rock, is saturated, granitic rock – a crystalline bedrock, belonging to the Fennoscandian Shield, formed 2 billion years ago. The bedrock contains tectonic lenses (in which the bedrock is less affected by ductile deformation) between ductile high-strain belts. The repository candidate area is located in the north-westernmost part of one of these tectonic lenses, which extends from the north-west of the Forsmark nuclear power plant south-eastwards to the areas around Öregund.

Three major sets of deformation zones with distinctive orientations have been identified at the locality. In addition to vertical and steeply dipping zones, there are also gently dipping zones which have higher hydraulic transmissivity than the vertical and steeply dipping deformation zones at the site. The frequency of open fractures is very low below 300m depth and the rock stresses are relatively high compared to typical values of the Swedish bedrock. The upper 100-150m bedrock overlying the repository site contains many highly transmissive fractures in the horizontal plane and in good hydraulic contact over long distances, but the rock at depth appears to have very low permeability with few transmissive fractures (at repository depth the average distance between transmissive fractures is more than 100 m).

Groundwaters in the uppermost 100 – 200 m of the bedrock have a wide range of chemical variability, with chloride concentrations in the range 200 to 5,000 mg/litre, that suggest the influence of both brackish marine and meteoric waters. At depths between about 200 m and 800 m the groundwater is indicative of brackish marine water with chloride concentrations fairly constant in the range 5,000 to 6,000 mg/litre. Below this at depths between 800 and 1,000 m the salinity increases to higher levels indicating brackish to saline non-marine groundwaters. These deep waters further show an increase in calcium with depth, which is indicative of water-rock interactions that occur under increasingly low flow to stagnant groundwater conditions with increasing depth.

The main safety-related features of the Forsmark site are:
• a low frequency of water-conducting fractures at repository depth;
• favourable chemical conditions, in particular reducing conditions at repository depth and salinity that would ensure stability of the bentonite; and
• the absence of metallic and industrial mineral deposits, minimising the potential for future intrusions into the facility.

2.2.3 Disposal concept
The KBS-3V disposal concept utilises copper canisters with a cast iron insert to contain the spent fuel. Each canister is 5 cm thick, 4.835 metres long, with a diameter of 1.05 metres. These canisters are manufactured from a single piece of copper and then welded closed, using friction stir welding, once filled with spent fuel. Each canister will be deposited in an individual, vertically-drilled deposition hole at about 500 metres depth and surrounded by bentonite clay, which will provide a buffer to protect the canister from rock movements and limit groundwater access to the canister. Should a copper canister fail, the bentonite buffer will also provide a diffusive barrier to limit the movement of radionuclides away from the canister.

The deposition holes will be drilled in the floor of deposition tunnels. Following deposition, these tunnels will be backfilled with a bentonite clay that has low hydraulic conductivity and a sufficient swelling pressure and density to keep the buffer in place, i.e. restricting upwards buffer expansion (the reference backfill material is bentonite with a montmorillonite content of 50-60%).

Deposition holes with high groundwater inflow will be avoided and the spacing of the canisters will be such as to ensure that the peak temperature in the buffer remains below 100°C, to fulfil a crucial requirement that the residual power at deposition must not exceed 1700 W/canister. The reference design layout for the Forsmark site represents a high, but not finally maximised, gross capacity with regard to potential deposition positions, over a footprint area of about 4 km², whilst allowing for a respect distance of 100 metres for any deformation zone with a trace length longer than 3 km.

2.3 Safety case approach
2.3.1 Regulatory requirements
The form and content of a Swedish safety assessment and the criteria for judging the safety of a disposal facility for spent nuclear fuel, are defined in regulations originally issued by SKI and SSI and now by SSM. The regulations are based on various relevant components of framework legislation, the most important being the Nuclear Activities Act and the Radiation Protection Act. Swedish regulations are influenced by guidance from a number of international bodies.

The principal regulatory acceptance criterion that SR-Site needs to address (Swedish regulation SSMFS 2008:37) [13] concerns the protection of human health and requires that the “annual risk of harmful effects after closure does not exceed $10^{-6}$ for a representative individual in the group exposed to the greatest risk”. “Harmful effects” refers to cancer and hereditary effects and the risk limit corresponds to an effective dose limit of about $1.4 \times 10^{-6}$ Sv/yr (which is about 1% of the effective dose due to natural background radiation in Sweden).

The regulation SSMFS 2008:21 [14] also requires descriptions of the evolution of the biosphere, geosphere and repository for selected scenarios and evaluation of environmental impacts of the repository for selected scenarios, including the main (base case) scenario, with respect to defects in engineered barriers and other identified uncertainties.
Swedish regulations [13] indicate that the timescale of a safety assessment for spent nuclear fuel should be one million years after closure – with a detailed risk analysis required for the first 1,000 years and a quantitative risk analysis up to approximately 100,000 years. Beyond 100,000 years quantitative risk calculations are not considered meaningful, but the safety case should still demonstrate that releases from the engineered and geological barriers are limited and delayed as far as reasonably possible, using calculated risk as one of several indicators.

2.3.2 Safety arguments

The main safety argument in SR-Site is the durability of the copper canisters. The detailed analyses in SR-Site demonstrate that, even with a number of cautious assumptions, canister failures within one million years will be rare. Deterioration of the barrier system to the extent that containment is lost is assessed to occur, as a statistical average, for less than one canister within one million years. Such a failure is calculated to be most likely due to erosion of the bentonite buffer, leading to advective conditions and enhanced corrosion of the copper canister. The other failure mode that could not be ruled out is an earthquake-induced shear movement in fractures intersecting canister deposition holes, leading to a canister shear failure. However, even with pessimistic assumptions, this shear failure is calculated on average to affect considerably less than one canister in the one million year assessment time period.

The safety philosophy behind the KBS-3V concept is summarised by the following safety principles, many of which are directly consistent with internationally-accepted approaches to geological disposal:

- By placing the repository at depth in a long-term stable geological environment, the waste is isolated from the human and near-surface environment. This means that the repository is strongly affected neither by societal changes nor by the direct effects of long-term climate change on the ground surface.
- By locating the repository at a site where the host rock can be assumed to be of no economic interest to future generations, the risk of human intrusion is reduced.
- The spent fuel is surrounded by several engineered and natural safety barriers.
- The primary safety function of the barriers is to contain the fuel within a canister.
- Should containment be breached, the secondary safety function of the barriers is to retard a potential release from the repository.
- Engineered barriers will be made of naturally occurring materials that are stable in the long term in the repository environment. The long-term properties of the materials are verifiable, for example from analogues.
- The repository will be designed and constructed so that temperatures that could have significant detrimental effects on the long-term properties of the barriers are avoided.
- The repository will be designed and constructed so that radiation-induced processes that could have significant detrimental effects on the long-term behaviour of the engineered barriers or of the rock are avoided.
- The barriers will be passive, i.e. they function without human intervention and without artificial supply of matter or energy.

These safety principles emphasise the importance of the copper canister as providing the primary safety function in the KBS-3V concept.

The understanding of safety is built upon a systematic identification of safety functions (i.e. those properties of the disposal system that contribute to its safety) and criteria for the safety functions. A repository reference evolution is analysed over several time frames.
using a structured approach that addresses each of the processes identified as relevant in terms of the safety functions. The reference evolution is defined as the main scenario, representing the reasonably expected evolution of the repository system from its initial state. The safety assessment then considers a set of additional scenarios that examine the possible ways in which the identified safety functions could be impaired and calculates the consequences of such impairments.

The safety functions are used as the basis for the selection of the additional scenarios. These comprise three buffer scenarios, representing different potential failed states of the buffer and three canister scenarios representing distinct canister failure modes (canister failure due to corrosion, isostatic load or shear load). The buffer scenarios are analysed first and the outcome of each buffer scenario is then considered in the analyses of the canister failure modes. Should the analysis of any buffer scenario lead to the conclusion that the corresponding failed buffer state can be ruled out, that state is not propagated. Scenarios related to future human actions and any other scenarios analysed (e.g. to understand barrier functions) are included as necessary. In summary, the scenario methodology is an investigation of all routes to the three identified canister failure modes aiming at either ruling them out or quantifying them, considering all conceivable evolutions of the system.

A cautious (even pessimistic) approach is taken when addressing uncertainties. Significant uncertainties are addressed when analysing the additional scenarios that focus on an evaluation of possible uncertainties that are not addressed in the reference evolution scenario. These uncertainties may be related to the initial state of the disposal facility, to processes governing its evolution or to external influences. For example, in the analysis of the buffer erosion scenario leading to advective conditions at the copper canister surface (one of the three failed buffer scenarios), the following issues were among those addressed:

- Could the initial buffer density for any reason be lower than the reference initial density assumed in the reference evolution?
- Are there any remaining conceptual uncertainties related to the buffer colloid release/erosion process (leading to the loss of buffer density) that are not addressed in the reference evolution?
- Could the groundwater composition and flow be less favourable for the arising of buffer advection than that deduced from the reference external conditions?

Rather than a detailed mathematical treatment of uncertainty, the approach to managing uncertainty in SR-Site involves a methodical, scientific reconsideration of all possible combinations of uncertainties that affect the safety functions and evaluation of each combination that is judged to be at all credible.

Confidence in the key results of radionuclide transport and risk calculations is enhanced by comparison with simple, analytical models (using the same data as the fully quantified numerical models). It is shown that the simple models can often closely reproduce the results of the detailed models.

Additional analyses to support SR-Site include consideration of future human action scenarios (including drilling and excavation into the repository); analyses to demonstrate optimisation and use of best available technique (BAT); and a systematic consideration of all features, events and processes (FEPs) that had been omitted from the safety assessment to determine any potential impact in the light of the completed scenario and risk analysis.

SR-Site provides a qualitative discussion of a reasonable evolution and consequences of the repository beyond the one million year assessment timescale, which links to arguments using natural analogues. Analogues of the KBS-3 repository materials and the processes affecting them are also discussed qualitatively. The view of SR-Site is that presentation of
natural analogues can provide support to long-term safety analyses by improving general perception and understanding of the concept of a geological repository.

2.3.3 Safety assessment approach and modelling

SR-Site adopts an eleven step approach to safety assessment, namely:

1. FEP processing
2. Description of the initial state (including the initial state of the engineered barriers and repository layout)
3. Description of external conditions
4. Compilation of process reports
5. Definition of safety functions and safety function indicators
6. Compilation of input data
7. Definition and analyses of reference evolution
8. Selection of scenarios
9. Analyses of selected scenarios
10. Additional analyses
11. Conclusions

The SR-Site main report is structured around these eleven steps. The initial state in SR-Site is defined as the state at the time of deposition/installation for the engineered barrier system and the natural, undisturbed state at the time of beginning of excavation of the repository for the geosphere and the biosphere. The evolution of the natural system is, therefore, at least in some aspects, followed from the time of beginning of excavation in the safety assessment. Any potential deviations from the initial state (for example that may arise from any accidents or omissions in the operational phase) can be considered as variant scenarios in the post-closure safety case.

Step 4 addresses the identification and handling of all relevant processes. These processes are identified based on earlier studies and FEP screening. Each relevant process is documented in one of the three process reports covering i) the fuel and canister, ii) the buffer, backfill and repository closure, and iii) the geosphere, with the following information:

- General description
- Dependencies between process and system variables
- Boundary conditions
- Model and experimental studies
- Natural analogues / observations in nature
- Time perspective over which process is relevant
- Handling in SR-Site safety assessment
- Handling of uncertainties
- Adequacy of references to support handling in SR-Site

The process reports provide a ‘recipe’ for handling the various processes, which is summarised in a process table that states whether a process is quantitatively modelled or can be neglected in all or specified conditions. The quantitative models generally include several interacting processes, often occurring in different parts of the system (and hence described in different process reports). The models form a network, with results from one model used as input to another. This model network is described graphically as an Assessment Model Flowchart (AMF). Two AMFs were developed for SR-Site, one representing the construction and operational phase and evolution during the initial
temperate period; and the second describing modelling under glacial and permafrost conditions. These two AMFs are illustrated below as Figure 2-1 and Figure 2-2.

Figure 2-1. The assessment model flow chart for the excavation/operation period and the initial temperature period after closure. In the AMF modelling activities covered in the present report can be seen as yellow rounded rectangles. Input data and results (input data to other models) presented in the Data report (DR) appear as blue rectangles (with reference to the section in the Data report). Data in other reports are presented as rectangles.
Figure 2-2. The assessment model flow chart for the permafrost and glacial conditions. In the AMF modelling activities covered in the present report can be seen as yellow rounded rectangles. Input data and results (input data to other models) presented in the Data report (DR) appear as blue rectangles (with reference to the section in the Data report). Data in other reports are presented as rectangles.

The AMFs identify a large number of modelling activities, which are undertaken using a wide range of computer codes. These codes are all documented in the Model Summary Report [15], stating how each code is used, why it is suitable for the task, how it has been developed and verified and how its input and output data are handled.

2.4 Safety case presentation

The SR-Site safety case main report runs to almost 900 pages (presented in three volumes, including a summary section). This main report is supported by:

- Six production reports, addressing:
  - Spent fuel for disposal;
  - Design, production and initial state of the canister,
Design, production and initial state of the buffer
Design, production and initial state of the backfill
Closure production
Underground openings construction

- FEP report
- Fuel and canister process report
- Buffer process report
- Backfill and repository sealing process report
- Geosphere process report
- Climate report
- Biosphere synthesis report
- Model summary report
- Data report
- Future Human Actions report
- Radionuclide transport report

Around 100 additional, more specialised, reports were produced in the SR-Site project. These are referenced either in the main report or in the supporting reports listed above.

The complete licence application runs to more than 10,000 pages. The summary section of the SR-Site main report runs to 38 pages and provides an overview of the main safety arguments. All reports are primarily written for technical audiences, and the graphics are mainly technical. There are few photographs and ‘non-technical’ illustrations, even in the summary section.

2.5 Conclusions and confidence

The overall conclusion of the SR-Site safety assessment is that a KBS-3V disposal facility that fulfils long-term safety requirements can be built at the Forsmark site. This conclusion is based on the long-term durability of the barriers of the KBS-3V concept, in particular the demonstration that the copper canisters with their cast iron insert provide sufficient resistance to the mechanical and chemical loads to which they may be subjected in the repository environment.

The properties of the Forsmark site are regarded as favourable for ensuring the durability of the barriers, as the groundwater flow through the repository will be limited due to the low density of water-conducting fractures in the rock at repository depth. (This was judged to be the main advantage of the Forsmark site over the Laxemar site\(^1\); the average spacing of conductive fractures at repository depth at Forsmark is over 100 metres, compared to 5-10 metres at Laxemar.)

A section of the concluding chapter of SR-Site discusses the confidence in the conclusions. Confidence is based on the knowledge of the Forsmark site obtained from the completed, surface-based investigations, which have demonstrated favourable conditions for safety and no site-related issues requiring resolution. Confidence is also found in the well-established KBS-3V design, with the engineered aspects of the repository system based on demonstrated technology. The underpinning series of “Production” reports support the SR-Site conclusions by explaining how the engineered barriers will be produced and emplaced, to achieve confidence in the required initial state of the disposal facility system.

\(^1\) Laxemar was investigated as an alternative site to Forsmark for the siting of a geological disposal facility.
National and international R&D and peer review studies are also cited as building confidence in the conclusions, together with the use of documented quality assurance routines and the complete analysis of issues identified as relevant to long-term safety according to an established assessment methodology.
3 Dossier 2005 Argile

3.1 Background and context

Agence nationale pour la gestion des déchets radioactifs (Andra) is the French public body responsible for the long-term management of all radioactive waste produced in France. Andra itself was established by the December 1991 French Waste Act as a public body and has the mission of implementing the deep geological disposal of high-level long-lived radioactive waste. Under the supervision of the French Ministries of Industry, Research and Environment, Andra operates disposal facilities adapted for low level radioactive waste. In addition, it is conducting a scientific research programme to study the possibility of deep geological disposal for high-level or long-lived radioactive waste. During the 1990s Andra investigated the possibility of developing a geological disposal facility in both granite and clay host rock formations and in 1998 the French government announced the decision to select the first underground laboratory in the clay formation of the Meuse/Haute-Marne site for further investigation into the feasibility of developing a geological disposal facility.

Dossier 2005 Argile was produced by Andra in December 2005 to assess the feasibility of a nuclear waste disposal facility in the context of a clay host formation. Dossier 2005 is based on the characteristics of the host formation and the first results were obtained from the Underground Research Laboratory at the Meuse/Haute Marne (MHM), which is located 500 metres underground in Bure, France. In this context, feasibility specifically refers to evaluating the possibility of constructing, operating and monitoring a reversible disposal facility in complete safety for man and environment. Establishing confidence in the safety assessment over long time scales (up to several hundreds of thousands of years) is an important aspect of this.

As mentioned on Andra’s website: “In the framework of the mission entrusted upon Andra by the Waste Act of 30 December 1991, the Agency presented to the Ministers for Research and Industry in 2005 a report on the feasibility of deep geological disposal for high-level and long-lived radioactive waste.” The dossier is intended to establish the framework for safety assessments and presents Andra’s adopted approach to demonstrating the existence of feasible technical solutions for geological waste disposal. At this stage, these solutions are posed as examples of the possible implementation of a waste facility in order to account for the fact that the disposal facility concepts may evolve over time. Therefore, the solutions presented in Dossier 2005 Argile are not intended to be optimised. Furthermore, the dossier is not advocating positioning the disposal facility at a particular site (as would be required to apply for a licence for construction and operation for example), but to assess its feasibility in a particular geological host formation (the Callovo-Oxfordian clay formation). This is undertaken by ensuring applicability of the results and studies at a known location (MHM underground research laboratory) are applicable to a larger zone. An area of 250 km² was thus defined. This approach ensures flexibility in selecting a site for the disposal facility.

Dossier 2005 Argile incorporates feedback from a peer review of the previous Dossier 2001 evaluation, which was aimed at reviewing Andra’s programme against internationally accepted practice. This review was organised by the OECD-NEA and included international experts from waste management organisations, safety authorities’ and/or their delivery partners. Dossier 2005 Argile has also been subject to a range of internal and external peer reviews, including the one organised by the OECD-NEA.

3.2 System description

3.2.1 Disposal inventory

Dossier 2005 Argile considers two broad waste categories:
1) High-level waste, also called C waste, is the reprocessing residue of spent fuel and consists of fission products and minor actinides separated during fuel recycling. These residues are vitrified in a glass matrix. High-level waste constitutes a very small volume of the total waste inventory, but contains the majority of the total radioactivity.

2) Intermediate-level waste, also called B waste or medium-level waste, originates mostly from the reprocessing of spent fuel (cladding, hulls and end caps). It also comes from residues (waste from effluent treatment, equipment, etc.) originating from the operation and maintenance of nuclear facilities. Medium-level waste exists in a wide variety of forms and represents the major part of the total waste volume (estimated to be between 70,000 and 80,000 m³).

Although not strictly speaking waste, Andra considers the possibility of the disposal of spent fuels (denoted by the letters ‘CU’) in Dossier 2005 Argile. These recyclable materials have a high activity and are potentially a significant source of heat generation. Uranium oxide (UOX) and mixed oxide (MOX) spent fuels are the two broad classes of spent fuel considered.

3.2.2 Geological environment

The MHM area is part of the eastern border of the Paris Basin. This basin is composed of an alternation of sedimentary strata with a clay dominance and limestone strata. In order of increasing depth, these are:

- a few cretaceous clayey-sandy superficial deposits of little thickness, crowning the highest points of the topography;
- the Tithonian limestones (called the Barrois limestones), which outcrop at the site;
- a marly Kimmeridgian;
- a limestone formation of the middle to upper Oxfordian;
- the Callovo-Oxfordian clay formation (the considered host rock). In this zone, the Callovo-Oxfordian is a homogeneous, hardly permeable stratum with its top located at a depth varying from 420 m to more than 600 m in the direction of the dip and its thickness also progressively varying from 130 m to the South to 160 m to the North of the zone; and
- the Dogger limestone formation overlays marls and binding clays.

The rock’s properties are well-known because of a major characterisation programme utilising information from boreholes, laboratory studies and field work. The proposed host formation has a low local hydraulic (vertical) head gradient and the surrounding water-bearing formations have low head gradients horizontally. Seismic surveys have not identified any faults in the host layer. The numerous analyses from borehole data, and from the underground laboratory indicate the formation is relatively homogeneous in terms of its properties.

3.2.3 Disposal concept

In an iterative approach between design and safety, Andra assigned safety functions to the components of the repository having a significant role, such as the host formation, waste packages and sealing components. The characteristics of these components relevant to safety were determined by taking into account the interactions with the environment and the possible uncertainties. The design of a “multi-function” system completes the notion of a “multi-barrier” system. Some components contribute to fulfilling the same function (‘complementarity’) or maintaining the function in case one of them fails (‘redundancy’).
The repository is expected to:

- prevent water circulation: the host formation (Callovo-Oxfordian) contributes to this, as well as the disposal facility design;
- immobilise the radionuclides within the package: the waste contributes to this objective, as well as the containers and the chemical conditions in the disposal cells; and
- delay and attenuate the migration of the radionuclides potentially released outside the disposal cells.

Various waste inventory production scenarios are considered in Dossier 2005 Argile. They were selected in order to provide access to a wide range of waste types, including hypothetical waste types, to address various relevant issues for the study of disposal facility performance.

For ILW ("B"), the solid wastes are either conditioned in bitumen, in concrete, or by compaction. The conditioned wastes will then be placed in concrete or steel drums, known as the primary package. The primary packages will then be placed in a concrete container to form a waste disposal package for handling and emplacement within disposal cells (cavities) within the disposal facility's concrete-lined disposal tunnels. The disposal cells for the ILW waste categories are proposed to be physically separated from the high-level ("C") waste zone to provide independence of waste evolution between the different waste types within the facility. The disposal tunnels are not proposed to be backfilled in order to allow reversibility. Andra presents a design (dimension of cells, emplacement of waste packages) which aims to minimise void spaces as recommended by the Basic Safety Rule. The ILW cells and the disposal tunnels and shafts will be sealed with bentonite (for its containment properties) surrounded by a concrete material (for mechanical constraints).

Figure 3-1. Illustration of intermediate-level waste containers (reproduced from [7] © Andra 2005)
Vitrified HLW is placed in stainless steel containers with welded lids. Each of these packages will then be placed within a thick carbon steel overpack and a low-alloy steel sleeve to enable handling and possible removal of packages for reversibility purposes. The main post-closure safety role of the packaging is to delay any contact of the waste with the geological environment before the temperature has stabilised to a level where radionuclide migration is well understood (through use of underpinning data and/or numerical models).

In the disposal cells, the high-level waste and spent fuel disposal packages are proposed to be arranged horizontally inside a metallic lining. Buffer packages, similar to the disposal packages but filled with an inert material, are proposed to be placed between each disposal package to contribute to maintaining the temperature inside the disposal cells below 90°C. This is the temperature domain covered by current knowledge and by the ability to understand phenomena and their couplings into the host clay formation.

Figure 3-2. Illustration of the disposal of high-level waste containers (reproduced from [21] © Andra 2005)

3.3 Safety case approach

3.3.1 Regulatory requirements

In accordance with the regulatory guidance in particular the Basic Safety Rule\(^2\) (RFS III.2.f), the main long-term safety indicator is the committed dose\(^3\) at potential outlets to the surface environment within the context of a critical group (representative of individuals

\(^{2}\) Issued in 1991 by the Nuclear Safety Authority in France.

\(^{3}\) In radiation protection or radiology the committed dose is a measure of the stochastic health effect on an individual due to an intake of radioactive material into the body.
susceptible of receiving the highest doses/chemotoxic impacts) and typical biosphere(s).
The Basic Safety Rule, which provides a basis for common discussion and understanding between the safety authority and Andra, recommends that this dose must remain lower than 0.25 mSv/year for a normal evolution for a period of at least 10,000 years. This radioactive dose corresponds to one quarter of the regulatory limit for public exposure of non-natural origin and approximately one tenth of the annual dose due to natural radioactivity.
Andra’s approach to defining a normal situation is described in Section 3.3.3 and, in brief, is thought of as the most likely evolution of the system. In keeping with international practice, Dossier 2005 Argile extends performance assessment calculations to a period of one million years, which is considered to be the time appropriate to radioactive decay of the wastes and repository evolution processes.
Other relevant objectives for geological disposal presented in the Basic Safety Rule include:

• the necessity of a multi-barrier disposal concept, namely the waste packages, the engineered barriers and the geological formation itself;
• the long-term geodynamic stability (in particular, no significant earthquake risk);
• no significant water flow in the host formation;
• containment properties of the host formation with respect to the radionuclides (to be determined and proposed by iterative assessment by the disposal facility designer);
• sufficient depth to protect the waste from various intrusions; and
• no exploitable outstanding natural resources in the vicinity.

The Basic Safety Rule provides a framework for the studies to be conducted. It requires safety to be quantitatively evaluated by the means of “situations” that encompass different possible evolutions of the repository.

3.3.2 Safety arguments

In respect of the Basic Safety Rule, the disposal facility is designed to isolate waste from natural surface phenomena and human activities.

Two overarching principles are employed to demonstrate “passive” post-closure safety within Dossier 2005 Argile:

1. Robustness: “The notion of robustness means that the characteristics of the elements comprising the repository must be such that they can guarantee maintaining their functionalities against reasonably imaginable disturbances despite residual uncertainties.”

2. Demonstrability: i.e., concepts are selected so that their level of safety can be checked as easily as possible and without calling for complex demonstrations.

These principles are established in practice by making use of multiple lines of reasoning, including safety assessment calculations, qualitative reasoning, analogies, experiments and/or by technological demonstration. As a means to build understanding about performance assessment, Andra has assigned a number of safety functions to those components of the disposal facility having a significant safety role, such as the host formation, waste packages, and other engineered elements (achieved through functional analysis). The concept aims, at first, to immobilise radionuclides inside the facility, then delay and attenuate the migration of any radionuclide that might be released from the waste. Over varying timescales, the disposal facility is designed to:
• Limit water flow and prevent shortcuts for the migration of radionuclides from the disposal system. The host formation contributes to this as well as the design of the disposal layout itself;

• Limit the release of radionuclides (immobilisation): this is fulfilled by the wasteform, the containers and the chemical conditions in the disposal cells; and

• Delay and attenuate the migration of radionuclides potentially released outside the disposal cells. This is fulfilled by the host formation which is the main transfer pathway, and the seals of structures.

These functions are provided over time by various and independent components (operating within relevant time frames). A key safety argument resides in the properties of the Callovo-Oxfordian argillites, notably its low permeability, retention capacity, geochemical properties and hydrogeological environment. The age of the rock formation, its tectonic stability and the proposed disposal facility depth provide confidence that these properties will remain very stable over the assessment timescale (i.e. over several hundred thousand years). The low-permeability clay host rock will act as a barrier to radionuclide migration, which will be diffusion controlled. The chemical composition of the interstitial water in the clay is also expected to cause the precipitation of various radionuclides as solid minerals, thus limiting radionuclide migration in groundwater. Furthermore, the clay layer contains a large proportion of smectite, a mineral that will tend to retard radioactive elements dissolved in groundwater, either by ion exchange or by sorption. The disposal layout and design are intended to preserve these favourable properties of the Callovo-Oxfordian argillites, in particular by minimising the disturbance caused by excavating the facility, the materials introduced and the presence of waste (notably with regard to released heat).

3.3.3 Safety case approach

Andra’s approach to understanding safety of the disposal system is broken down into the following aspects, which are also illustrated in Figure 3-3 below:

• define expected safety functions and related requirements;

• acquire knowledge to substantiate system behaviour and its evolution with time;

• acquire knowledge on engineering solutions;

• analyse uncertainties to define a technical solution (design) to mitigate uncertainties;

• define a number of scenarios (split into normal evolution scenario (NES) and altered evolution scenarios (AES) to cover remaining uncertainties;

• quantitative assessment of performance and safety indicators (including dose); and

• provide feedback from the assessments to aid design and further knowledge acquisition.
The safety approach is based on the behaviour of the disposal system described on the basis of the body of available scientific knowledge. Andra embodies this knowledge in the form of phenomenological analysis that provides a simplified but cautious representation of the phenomena and their evolution over time. Through analysis of thermal, hydrological, mechanical and chemical (THMC) processes, models, and the body of scientific information Andra has defined what is referred to as a Phenomenological Analysis of Repository situation (PARS).

From the PARS and the Function analysis, a qualitative safety analysis (QSA) was conducted by Andra in which there is a systematic analysis of uncertainties on THMC processes and their effects on safety functions.

The QSA contributes to the evaluation of the robustness of the repository by exploring possible impaired functioning of the disposal system (degradation of performances, waste packages defects, cover failure, crosscut of the Callovo-Oxfordian). As a result, Andra
established scenarios which are simplified descriptions of the repository initial state and its evolution with time.

A normal evolution scenario (NES) provides a bounding value for all likely or probable future evolutions and depicts the expected evolution of the disposal system in which all components fulfil their expected functions. “Altered evolution scenarios” (AES) define “altered situations” that encompass unlikely events and are based on a breakdown of a safety function (as regards to results from the QSA). In these cases, the altered scenarios may not necessarily represent physically possible situations.

Andra’s assessment aims to overestimate the disposal system’s impact from a “cautious” viewpoint without being excessively pessimistic. The normal evolution scenario itself is described as being “…based on cautious hypotheses. This scenario is not intended to represent the future reality, but rather to encompass the most probable situations through a cautious approach.” It is inextricably linked with a safety calculation model that is used for system evaluation, yielding a quantified performance measure. The long-term calculated dose is considered as an indicator of the impact and not a prediction of this impact.

To be comprehensive, the qualitative safety analysis is complemented by comparing the results with analyses conducted internationally including comparison with international FEP databases (e.g. NEA Database, Nagra Database).

In addition to the assessment of the consequences of radioactivity, the chemical impact of the disposal facility is also studied and quantitative assessments for a list of chemotoxic species that are potentially the most harmful are included.

Modelling approach

Andra makes use of modelling for a variety of aspects of its safety assessment and the broad classification of different types of models as follows:

1. A best-estimate model

This designates a model that is based on the most comprehensive understanding of the process to be modelled or offering the best match between observations and the numerical results. These include basic physical models and mechanistic models representing Fick’s law or Darcy’s law, for example. Examples of the latter include all models subject to a broad-reaching experimental validation and/or a sound international consensus among experts in the field.

2. A conservative model

This designates a model for which it is possible to show that its use tends to overestimate the disposal system’s impact, in light of the associated parameter uncertainties of all the relevant processes.

3. A pessimistic model

This designates a model that is not based on process-level understanding, but which is used exclusively to confidently overestimate the disposal system’s impact.

4. An alternative model

This designates a model that is not considered to be closest to the process-level understanding, but is offered as an alternative interpretation, although it cannot be classified on a phenomenological, conservative or pessimistic scale.

Furthermore, the same four categories apply to the choice of model values. However, a disposal facility’s components’ most probable behaviour is represented with conservative value choices where there is uncertainty. The exception to this is in Andra’s probabilistic calculations which sample parameter distributions based on available data and, in some instances, expert judgement (see subsection on Uncertainty further down).
Andra does not specifically refer to a hierarchy of models in the same way that RWMD does. However, it is acknowledged that there are several overlapping process models that can be used to account for a given phenomenon. From this, the ALLIANCES\(^4\) project was set up to produce an efficient, user-friendly simulation environment to enable the user to integrate computer codes from various origins and couple the various phases of calculation within a single environment. The Alliances structure is centred on a common data model forming a pivot of communication between each of the calculation modules.

Andra made use of “Alliances version 1” within the Dossier 2005 Argile which was a helpful tool for informing repository design as it was capable of undertaking performance and safety assessments. Its development included the main safety calculation characteristics and incorporated:

- a large amount of data;
- complex calculation sequences involving different models;
- a large calculation volume (several thousand simulations); and
- a requirement to control data and results.

Andra’s aspiration is that, in the future, the platform could be used to perform multi-physical simulations. It would then be a complete modelling, analysis and design tool for disposal system study which could provide support in a development or construction phase.

Andra’s main presentation of safety is through deterministic calculations based on a series of scenarios (NES, AES) reflecting different potential situations. These assessments are complemented by a probabilistic analysis of the system. This assessment considers molar flow rates from the Callovo-Oxfordian and access structures, but the distribution of radiological impact is not assessed. This is in accordance with the Basic Safety Rule. The objective is to identify the parameters which, due to their uncertainty, have the greatest influence on the uncertainty of the result; however, Andra clearly states that its main safety approach is to use deterministic assessments. Probability distributions are only established for the parameters relevant to the engineered structures and the geological barrier, such as permeabilities, porosities, diffusion coefficients, solubilities, etc. The probability distributions of each of the parameters and associated parameter correlations are established on the basis of the variability of experimental data and their correlations as well as possible constraints imposed by the Alliances tools (available distributions, types of correlations that could be considered, etc.). For each physical indicator, three statistical coefficients have been calculated according to time and the maxima of the indicator, to examine the strength of correlation between the input parameter and results. Several conclusions are drawn, with the main one being that the sorption coefficient of iodine in the Callovo-Oxfordian is identified as the dominant parameter.

**Scenarios**

The management of scenario uncertainties (related to safety functions and disposal system performance) falls into one of the following approaches:

- Andra’s preferred approach is “risk management”, i.e. that where possible the disposal facility should be designed to make the system robust, or insensitive, to uncertainties (e.g. performance of the engineered system with regard to seismicity of area).

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\(^4\) The goal of the ALLIANCES project, developed in collaboration between CEA (Commissariat à l’Energie Atomique) ANDRA and EDF (Electricité de France), is to provide a working environment for the simulation and analysis of phenomena to be taken into account for waste storage and disposal studies. More information can be found using [http://www.opencascade.com/customers/successmain/alliances](http://www.opencascade.com/customers/successmain/alliances).
• The second method involves including the risk or uncertainty in the normal evolution assessment of the disposal system’s performance, i.e. by taking into account the possibility by choosing the model or parameters values for each process on the knowledge acquired which may be pessimistic worst-case scenario for normal evolution, either in the reference calculation, or in the sensitivity analysis.

• For scenario uncertainties that cannot be dealt with using the two previous methods, the sequence of events leading to a variant scenario, must be identified and evaluated.

• In addition to the normal evolution scenario, the altered evolution scenarios considered in Dossier 2005 Argile are:

1. A “package failure” scenario in which a series of packages are assumed not to fulfil the expected functions;
2. Scenarios where all, or a part of, the sealing system is assumed to have failed;
3. Scenarios where an unintentional human intrusion event disturbs the disposal facility, for example a bore-hole intercepting a drift or a disposal cell for each type of waste; and
4. A scenario considering hypothetical simultaneous degradation of repository functions.

Management of Uncertainty

Dossier 2005 Argile includes a specific chapter on the treatment of uncertainty within the safety evaluation. Andra treats uncertainties according to the time periods, components or parts of the disposal system and its environment to which they relate; and classifies the type of uncertainty according to, for example, waste inventory, scientific knowledge, etc. To illustrate this, the types of uncertainties identified include:

• Uncertainties in the behaviour of the materials and the packages (metals, cements, glasses, etc.) and their degradation over time. Andra recognises that a point will be reached where their behaviour can no longer be reasonably predicted and hence the materials can no longer be considered to play a favourable role for safety. Nonetheless, if the environmental conditions are controlled, these materials may retain their favourable properties over very long periods.

• Following initial excavation of the disposal facility there will be a period when the thermal, mechanical and hydrological processes are disturbed before settling into equilibrium. It is recognised that the time to reach equilibrium and the nature of the equilibrium can involve uncertainties.

• Uncertainty in the surface environment increases with time because of major climatic evolutions, such as the periodic glaciations.

Three principles guide Andra in building confidence when dealing with uncertain parameters and processes:

1. A reference knowledge base is used that details the most complete understanding at the time of writing. This helps to identify any knowledge gaps.

2. A systematic approach to system analysis. As an example, Andra developed an inventory of the processes and associated models for each of the disposal system scenarios. This in return highlighted the limits of knowledge or understanding and identified the corresponding uncertainties.
3. Care in the traceability of model and parameter choices, and in defining variation ranges for these. When creating conceptual models representing the broad processes within the disposal system, the various parameters, models or data used are systematically listed, and are described in terms of conceptual understanding uncertainties, ranging from the values best representing the available data and reasonably covering possible variability in that data over space and time, to conservative or even pessimistic values that take all the uncertainties into account.

Uncertainty is the subject of a systematic study that identifies:

- the component affected by the uncertainty with, if relevant, the effects caused by one component on another by means of a perturbation;
- which performance aspects of safety functions can become altered. A qualitative, but reasoned assessment, including the use of specific calculations if relevant, is conducted on the risk of a significant reduction in the expected performance;
- the time period involved if applicable, and if such information is useful.

Then, component by component, measures to manage uncertainties are proposed such as:

- design measures;
- definition of calculation cases:
  - within the “normal evolution scenario” and its sensitivity analysis (by adjusting the level of conservatism for the parameters for example); and
  - within the altered evolution scenarios and their sensitivity analyses.

Uncertainties associated with parameter values are considered on case-by-case bases, by the experts proposing the values and models, in the documents that provide the framework for the modelling process. If the degree of uncertainty relating to the models or parameters is low to moderate, the parameters or models regarded as having greatest scientific justification are chosen for the reference calculation. Where it is important to study the impact of a more unfavourable (i.e. conservative/pessimistic) value, this parameter is investigated in a sensitivity study. Where the degree of uncertainty is high, the conservative or pessimistic parameter value or model is selected. Andra has taken expert judgement in several instances over the choices of parameter values. Where this is the case, the value and the brief reason(s) for this choice are provided such as, for example, “this value is considered most likely in the repository context”.

For the complementary probabilistic calculations, the probability distributions of each of the parameters and the correlations between parameters are established on the basis of the variability of experimental data and their correlations as well as possible constraints imposed by the Alliances tools (available distribution laws, types of correlations that could be considered, etc.).

3.4 Safety case presentation

Andra refers to Dossier 2005 Argile through a series of reports at varying levels of detail as shown in Figure 3-4. At level 1, is a synthesis report which provides an overview of Andra's approach to developing a solution for implementing a disposal facility. This report makes greater use of figures and presentation aspects than the supporting level 2 volumes. There are three level 2 assessment volumes, or ‘tomes’, that are interlinked with some sections repeated within the documents where the information is thought to be relevant (and this is noted within the text). The level three volumes present the overall design of the solution, the evolution of the disposal facility and the safety evaluation (qualitative and quantitative), both for operational and post-closure safety (the second part being more developed at the stage of the feasibility in 2005).

More detailed underpinning reports (levels 3 and 4) are referenced throughout.
The level 3 reference documents detail the state of available knowledge and collectively highlight the gaps in understanding. Thus they help to identify the sources of uncertainty and to focus appropriate needs-driven knowledge acquisition.

The evolutionary analysis of disposal system is described through passages of descriptive text, which reference THMC and geological models, as well as diagrams.

Andra has produced various leaflets designed for non-technical audiences. In addition, Andra has produced a high-level 40 page summary document for a non-technical audience covering different aspects of geological disposal in France.

Graphics are used throughout Andra’s documents to aid presentation and clarification of the discussion in the text. Diagrams range from graphs to illustrations of the disposal facility and technical drawings of container designs. In order to present results more clearly, the majority of graphs are included in the body of the safety evaluation volume in thumbnail format, to avoid breaking up the text too often, and shown in full in the Appendix (see Figure 7-1 in section 7 for an illustration of this).

### 3.5 Conclusions and confidence

Andra states that the feasibility of a disposal system based in an argillaceous host rock has been demonstrated with a reasonable degree of confidence. The geological host rock is of greatest importance with regard to ensuring long-term safety in the analysis undertaken in Dossier 2005 Argile. High confidence in the conclusions of Dossier 2005 Argile is conveyed in a number of ways. Firstly, the multifunctional concept gives the disposal facility a robust approach to reasoning the long-term safety of the disposal system.

Secondly, the conclusions of the safety analysis are relatively insensitive to the residual uncertainties. As such the altered scenarios do not lead to a significantly higher radiological impact than that of the normal evolution scenario.

Andra’s conclusions are based not only on the calculations of performance but also on the more qualitative analyses, dealing with the manner in which the design system is defined to fulfil the safety objectives and functions as well as to be robust to the uncertainties. The
calculations confirm that the proposed technical design solutions at the stage of 2005 are expected to effectively fulfil the safety functions which are assigned to them and that the individual radiological exposures are acceptable for all scenarios that have to be considered. The argillaceous host formation delays the potential release of radioactive substances over hundreds of thousands of years and after one million years, nearly all the radioactive elements are completely attenuated. Only iodine-129, chlorine-36, calcium-41 and selenium-79 show notable flows at the Callovo-Oxfordian layer boundaries.

In terms of potential gas production, Dossier 2005 Argile argues that there is only a small radiological inventory relevant to radiolytic gases and there is a high probability of the gases dissolving within the repository groundwater. Therefore, a gas pathway does not appear to be a very significant channel for the transfer of radionuclides.

Further confidence in Andra's studies and performance calculations is obtained through the agreement with experimental site data that has been acquired within the drifts of the MHM laboratory. These show good consistency with the values considered for Dossier 2005 Argile.

Dossier 2005 Argile was peer reviewed by the NEA in 2006 [23] at the request of the French Government. Overall, the review found Andra's scientific and technical programme to be fully consistent with international best practice and, in several areas, to be on the forefront for waste management programmes. The review commends many aspects of Dossier 2005 Argile and the way in which Andra document the approach to building confidence in safety and recognising further research needs.

Points of interest that RWMD may draw upon include:

- There may be merit in discussing any chosen approaches to building confidence in safety in an international context against that of other programmes in order to convey the advantages and limitations of any given approach: "[definition of safety functions and their further analysis] applied by Andra is an interesting contribution to the increasing use of functional approaches in safety assessments. Andra's method would benefit from being discussed and compared to others in an international context in order to better explore its advantages and limitations".

- It is important to not only consider the traceability of information within an organisation for any evaluation of safety, but also the accessibility of this information: "[the review team] found Andra's overall strategy for evaluating long-term safety to be credible and comprehensive... and contains formalised ways of integrating phenomenology, science and safety – including the integration of teams and personnel – which significantly contribute to credibility and traceability... however, the methodology is somewhat complex and demanding to comprehend, as illustrated by the fact that some of its fundamental aspects had to be clarified at late stages of the review and by evaluating a number of lower level documents."

- Due consideration should be given to how the safety analysis should inform design to ensure the needs of the safety assessment are fulfilled. One aspect noted in the review includes: "[the qualitative safety analysis] also has the potential to inform decisions concerning the repository design or the development of scenarios from first principles. This possibility could be further explored by Andra."
4 WIPP Compliance Certification Application

4.1 Background and context

The Waste Isolation Pilot Plant (WIPP) is a deep geological disposal facility designed to permanently dispose of defence related transuranic (TRU) waste in the state of New Mexico, USA. The WIPP is the culmination of part of a disposal programme which started in 1955 when the US Atomic Energy Commission (AEC) asked the National Research Council (NRC) to examine the disposal of radioactive waste disposal on land. The NRC Committee on Waste Disposal published an initial report [24] in April 1957 in which it stated:

“The most promising method of disposal of high-level wastes at the present time seems to be in salt deposits …”.

Following publication of this report, the Oak Ridge National Laboratory carried out several years of research to better understand the features and processes associated with the disposal of radioactive waste within a salt environment. This included field experiments to simulate the effect of waste in a disposal facility. In parallel the US Geological Survey investigated [25] salt deposits which may be suitable for the construction of a facility, identifying a number of broad locations, including the Permian basin underlying parts of Kansas, Colorado, Oklahoma, Texas and New Mexico.

In 1970 the AEC selected a disused salt mine in Lyons, Kansas as a candidate site for a high-level radioactive waste repository. This site was abandoned in 1972 after geologists discovered significant exploration boreholes which had been drilled in the site, and began to also encounter political opposition. A further nationwide search for a disposal facility focused on a site within the Permian basin approximately 30 miles east of the city of Carlsbad in New Mexico, in direct response to a request from the city of Carlsbad leadership to come and look at the salt in this area. A site was selected for field investigations. Following the drilling of a deep borehole, pressurised brine was discovered, leading to the site being abandoned in favour of a nearby replacement site, approximately 26 miles south-east of Carlsbad in Eddy County. This site was selected for detailed investigation. In 1979 the US Congress authorised (through public law 96-164) this site:

“...for the express purpose of providing a research and development facility to demonstrate the safe disposal of radioactive wastes resulting from defense activities and programs of the United States.”

Following a legal challenge from the Attorney General of New Mexico – which resulted in a Stipulated Agreement (including a Consultation and Cooperation Agreement) being signed by the state, Department of Energy and Department of the Interior – full scale underground excavation began in 1982. Finally, in 1992 the US Congress passed the WIPP Land Withdrawal Act which set aside the land required to build the facility5.

The WIPP Compliance Certification Application (CCA) is an application [8] by the US Department of Energy (DOE) for a GDF to dispose of transuranic (TRU) waste at the Eddy County site. In this context the application refers to compliance with the US Code of Federal Regulations (CFR).

The WIPP Land Withdrawal Act required the US Environmental Protection Agency (EPA) to act as regulator and issue regulations on the requirements for such a facility, to certify compliance with such regulations prior to emplacement of first waste, and then to re-certify

5 A history of the WIPP project is provided at [http://www.emnrd.state.nm.us/WIPP/wippchronology.html](http://www.emnrd.state.nm.us/WIPP/wippchronology.html) (accessed 2013).
compliance every five years thereafter prior to closure. The EPA added two sets of regulations to the US Code of Federal Regulations, namely:

- 40 CFR Part 191 [26], ‘Environmental radiation protection standards for management and disposal of spent nuclear fuel, high-level and transuranic radioactive wastes’, which provides generic regulations for the disposal of spent nuclear fuel, and transuranic and high level waste; and
- 40 CFR Part 194 [27], ‘Criteria for the certification and re-certification of the waste isolation pilot plant’s compliance with the 40 CFR Part 191 disposal regulations’, which develops the generic criteria within Part 191 specifically for the WIPP facility.

40 CFR Part 191 is further split into three subparts:

- Subpart A, titled ‘Environmental Standards for Management and Storage’ which sets limits on the annual radiation dose experienced by members of the public due to the facility prior to closure;
- Subpart B, titled ‘Environmental Standards for Disposal’ which sets probabilistic limits on cumulative radionuclide releases to the accessible environment for 10,000 years following closure. It also sets limits on the annual radiation dose experienced by members of the public within the accessible environment throughout a 10,000 year period, assuming undisturbed performance (i.e. without human intrusion); and
- Subpart C, titled ‘Environmental Standards for Groundwater Protection’ which sets limits for radionuclide contamination in groundwater for 10,000 years after disposal.

The regulations also require that DOE demonstrates sufficient confidence that these regulations can be met. Much of the methodology for the probabilistic performance assessment (including consideration of human intrusion) is defined within the regulations.

The WIPP CCA was submitted to EPA in October 1996 and concludes that the planned facility would meet the required regulations. In May 1998 EPA announced that the application complied with 40 CFR Part 191, Subparts B and C, and that the facility was therefore deemed safe for the disposal of waste. As a result, the facility began accepting waste on 26th March 1999, and since this date approximately 150,000 waste containers have been emplaced; the facility also continues to operate to date.

Two successful re-certifications (which are required periodically by law) have also occurred:

- WIPP Compliance Recertification Application (CRA) by DOE, March 2004 [28], with a WIPP Compliance Recertification Decision by EPA, March 2006; and
- WIPP Compliance Recertification Application (CRA) by DOE, March 2009 [29], with a WIPP Compliance Recertification Decision by EPA, November 2010.

The EPA emphasises that:

“Recertification is not a reconsideration of the decision to open WIPP, but a process to reaffirm that WIPP meets all requirements of the disposal regulations. The recertification process is not used to approve any new significant changes proposed by DOE; any such proposals will be addressed


separately by DOE. Recertification ensures that WIPP’s continued compliance is demonstrated using the most accurate, up-to-date information available.”

Although a number of changes occurred between the original CCA and subsequent CRAs, these differences are not important to the high-level discussion provided in this study.

4.2 System description

4.2.1 Disposal inventory and waste types

The WIPP facility is certified for the permanent disposal of transuranic waste. DOE Order 5820.2A [30] defines TRU waste as

“waste that is contaminated with alpha-emitting transuranium radionuclides [of atomic number greater than 92] with half-lives greater than 20 years and concentrations greater than 100 nanocuries per gram”.

The bulk of the waste destined for the WIPP has been produced over the past 70 years as part of the US nuclear weapons programme. Plutonium contaminated material (PCM) forms a large part of the waste and includes items such as protective clothing, glassware and laboratory equipment which has become contaminated during use (see Figure 4-1).

Figure 4-1 X-ray of a drum containing typical contact-handled WIPP waste

Waste within WIPP is separated into two broad classes, namely contact-handled (CH) TRU waste and remote-handled (RH) TRU waste. Contact-handled waste emits primarily $\alpha$ and $\beta$ radiation and can be safely handled by operators in close proximity without special shielding requirements. In contrast, remote-handled waste emits penetrating $\gamma$ radiation and/or neutrons and therefore requires shielding to be present during transport and

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8 This included significant updates to structure and presentation of the application, changes to quality assurance programme documentation, and a reconsideration of FEP screening, models and their parameters, and inventory estimates (which in turn lead to an updating of CCDF radionuclide release curves in recertification applications).
emplacement. Some material is also termed mixed-waste and contains both radioactive materials (regulated under the US Atomic Energy Act) and hazardous chemical compounds (regulated under the US Resource Conservation and Recovery Act). Around 96% of the volume of TRU waste for disposal is expected to be contact-handled.

4.2.2 Geological environment

The WIPP facility is constructed within the Salado formation of the Delaware Basin – a bedded salt deposit formed about 225 million years ago during the evaporation of an ancient ocean, called the Permian Sea. The site itself is located about 26 miles to the east of the city of Carlsbad in the state of New Mexico. The primary salt formation containing the WIPP is about 600 metres [2,000 feet] deep, being reached approximately 250 metres [850 feet] below the surface (see Figure 4-2). The Salado formation consists mainly of sodium chloride rock which is overlain by dolomite rock.

The basin contains natural resources, including oil and gas, together with significant quantities of potash. Although the location of the WIPP site was selected to minimise nearby natural resources, future mining poses a human intrusion risk to the facility which must be assessed in the safety case. As a result the WIPP Performance Assessment, contained within the CCA, must determine whether potential human intrusions in the future offset the otherwise favourable characteristics of the site.

![Figure 4-2 WIPP geological setting](image)

The WIPP site is part of the Delaware Basin and is geologically stable, with little earthquake activity.
4.2.3 Operations

Contact-handled waste

Contact-handled TRU waste is received onto the WIPP site in authorised waste emplacement containers. These containers range from a 208 litre [55 US gallon] steel drum to a 1878 litre [496 US gallon] standard waste box, with the complete list of possibilities contained within the compliance certification application and its updates. If required, sets of ten drums may also be overpacked using a ten-drum overpack, referred to as a ‘TDOP’.

Containers are then grouped on a pallet according to size and type, wrapped in plastic sheeting (to provide mechanical stability), and then stacked in rows in disposal vaults (see Figure 4-3) using a stacker truck. Each disposal vault is 91 metres [300 feet] long, 10 metres [33 feet] wide and 4 metres [13 feet] high, allowing it to hold approximately 12,000 208 litre [55 US gallon] drums (with a three-drum stacking height). After emplacing the waste, woven polypropylene sacks containing granular magnesium oxide (MgO) are placed on the top tier of the waste. These are intended to absorb any moisture and control the solubility of radionuclides in the event of a brine intrusion into the repository. In the long-term the MgO will be distributed between the waste containers as a result of the movement of the surrounding geological salt which is expected to creep and destroy the sacks holding the MgO.

Remote-handled waste

Remote-handled TRU waste is received onto the WIPP site in an RH-TRU waste canister (again described within the compliance certification application). Each canister is then transferred to a disposal vault (as described above) and emplaced into a horizontal borehole drilled into the wall of the vault. This operation is carried out using a specially designed system referred to as the Horizontal Emplacement and Retrieval Equipment (HERE), shown in Figure 4-4. Once the canister has been emplaced, the HERE further inserts a concrete shielding plug into the borehole. This plug, together with the shielding provided by the surrounding rock, allows the vault to be accessed by workers without additional shielding. As a result, the bulk space within the vault may be used for contact-handled waste via a stacker truck; this allows an efficient use of space within the facility.

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9 DOE initially emplaced additional mini-sacks of Magnesium Oxide between the waste containers and the walls of the disposal room. DOE proposed discontinuing this practice in 2000 for reasons of industrial and radiological safety to workers; EPA agreed to the change request in 2001.
4.3 Safety case approach

4.3.1 Regulatory requirements

As described in Section 4.1, the principal regulations for the WIPP are contained within 40 CFR Parts 191 and 194 [26, 27]. These regulations set down requirements on waste containment and groundwater protection, together with requirements on the form of compliance certification and re-certification applications, and participation by the public in the certification process.

The section on waste containment describes the basis of the assessment of post-closure safety of the facility. These are generally more explicit than the regulations for similar proposed European facilities. Safety of the facility must be determined through a numerical Performance Assessment (PA), defined within 40 CFR 191.12:

“Performance assessment means an analysis that:

(1) identifies the processes and events that might affect the disposal system;
(2) examines the effects of these processes and events on the performance of the disposal system; and

(3) estimates the cumulative releases of radionuclides, considering the associated uncertainties, caused by all significant processes and events. These estimates shall be incorporated into an overall probability distribution of cumulative release to the extent practicable.”

40 CFR 191.13 specifies the following high-level containment requirements:

“(a) Disposal systems for spent nuclear fuel or high-level or transuranic radioactive wastes shall be designed to provide a reasonable expectation, based on performance assessments, that the cumulative releases of radionuclides to the accessible environment for 10,000 years after disposal from all significant processes and events that may affect the disposal system shall:

(1) Have a likelihood of less than one chance in 10 of exceeding the quantities calculated according to Table 1 (Appendix A); and

(2) Have a likelihood of less than one chance in 1,000 of exceeding ten times the quantities calculated according to Table 1 (Appendix A).

(b) Performance assessments need not provide complete assurance that the requirements of § 191.13(a) will be met. Because of the long time period involved and the nature of the events and processes of interest, there will inevitably be substantial uncertainties in projecting disposal system performance. Proof of the future performance of a disposal system is not to be had in the ordinary sense of the word in situations that deal with much shorter time frames. Instead, what is required is a reasonable expectation, on the basis of the record before the implementing agency, that compliance with § 191.13(a) will be achieved.”

The referenced “Table 1 (Appendix A)”, listing the release limits specified by EPA for the WIPP, is reproduced below:

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>Release Limit $I_i$ per 1,000 MTHM$^{10}$ or Other Unit of Waste (curies)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Americium-241 or -243</td>
<td>100</td>
</tr>
<tr>
<td>Carbon-14</td>
<td>100</td>
</tr>
<tr>
<td>Cesium-135 or -137</td>
<td>1,000</td>
</tr>
<tr>
<td>Iodine-129</td>
<td>100</td>
</tr>
<tr>
<td>Neptunium-237</td>
<td>100</td>
</tr>
<tr>
<td>Plutonium-238, -239, -240, or -242</td>
<td>100</td>
</tr>
<tr>
<td>Radium-226</td>
<td>100</td>
</tr>
<tr>
<td>Strontium-90</td>
<td>1,000</td>
</tr>
<tr>
<td>Technetium-99</td>
<td>10,000</td>
</tr>
<tr>
<td>Thorium-230 or -232</td>
<td>10</td>
</tr>
</tbody>
</table>

$^{10}$ Metric tons of heavy metal exposed to a burnup between 25,000 megawatt-days per metric ton of heavy metal (MWd/MTHM) and 40,000 MWd/MTHM.
Tin-126 1,000
Uranium-233, -234, -235, -236, or -238 100
Any other alpha-emitting radionuclide with a half-life greater than 20 years 100
Any other radionuclide with a half-life greater than 20 years that does not emit alpha particles 1,000

To supplement 40 CFR 191, the EPA also produced a further federal regulation which clarifies the required approach to performance assessment for the Certification and Recertification of the WIPP compliance. Federal regulation 40 CFR 194.34 states:

"Results of performance assessments.

(a) The results of performance assessments shall be assembled into "complementary, cumulative distribution functions" (CCDFs) that represent the probability of exceeding various levels of cumulative release caused by all significant processes and events."

DOE defines a normalised release for comparison with the release limits specified above for releases to the accessible environment which contain a mix of radionuclides. This is defined through the following summation:

\[ R = \sum_{i} \frac{Q_i}{L_iC} \times (1 \times 10^6 \text{ Curies}) \]

where
- \( R \) is the normalised release [-];
- \( Q_i \) is the cumulative release of radionuclide \( i \) to the accessible environment during the assessed 10,000 year period after closure [curies];
- \( L_i \) is the release limit for radionuclide \( i \) given in the table above [curies]; and
- \( C \) is the total quantity of TRU waste emplaced into the repository [curies].

In the 2004 CRA the total quantity of TRU waste to be emplaced is listed as \( 2.48 \times 10^6 \) curies (\( 9.18 \times 10^4 \) TBq).

4.3.2 Safety arguments

The safety case for WIPP is principally reliant upon the isolating properties of the salt host rock surrounding the facility. WIPP literature presents the following advantages for a salt environment, and these underpin the more detailed safety arguments within the compliance certification application:

- The presence of a salt geology is itself an indicator of hydrodynamic stability (the Salado formation is believed to have existed for around 250 million years);
- The dry conditions imply the lack of a principal groundwater pathway for radionuclide migration;
- Salt is soft and easy to mine;
- Fractures which form in the surrounding geology during construction will self heal due to creep (while the porosity remains low) and this also removes air pockets around waste containers without the use of an additional backfill. However, in the region immediately around a container, waste degradation processes, such as gas production, may still occur and are explicitly considered in the application;
• Salt has approximately five times the thermal conductivity of typical crustal rocks; and
• Salt provides effective radioactive shielding to operators (similar to concrete) once the waste is emplaced. This provides protection to workers emplacing contact-handled waste in a vault which already contains remote-handled waste.

The compliance-certification application adds that the selected site:

• Has a host rock which is impermeable, relatively uncomplicated (both lithologically and structurally) and has little interstitial brine;
• Is unlikely to be affected by long-term climate changes within the 10,000 year regulated period;
• Is in an area where future groundwater use is likely to be low and there are no permanent surface waters;
• Has a predictable low flow of groundwater;
• Avoids endangered species and special habitats; and
• Is believed to be seismically stable.

The approach to safety within the WIPP CCA differs significantly from, for example, the SKB safety case, in that it places a low reliance on engineered barriers and a high reliance on the containing and isolating properties of the host rock for undisturbed performance.

4.3.3 Safety assessment approach and modelling

Features, events and processes
The CCA considers the FEPs applicable to the WIPP disposal facility. To compile the list, DOE started with a comprehensive FEP list developed for the Swedish Nuclear Power Inspectorate [31], to which a number of WIPP specific FEPs were added. FEPs relating to constructional, operational and decommissioning errors were not included within this initial list; instead it was assumed that DOE would introduce sufficiently robust management procedures that these errors are prevented from happening. In addition FEPs relating to the containment of hazardous metals, volatile organic compounds and other chemicals not regulated by 40 CFR Part 191 were not included within the initial list.

Once a comprehensive list of FEPs had been identified, the resulting list was screened to remove:

• FEPs that were excluded as a result of the regulations within 40 CFR Part 191 and 194 (for example due to differing assumed human activities or required performance measures);

The final WIPP FEP list is provided within the compliance certification application. In this FEPs are separated into:

• natural FEPs;
• waste and repository induced FEPs; and
• human initiated FEPs.

In total, 237 FEPs were listed in the original CCA although this number changed slightly within subsequent CRAs following a reassessment [32].

Once a comprehensive list of FEPs had been identified, the resulting list was screened to remove:

11 This list had been compiled by drawing together a number of existing international lists, including three from the UK programme.
• FEPs which were deemed to have a sufficiently low consequence (or which would have a beneficial consequence); and
• FEPs which have a sufficiently low probability of occurrence (as specified within the regulations).

FEPs which remained after the screening were taken through to the scenario development process [33] for the performance assessment. These scenarios utilise computer codes which model the future interaction of the repository with the natural system, for both undisturbed performance (when the disposal system evolves as expected) and disturbed performance (when a future human intrusion or certain unlikely natural events occur). The overall assessment approach is summarised in Figure 4-5.

![Figure 4-5 Schematic diagram showing the approach to assessing safety of the WIPP facility](image)

**Modelling**

The WIPP performance assessment lists three basic entities which are required to assess the long-term safety of the facility. These are shown in Table 4-2.

| EN1 | A probabilistic characterisation of the likelihood of different futures occurring at the WIPP site over the 10,000 year modelled period |
| EN2 | A procedure for estimating the radionuclide releases to the accessible environment associated with each of the possible futures that could occur at the WIPP site over the 10,000 year modelled period |
| EN3 | A probabilistic characterisation of the uncertainty in the parameters used in the definition of EN1 and EN2 |

EN1 requires an understanding of what is termed the *stochastic* (or aleatory\(^{12}\)) uncertainty in the system and EN3 of the *subjective* (or epistemic\(^{13}\)) uncertainty.

---

\(^{12}\) Aleatoric uncertainties are broadly those which cannot be suppressed by more accurate measurements, for example human behaviours in the future.

\(^{13}\) Epistemic uncertainty results from data we could know in principle, but don't in practice.
The required CCDFs for the performance assessment are generated using a single computational modelling system to represent the disposal region and surrounding geology for all the modelled scenarios.

**Figure 4-6 Computer Models used in the 2004 CRA Performance Assessment, and indicative flow of information (adapted from Figure PA-2 of the submission)**

Scenarios are defined as representations of potential evolutions of the disposal system, and are composed of specific combinations of FEPs, generated following the methodology advanced within [3]. The performance assessment considers six scenarios, one of which represents undisturbed performance and five of which represent disturbed performance.

‘Undisturbed performance’ is defined specifically within 40 CFR 191.12 (‘Definitions’) as:

> "Undisturbed performance means the predicted behavior of a disposal system, including consideration of the uncertainties in predicted behavior, if the disposal system is not disrupted by human intrusion or the occurrence of unlikely natural events."

‘Disturbed performance’ includes the effect of human intrusion into the facility, together with certain unlikely natural events. The five disturbed performance scenarios are shown in Table 4-3.
Table 4-3 Disturbed performance scenarios within the WIPP performance assessment

<table>
<thead>
<tr>
<th>Scenario</th>
<th>Definition</th>
</tr>
</thead>
<tbody>
<tr>
<td>E</td>
<td>Deep Drilling Scenarios</td>
</tr>
<tr>
<td></td>
<td>Any future which involves at least one deep drilling event which intersects the waste disposal area. These are further sub-divided:</td>
</tr>
<tr>
<td></td>
<td>• E1 – Any future with one inadvertent penetration of waste that also penetrates a Castile brine reservoir</td>
</tr>
<tr>
<td></td>
<td>• E2 – Any future with an inadvertent penetration of the waste that does not also penetrate a Castile brine reservoir</td>
</tr>
<tr>
<td></td>
<td>• E1E2 – Any future during which more than a single E1 events occurs</td>
</tr>
<tr>
<td>M</td>
<td>Mining Scenario</td>
</tr>
<tr>
<td></td>
<td>Any future which involves mining within the controlled area.</td>
</tr>
<tr>
<td>ME</td>
<td>Mining and Drilling Scenario</td>
</tr>
<tr>
<td></td>
<td>Any future which involves both deep drilling and mining within the controlled area.</td>
</tr>
</tbody>
</table>

To incorporate these scenarios, future events (the stochastic uncertainty) are determined using random sampling over the 10,000 year modelled period. Each potential future is defined by the following seven parameters, to each of which a probability distribution function is assigned:

1. The time of each intrusion;
2. The location of each borehole (implying the type of waste which was accessed);
3. The probability of accessing waste material, as opposed to salt rock, within that waste panel;
4. The activity of the waste accessed (if waste material was penetrated);
5. The type of borehole plugging used after the intrusion event;
6. Whether a pocket of pressurised brine is accessed in the Castile brine reservoir by the borehole; and
7. The time at which mining occurs (if mining occurs in this potential future).

A single realisation of the WIPP PA model utilises Latin Hypercube Sampling (LHS) to select the 57 parameter values which determine the undisturbed evolution of the system (the subjective uncertainty), i.e. rates of gas generation through corrosion of metal, degradation of organic material etc. For each realisation, 10,000 possible futures are determined by randomly sampling the seven parameters above. Each of these possible futures is run through the performance assessment modelling system (whose component codes are shown in Figure 4-6) to produce 10,000 possible curves for the normalised release, $R(t)$. A code then assembles these results into a single CCDF for each...

---

14 The term controlled area is defined by federal regulations as "(1) A surface location, to be identified by passive institutional controls, that encompasses no more than 100 square kilometers and extends horizontally no more than five kilometers in any direction from the outer boundary of the original location of the radioactive wastes in a disposal system; and (2) the subsurface underlying such a surface location."
realisation. A total of 100 realisations, together termed a replicate, are evaluated to produce 100 CCDF curves – as shown in Figure 4-7. From these a mean CCDF curve may be calculated for the system. To test whether results from the PA model are converged, three such replicates are evaluated (see Section 4.5) – as shown in Figure 4-8 – and the variation of the three mean CCDFs compared. As a result;

10,000 futures x 100 realisations per replicate x 3 replicates = 3 million calculations

were carried out as part of the PA. The 3 million calculations involved approximately 100 million intrusion events into the WIPP facility. Note that 3 million calculations does not necessarily imply that each individual code is run this many times as a result of the use of a scaling methodology (see Section 6.4.13.2 of the 1996 submission).

A further three replicates were calculated in 1997 as part of the Performance Assessment Verification Test (PAVT) required by the EPA after submission of the original CCA.

Full details of the 57 uncertain parameters which account for the subjective uncertainty in the system are listed in detail along with their justification within an appendix to the submission. Each description includes discussion of the parameter description, link to the PA database record, the computational code in which the value is used, PDF, units, discussion, link to WIPP data entry form and references.

As a result of the low radionuclide migration within the salt surrounding the WIPP facility, the greatest uncertainty that results in radionuclide release to the accessible environment lies with the parameters which define future human intrusions into the site through mining, deep drilling, or mining with deep drilling.

4.4 Safety case presentation

The 1996 CCA and 2004 CRA comprise a single volume containing the following nine chapters:

1. Introduction
2. Site characterisation
3. Facility description
4. Waste description
5. Quality assurance
6. Containment requirements
7. Assurance requirements
8. Individual and groundwater protection requirements
9. Peer review

The 2009 CRA introduced a different structure which more closely followed the format of the relevant federal regulations.

Each application provides a substantial level of additional detail through appendices, some of which include additional documentation through attachments. As an example, results of the performance assessment are contained within chapter 6 of the submission, which in turn references ‘Appendix PA’. This appendix is supplemented by additional attachments, for example:

- PA-SCR which provides information on all FEPs that have been considered, which have been screened out and which have been included in modelled scenarios, together with a reason for each screening decision;
• PA-PAR which provides information on parameter values that were used in the performance assessment together with justification and provenance for these parameter values; and
• PA-MASS which provides information on assumptions made within the performance assessment for each sub-model. This also includes a detailed discussion of each conceptual model and the FEPs to which it relates.

Information on inventory and package design is provided within Appendix Data, Attachments D and H [28].

Although the structure of the WIPP submission is not formally hierarchical, unlike RWMD’s generic Disposal System Safety Case or LLWR’s 2011 Environmental Safety Case, the use of appendices and appendix attachments creates a similar effect, and creates a document that can be read at several levels of detail (although each of these is generally more technical than, for example, a Level 1 LLWR document). Each chapter, appendix and appendix attachment contains a useful introductory section to provide a technical overview of the contents of that section.

4.5 Conclusions and confidence

The overall conclusion of the performance assessment is that the WIPP facility meets the Containment Requirements specified in 40 CFR 191, i.e. that the normalised release (as defined in Section 4.3.1) lies below the prescribed limits for the entire 10,000 year modelled period with sufficient confidence. This is demonstrated by the family of CCDFs within Figure 4-7 lying completely to the left of the EPA containment requirements line. The family shown contains 100 CCDFs.

Sufficient confidence in this result is defined explicitly within 40 CFR 194.34 [27], although it is left to DOE to decide how best to demonstrate compliance with these regulations:

“The number of CCDFs generated shall be large enough such that, at cumulative releases of 1 and 10, the maximum CCDF generated exceeds the 99th percentile of the population of CCDFs with at least a 0.95 probability ...

Any compliance application shall provide information which demonstrates that there is at least a 95 percent level of statistical confidence that the mean of the population of CCDFs meets the containment requirements of § 191.13 ...”

The method selected by DOE to demonstrate compliance (including convergence of the mean) is termed ‘multiple replication’ [34]. In this, a complete set of 100 CCDFs is generated (i.e. replicates of the data shown within Figure 4-7) three times using identical models and parameter values but with different random seed values. The variability of calculated results then allows the sufficiency of the Monte Carlo sample size to be evaluated and the implied confidence in model results to be determined. This is achieved by comparing the overall mean which is calculated from the three replicates with the upper and lower 95th percentile confidence interval of the Student-t distribution which has been estimated from these three individual mean CCDFs (Figure 4-8).

Through these results DOE state that they have demonstrated:
• that WIPP is in compliance with the containment requirements specified within 40 CFR 191; and
• that the sample size of 100 CCDFs (within each replicate) is adequate for sufficient confidence in this compliance.
A more detailed examination of the data within these CCDFs allows the contribution from different release pathways to be determined. The CCA groups radionuclide releases to the accessible environment into five categories:

- releases through transport in the groundwater;
- releases from cuttings;
- releases from cavings;
- releases from spallings; and
- releases from the direct release of brine at the surface during drilling.

Releases from cuttings and cavings dominate the mean normalised release, with releases from spallings and direct brine release being approximately an order of magnitude lower. The mean CCDF for releases through groundwater transport in the Salado or Culebra formation have $R < 10^{-6}$ EPA units and therefore make virtually no contribution to the total. Hence, releases due to human intrusion in the disturbed scenario account for essentially all predicted releases from the WIPP facility over the modelled period.

DOE states (6-174) that "In general DOE has not attempted to bias the PA toward a conservative outcome, and the mean CCDF represents a reasonable estimate of the expected and, in the case of future human activities including intrusion, prescribed, performance of the disposal system".

Figure 4-7 Distribution of CCDFs for normalised radionuclide releases to the accessible environment for the WIPP (replicate 1, reproduced from CRA 2004, Figure 6-34)
Figure 4-8 Overall mean from three replicate mean CCDF compared with estimated confidence intervals based on these replicates (reproduced from CRA 2004, Figure 6-38)
5 Yucca Mountain Total System Performance Assessment

5.1 Background and context

Early studies concerning the management of higher activity wastes in the USA focused on the potential for disposal in salt deposits, based in part on a National Academy of Sciences [35] report which suggested that this was the most immediately promising method of disposal. In the early 1970s the Atomic Energy Commission (AEC) began exploring an area of deep salt beds near Carlsbad, New Mexico as a potential repository site for high-level radioactive waste. Disposal at the site, which became operational in 1999 as the Waste Isolation Pilot Plant, was limited to defence-related transuranic waste.

The search for a suitable site for geologic disposal of spent fuel and high-level waste continued throughout the 1970s, first under the AEC and later under its successor agencies, first the Energy Research and Development Administration (ERDA) and finally the Department of Energy. Among the options considered were clayey shales of the Midwest; bedded salt formations in Michigan, Texas, and Utah; salt domes in Louisiana and Mississippi; basalt formations in Washington state at Hanford; welded volcanic tuff at Yucca Mountain in Nevada; and deep crystalline rocks in the north-central, north-eastern, and south-eastern United States. In 1986, three sites were chosen for detailed investigations; Hanford, Deaf Smith County in Texas, and Yucca Mountain. Congress revisited the issue of nuclear waste management in 1987 and the resulting NWPA Amendments Act of 1987 designated Yucca Mountain as the sole site to be considered for a permanent geologic repository for spent nuclear fuel and HLW [36].

The US Department of Energy (DOE) Office of Civilian Radioactive Waste Management submitted the Yucca Mountain Repository Licence Application to the US Nuclear Regulatory Commission (NRC) in 2008 [9]. The application was to build a repository for higher activity nuclear waste, including spent fuel (SF) and vitrified high level waste (HLW). In 2010, following the Obama Administration decision to halt work on a repository at Yucca Mountain, DOE sought permission to withdraw its licence application [37].

5.2 System description

5.2.1 Disposal inventory and waste types

Commercial SF, DOE SF (including naval SF), and HLW were planned for disposal at the repository. The inventory was limited by law to 70,000 Metric Tons of Heavy Metal (MTHM) [9], where mass of Heavy Metal refers to the mass of actinide elements (elements with atomic numbers greater than 89) in the fuel. This capacity limit was a legal limit, not a physical limit, and was to be allowed to be exceeded once a second repository was in operation. The Total System Lifecycle Cost estimate of 2008 evaluated a repository of 122,100 MTHM as a reasonably foreseeable scenario [38].

The majority of the SF was expected to consist of commercial SF, which comprised approximately 63,000 MTHM. Approximately 292,000 commercial SF assemblies (consisting of 167,000 boiling water reactor assemblies and 125,000 pressurised water reactor assemblies) were expected to be generated by 2040, requiring around 7,500 waste canisters. DOE SF represented an inventory of approximately 2,500 MTHM [9].

The majority of HLW considered for disposal came from the defence nuclear programme, of which approximately 900 canisters were expected to comprise vitrified glass containing plutonium. The DOE HLW allocation was 4,667 MTHM, representing approximately 9,334 HLW canisters [9].
5.2.2 Geological environment

Yucca Mountain is located in a desert area of southern Nevada, roughly 160 km northwest of Las Vegas and 80 km northeast of Death Valley in California. The Yucca Mountain site is on federal land, adjacent to the Nellis Air Force Bombing Range and the Nevada Test Site (now called the Nevada National Security Site) which was used in the past for nuclear weapons testing.

Yucca Mountain consists of a series of north–south trending ridges extending approximately 40 km. The ridge of Yucca Mountain reaches an elevation of 1929 metres above sea level [39]. Yucca Mountain is composed of successive layers of fine-grained volcanic tuffs, underlain by older carbonate rocks. The tuffs were formed when volcanic gas and ash erupted and flowed quickly over the landscape, then cooled and solidified or settled from the atmosphere. In instances when the temperature was high enough, the ash was compressed and fused to produce a welded tuff. Non-welded tuffs, which typically occur between welded layers, were consolidated at lower temperatures, and are both less dense and more porous.

The proposed placement of the repository was in a rock layer of welded tuff, known as the Topopah Spring Tuff unit, as indicated in Figure 5-1. The water table in this region lies between 500 and 800 metres below the ground surface. The repository would have been placed about 200 to 500 metres below the ground surface and would therefore have been approximately 300 metres above the water table. The deep water table and thick unsaturated zone result from a combination of relatively high relief, low annual precipitation, low infiltration rates, and high rates of evaporation in the arid environment.

In the unsaturated zone the processes of infiltration and subsequent percolation to the water table determine the amount of water that would seep into repository drifts and possibly come in contact with waste packages, thus enabling the dissolution of waste and transport of radionuclides away from the repository [39].

The Yucca Mountain area belongs to the Death Valley regional groundwater flow system. Groundwater moves from areas of recharge in the mountains of central and southern Nevada to areas of discharge south and west of the Nevada National Security Site and into Death Valley [40].
5.2.3 Disposal concept

The waste package would have consisted of a single design with six configurations. The different waste package configurations contained multiple internal structures and different external dimensions to allow acceptance of various waste forms. The waste package consisted of two concentric cylinders in which the waste forms would have been placed. The outer corrosion barrier was made of Alloy 22, a corrosion-resistant nickel-based alloy and the inner vessel in the design was stainless steel [9].

Before the inner vessel was sealed, the waste package would have been evacuated and helium added as an inert fill gas. The helium would help transfer heat from the wasteform to the wall of the inner vessel. Each waste package would be structurally supported by a pallet in an emplacement drift. Titanium drip shields would have been installed to protect waste packages from dripping water and rock-fall during the post-closure period [9]. The proposed design is illustrated in Figure 5-2.

The emplacement drifts were planned to be large circular tunnels, nominally 5.5 metres in diameter, used to emplace about 11,000 waste packages. The total subsurface emplacement area required to accommodate the waste packages is about 5 km². In addition to emplacement drifts, one of the panels includes up to two drifts dedicated to performance confirmation testing plus an associated observation drift [9].

It was intended that the waste packages would have been placed in the emplacement drifts in a line with spacing of approximately 10 cm and a line-averaged heat load of 1.45 kW/m. Pre-closure forced ventilation would have been active from the start of emplacement and continued until 50 years after the last waste package was emplaced [39].
5.3 Safety case approach

5.3.1 Regulatory requirements

The US Environmental Protection Agency (EPA) has responsibility for establishing standards for radionuclide releases and radiation exposures arising from the operation of waste management facilities. The NRC has the responsibility for licensing waste management facilities and for establishing licensing requirements for the construction and operation of these facilities. The NRC licensing requirements must be consistent with the applicable EPA standards.

EPA and NRC regulations have evolved over the course of investigations at Yucca Mountain, for example by increasing the timescale for assessments from 10,000 years to 1,000,000 years. They are more detailed and prescriptive than equivalent regulations which apply in other repository programmes [41]. As noted in the NEA-IAEA International peer review of the performance assessment supporting the site recommendation [41], this contributes to a tendency to focus more on demonstrating numerical compliance with quantitative criteria compared to other equivalent safety cases. The regulations define:

- The scope of the performance assessment, including criteria for the screening of FEPs, and defining characteristics of the “Reasonably Maximally Exposed Individual” (RMEI);
- The use of probabilistic performance measures implemented through a Monte Carlo performance and uncertainty analysis;
- The requirement for the identification and description of multiple barriers that contribute to waste isolation and containment;
Compliance limits for the estimated mean annual dose, groundwater concentrations and for individual protection following human intrusion. For example, the mean annual dose for a RMEI for 10,000 years post-closure must be no more than 0.15 mSv. The 1,000,000 year individual protection standard is that the mean annual dose is no more than 1 mSv; and

- The timescales over which performance must be assessed.

5.3.2 Safety arguments

The key features of the safety case for disposal at Yucca Mountain have been summarised in the following attributes [42];

- Limited water contact with waste packages. The location of the proposed repository is above the water table and the semi-arid climate limits percolation and drip-shields would divert any water from the waste packages.

- Long waste package lifetime. The packages are designed with a very robust corrosion-resistant outer barrier.

- Low rate of release of radionuclides from breached waste packages due to limited water in the near field, leading to a diffusion-controlled environment.

- Reduction in the concentration of radionuclides as they are transported from waste packages. Sorption to the fracture surfaces and dilution in the larger volumes of groundwater below the repository will act to reduce radionuclide concentrations.

The identification and description of the contribution to safety made by multiple barriers was a requirement of the regulations. A barrier was defined as any material, structure, or feature that prevents or substantially reduces the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment, or prevents the release or substantially reduces the release rate of radionuclides from the waste. The repository system was considered in terms of the Engineered Barrier System (EBS) an Upper Natural Barrier and a Lower Natural Barrier [39]. The EBS included the drip shield, waste package, waste form and other engineered components. The EBS was intended to prevent or reduce the rate of movement of water to the waste, the release rate of radionuclides from the waste and the rate of movement of radionuclides from the repository.

The Upper Natural Barrier included the geologic units from the surface to the repository horizon. The principal barrier function of the Upper Natural Barrier was to prevent or reduce the rate of movement of water from the surface to the repository, thereby reducing the amount of water that could enter the emplacement drifts.

The Lower Natural Barrier included the unsaturated and saturated volcanic tuff units below the repository and older bedrock units and alluvial deposits below the water table between Yucca Mountain and the accessible environment in the Amargosa Valley. The Lower Natural Barrier would have prevented or reduced the rate of movement of radionuclides from the repository to the accessible environment.

Uncertainties associated with each barrier's capability were described in the Licence Application. The analysis supporting barrier capability, including quantitative metrics of barrier effectiveness, were developed using the TSPA models.

5.3.3 Safety assessment approach and modelling

A series of safety assessments has been undertaken for the Yucca Mountain repository over a period of twenty years. The results of the studies of individual aspects of the repository have been used as inputs to a Total System Performance Assessment (TSPA).
Advances in geo-scientific understanding, design concepts, modelling capability, and regulatory requirements were incorporated into each version of the TSPA. Some of the key Performance Assessments are listed below.

- **TSPA-1991** [43], **TSPA-1993** [44], **TSPA-1995** [45]
  Development of a Monte Carlo methodology for the key processes identified at the site.

- **TSPA-VA (Viability Assessment) (1998)** [42]
  The TSPA-VA was used to provide information to stakeholders on the progress of the Yucca Mountain project, and identify any critical issues which would need to be addressed before a decision could be made to recommend the site for a repository.

- **Licence Application Design Selection (LADS) (1999)** [46]
  This was used to evaluate the relative merits of design alternatives.

  The TSPA-SR provided the technical basis for the decision by the Energy Secretary to recommend the Yucca Mountain site for development as a geological disposal facility.

  The TSPA-LA supports the DOE's Licence Application for authorisation to construct a repository at Yucca Mountain. It provides probabilistic estimates of long-term performance taking into account uncertainties in the future occurrence of disruptive events. The consequences of disruptive events are weighted by their probability of occurrence in any given year, and the probability that the event had already occurred in previous years.

The steps in the Performance Assessment used for the Licence Application [39] were listed as:

1. System description and definition of multiple barriers
2. Screening of FEPs and development of scenario classes (i.e. sets of similar scenarios)
3. Development of models and abstractions, along with their scientific basis, for logical groupings of FEPs within scenario classes
4. Evaluation of uncertainty in model inputs
5. Construction of integrated TSPA model using all retained FEPs and calculations for the scenario classes to identify appropriate models for each scenario class (‘modelling cases’)  

The FEP identification process involved several different methods, which included: using Version 1.0 of the NEA International FEP Database; brainstorming by subject matter experts during technical workshops and reviews; Top-Level-Down elicitation to develop general event-tree logic diagrams for nominal flow, tectonic processes, igneous activity and EBS degradation modes. Hybrid procedures were used in the refinement and reclassification of the NEA FEPs to make them relevant to the project [39].

A classification scheme for FEPs, specific to Yucca Mountain, was developed based on a combination of location, fields of effect, radionuclide mobilisation phenomena, and causative factors. The list of FEPs identified and classified was analysed to determine which FEPs should be included in the performance assessment analyses, and which should be excluded. A FEP was included or excluded based on any one or more FEP screening criteria.
The process of FEP screening was influenced by NRC regulations (10 CFR 197.36 as amended October 2008), which state that the performance assessments should not include consideration of very unlikely FEPs. This was defined as those that were estimated to have less than one chance in 100,000,000 per year of occurring.

Some FEPs were excluded by regulations that limit the scope of analysis to specific characteristics, concepts, and definitions. The regulatory requirements most commonly used for screening FEPs include the characteristics of the reference biosphere, geologic setting, and the Reasonably Maximally Exposed Individual.

Regulations required the inclusion of certain FEPs in performance assessments for the period between 10,000 and 1,000,000 years post-closure. These included the effects of waste package corrosion, seismic and igneous event scenarios, and climate change.

The next step in the analysis of FEPs was their aggregation into scenario classes. The objective of scenario class development for the TSPA was to define a limited set of scenario classes that could be analysed quantitatively while still maintaining comprehensive coverage of the range of possible future states of the repository system. Four scenario classes were defined for analysis, leading to seven modelling classes:

**Nominal scenario class**

*Nominal modelling case*

In this scenario, the first waste package failure (by stress corrosion cracking) would occur at approximately 30,000 years, and the drip shields would begin to fail by general corrosion at approximately 260,000 years.

**Early failure scenario class**

*Waste package modelling case*

A small number of waste packages were assumed to fail at the time of repository closure. Intact drip shields overlying early failed waste packages were assumed to degrade by general corrosion after repository closure.

*Drip shield modelling case*

A small number of defective drip shields were assumed to fail at the time of repository closure, and waste packages underlying any failed drip shields and exposed to seepage were conservatively considered as failed.

**Seismic scenario class**

*Ground motion modelling case*

The Seismic Ground Motion Modelling Case considered the possible failures of drip shields and waste packages due to mechanical damage associated with seismic vibratory ground motion, including: accumulation of rock-fall on drip shields; collapse of the drip shield framework; accumulated stress leading to stress corrosion cracking of waste packages; and rupture or puncture of waste packages.

*Fault displacement modelling case*

The Seismic Fault Displacement Modelling Case includes disruption of waste packages and drip shields by the displacement of faults.

**Igneous scenario class**

*Intrusion modelling case*

This modelling case represented a magma dyke that intrudes into the repository causing subsequent release of radionuclides to the groundwater. After a magmatic dyke intersects the repository, radionuclide release and transport away from the
repository would be similar to the Nominal Modelling Case. All the drip shields and waste packages were assumed to have been damaged.

**Eruption modelling case**

This modelling case represented a hypothetical volcanic eruption from a volcanic conduit that passes through the repository. Waste packages in the direct path of the conduit were assumed to be destroyed, and the waste in those packages entrained in the eruption. Contaminated volcanic ash would be erupted, transported in the atmosphere and deposited at the land surface.

For comparison against individual protection standards, the total mean annual dose is defined as the expected value of dose to the Reasonably Maximally Exposed Individual, where the contributions to dose from the modelling classes are aggregated.

As specified by NRC regulations, for the purposes of the performance assessment, the Reasonably Maximally Exposed Individual was assumed to be located 18 km south of the repository in the Amargosa Valley where the depth to groundwater would allow some agriculture. As prescribed by regulation, the present day characteristics of the Amargosa Valley area were used to describe the reference biosphere and the pathways by which radionuclides released into the environment would reach the Reasonably Maximally Exposed Individual.

Colloid facilitated radionuclide transport was considered in the performance assessment. Colloids were considered to arise from wasteform degradation, degradation of the waste packages and invert (see Figure 5-2) and to be present in natural groundwater. Besides reversible sorption, irreversible (kinetic) sorption of plutonium and americium onto iron oxyhydroxide colloids and waste form colloids derived from HLW glass and spent fuel was considered.

A stylised analysis of possible human intrusion into the repository was considered using the assumptions that while drilling a borehole the drilling apparatus directly intersects a degraded waste package and continues into the saturated zone underlying the repository. The human intrusion was assumed to cause release of waste from a penetrated waste package. Estimates of waste package degradation suggested that, using current technology, a degraded waste package could not be penetrated by drilling before about 200,000 years post-closure, as the drill bit was assumed not to penetrate the drip shield. Consequently the analysis considered the effects of a drilling intrusion at 200,000 years.

Natural analogue results were used in the validation process of models supporting the TSPA-LA. The natural analogues relevant to the Yucca Mountain repository were discussed in a Natural Analogue Synthesis Report [49] and included natural analogues of materials intended for use in the Yucca Mountain repository as well as the results of some investigations of analogues for geologic processes. The information from natural analogues contributed to the understanding of drift stability, degradation of the waste forms and elements of the EBS, seepage, unsaturated zone flow and transport, coupled processes, saturated zone transport, the biosphere, and disruptive events such as volcanism and seismic events.

The potential for climate change was represented by a range of climate states, including present-day, glacial and monsoon states [39]. Changes to the water infiltration rate through the repository were represented up to 10,000-years post-closure. An instantaneous change from one steady-state flow field to another was considered a reasonable approximation, given the uncertainties and the inability to observe climate changes directly. To avoid the consideration of climate changes beyond 10,000 years, the NRC prescribed the use of a long-term average infiltration rate for the longer timeframe.

The performance assessment process for the licence application used a hierarchical modelling approach [39, 50]. The foundation of the process was an assimilation of the
information from site characterisation and engineering design. The second stage of the PA process consisted of the development and testing of process models that included the retained FEPs and consideration of their outcomes regarding repository performance. The third stage of the PA process involved the development of abstracted models. These abstractions were progressive simplifications of the detailed models of physical and chemical processes to more efficient numerical models. Abstractions consisted of statistical or mathematical abstractions, including look-up tables, equations representing response surfaces, probability distributions, linear transfer functions, or reductions of model dimensionality. The top level of the PA process consisted of the integrated total system model. The total system model was a numerical model that was used to simulate the integrated behaviour of the entire Yucca Mountain repository system.

The TSPA-LA model analyses were probabilistic in order to capture the full range of potential outcomes. Uncertainty in the TSPA-LA Model was characterised as either epistemic or aleatory uncertainty.

- Epistemic uncertainty, also referred to as ‘reducible’ uncertainty, concerns the state of uncertainty in knowledge about a parameter value due to limited data or alternative interpretations of the available data. Epistemic uncertainty can be reduced, in principle, using the results of experimental testing and additional data collection.

- Aleatory uncertainty, also referred to as ‘irreducible’ uncertainty, concerns whether or not there is a chance occurrence of a FEP. No amount of exploratory work will allow determination of whether or not a chance event will or will not occur at any given time, but determining a range of likelihoods of occurrence for a given timeframe is generally supportable through use of various formalised means for combining scientific insights from experts in the field.

Epistemic uncertainty was incorporated in the TSPA through Latin hypercube sampling of cumulative distribution functions and Monte Carlo simulation with multiple realisations. There were approximately 400 uncertain epistemic parameters in TSPA-LA.

The stability of the mean and median annual dose was demonstrated for each modelling case by means of replicated sampling. Three sets of calculations were performed for three independent samples of epistemic uncertainty. Each set of calculations resulted in estimates of the mean and median annual dose, along with the 95th and 5th percentiles of expected annual dose. Comparison of these statistics demonstrated that the sample size was sufficient to obtain statistically stable estimates of mean and median annual dose.

TSPA-LA model results were compared to results calculated using a simplified TSPA analysis of the Yucca Mountain repository that was developed to corroborate the TSPA-LA model using a higher level abstraction than that used in the TSPA-LA model. The simplification involved removing detail from the TSPA-LA model to capture spatial and temporal variability.

A Performance Margin Analysis (PMA) was used to quantitatively evaluate conservatisms in the TSPA-LA model to quantify the extent to which they overestimated the mean annual dose. When compared to the total mean annual dose calculated with the TSPA-LA model, the lower total mean annual dose calculated by the PMA demonstrated that the conservatisms incorporated in the TSPA-LA Model did not introduce risk dilution.

A sensitivity analysis was conducted in order to identify the dominant sources of uncertainty in total expected dose to the Reasonably Maximally Exposed Individual. The primary sensitivity analysis procedures in use involved the determination and presentation of partial rank correlation coefficients, stepwise rank regression analyses, and scatter-plots. For different time frames in the analysis, different parameters emerged as important to the overall uncertainty in the results. Of particular importance to the calculated risk were:
• The probability of an igneous event, expressed as the annual frequency of an intersection of the repository by a volcanic dyke. Uncertainty in this parameter arose from uncertainty about igneous activity that may affect the repository.

• The residual stress threshold for the Alloy 22 waste package outer barrier, expressed as a percentage of the yield strength. If the residual stress in the waste package outer barrier exceeded this threshold value, stress corrosion cracks could form, which could allow radionuclides to migrate from the waste package.

• The temperature dependence of the Alloy 22 waste package outer barrier general corrosion rate. This parameter directly influences the short-term and long-term general corrosion rates of the Alloy 22. Larger values resulted in earlier, higher general corrosion rates during the thermal period and lower long-term corrosion rates when the repository temperatures were near the ambient temperature.

5.4 Safety case presentation

The Licence Application is a submission to address specific regulatory acceptance criteria allowing the construction of a repository. It is a detailed set of documents, focussed on systematically demonstrating that each of the regulatory criteria is satisfied. The Licence Application includes two volumes: General Information (GI) and the Safety Analysis Report (SAR).

The General Information volume includes a general description of the repository, a proposed schedule for construction, description of the receipt and emplacement of waste, a Physical Protection Plan, details of the Material Control and Accounting Programme and a description of the site characterisation. The Safety Analysis Report is subdivided into repository safety before permanent closure, repository safety after permanent closure, research and development programme to resolve safety questions, performance confirmation programme and management systems. The underpinning technical and scientific documents used to support the safety case are generally publicly available.

In addition, the TSPA model and analysis for the Licence Application is described in [39]. This report is in four volumes, totalling approximately 4,000 pages.

5.5 Conclusions and confidence

The TSPA-LA model results indicate that the highest projected total mean and total median annual doses to the Reasonably Maximally Exposed Individual are less than the individual protection standard required by NRC regulations during the 10,000 years after repository closure. In addition, the highest projected dose values up to 1 million years post-closure are more than two orders of magnitude below the individual protection limit [39]. Figure 5-3 shows the contribution to the total mean annual dose histories for the drip shield, waste package, igneous intrusion, volcanic eruption, seismic ground motion, and seismic fault displacement modelling cases. The seismic ground motion and igneous intrusion modelling cases provide the largest contributions to the total mean annual dose for the post-closure period from 10,000 to one-million years.
The scenarios that most affect repository performance are seismic ground motion, igneous intrusions, and waste package failure due to general corrosion. The key radionuclides contributing to the total mean annual dose were found to be technetium-99 and plutonium-239 up to 20,000 years post-closure; and iodine-129, radium-226, neptunium-237 and plutonium-242 after this. The contribution of plutonium-239 in the earlier time period is due to transport in the dissolved phase and reversible sorption to colloids [39].

An international peer review of the Total System Performance Assessment supporting the site recommendation process (TSPA-SR) was undertaken in 2000 [41]. This review noted that the approach of building on an iterative series of performance assessments conformed to international best practice. It was noted that the structured abstraction process linking process-level models to assessment models was at the forefront of international developments.

The international peer review noted that regulatory requirements set down and proposed for the Yucca Mountain Project were somewhat more prescriptive than in many other countries, both in specifying compliance requirements and in directing how these must be met. It was suggested that the way the regulations were formulated contributed to a tendency of the TSPA-SR to focus more on demonstrating numerical compliance with quantitative criteria than on demonstrating an understanding of repository performance. Some issues raised in the international peer review of the TSPA-SR were discussed in the TSPA-LA. These include:

- The TSPA-LA has a detailed description of the development of a systematic methodology for identifying and treating all types of uncertainty (as this was identified as a weakness in the international peer review).
- The potential for risk dilution is considered in more detail in the TSPA-LA.
In the TSPA-SR, chlorine-36 and caesium-135 were screened out of the list of radionuclides considered in the performance assessment. These radionuclides were noted by the international peer review as being significant in other performance assessments and were included in the TSPA-LA.

In parallel with the work undertaken by the DOE and its contractors, the Electric Power Research Institute (EPRI) carried out its own performance assessments of the potential Yucca Mountain repository, using the DOE’s work for most of the basic input data, but developing an independent analysis. The EPRI studies have been reported in a series of reports, for example [51].

In 2010, the NRC began orderly closure of its Yucca Mountain activities. As part of the orderly closure, NRC staff prepared a series of four Technical Evaluation Reports (TERs) as knowledge management documents, however these do not include conclusions as to whether or not the Yucca Mountain Licence Application would have satisfied the NRC’s regulations [52].
6 LLWR Environmental Safety Case

6.1 Background and context

The Low Level Waste Repository (LLWR) has been the UK's principal facility for the disposal of solid Low Level Waste (LLW) since it opened in 1959. It is regulated by the Environment Agency (EA) under the Environmental Permitting Regulations (2010 England and Wales) [53]. The facility is currently managed by LLW Repository Ltd on behalf of the NDA.

LLW Repository Ltd submitted a Post Closure Safety Case (PCSC) and an Operational Environmental Safety Case (OESC), which together formed their Environmental Safety Case (ESC), to the Environment Agency (EA) in 2002. The EA reviewed this submission and concluded that it did make the case that the facility was currently safe, but it did not make an adequate case for continued disposal. As a result of this submission, LLW Repository Ltd was issued with a Radioactive Substances Act (RSA) 93 authorisation (now superseded by an Environmental Permit) in May 2006 that only allowed disposals to continue in Vault 8, which was expected to reach capacity in 2009\textsuperscript{15}. This current permit was originally issued in May 2006, with a variation for the new Environmental Permitting Regulations in December 2010. There was no authorisation for disposal beyond Vault 8, however planning permission was granted for storage in Vault 9 when constructed.

The 2011 ESC (the subject of this study) [10] was prepared and submitted to the EA in fulfilment of a specific requirement of the current Environmental Permit. This requirement is stated in Schedule 9 of the Permit;

“For an ESC to be submitted no later than 1st May 2011. The EA will review the updated ESC and following this they will consider and consult on the authorisation for further disposals in 2012/13.”

The LLWR is located on the West Cumbrian coastal plain, close to the village of Drigg and approximately 5km southeast of Sellafield. Figure 6-1, shows the location of the LLWR and its proximity to the coast. This location is a key component of the 2011 ESC as the site is expected eventually to be eroded into the sea. Waste was initially disposed in seven trenches where loose waste was ‘tumble tipped’ in bags or drums. The trenches were used for disposal until 1995. The trenches are currently covered by an interim cap of soil, containing a plastic membrane to minimise the infiltration of water into the wastes.

\textsuperscript{15} To date there is still a small amount of space in Vault 8 for disposals.
From the late 1980s onwards a concrete disposal vault was constructed (Vault 8, completed in 1988), allowing the disposal of wastes in containers. Waste was first received in Vault 8 in 1988. Waste for Vault 9 had already started to be placed in a prepared area in 2009, prior to the vault’s completion in 2010. Figure 6-2 shows the different disposal vaults of the LLWR.
Figure 6-2 Disposal vaults in the LLWR

6.2 System description

6.2.1 Disposal inventory and waste types

The LLWR receives waste from a range of producers, including nuclear power stations, facilities that manufacture and reprocess nuclear reactor fuel, defence establishments, general industry, radioisotope manufacturing sites, hospitals, universities and from the clean-up of contaminated land and buildings. The wastes are normally grouted in steel containers before emplacement. The wastes are, with some exceptions, either supercompacted pucks or non-compactable materials such as soil, concrete and bulk metallic items. Certain materials are required to be absorbed or encapsulated prior to disposal. On occasions, very large items of waste that it is not practicable to size reduce are disposed directly in the vault and grouted in place. The largest consignor of waste to the LLWR is the nearby Sellafield nuclear site. Currently about 30% of the waste received originates from Sellafield and waste from other sites is compacted at Sellafield before being sent to the LLWR.

The UK Radioactive Waste Inventory (RWI) [54] lists all LLW (including existing waste and that planned to arise in the future) that it is expected will need to be managed in the future. The volume of this waste in its raw state is approximately three million cubic metres. However, some of the LLW in the UK RWI is very low level waste (vLLW) and can be disposed of in landfill-type facilities and therefore does not need to be disposed of in the engineered vaults at the LLWR.

6.2.2 Geological environment

The geological structure in the region of the LLWR consists of Quaternary age deposits (up to 2.6 million years old) overlying older bedrock. Quaternary deposits at the LLWR site are a result of complex glacial processes, which were responsible for the deposition of a sequence of deposits of clay, sands and gravels up to 60m thick. The Quaternary deposits overly Triassic Ormskirk Sandstone (around 240 million years old) in the vicinity of the LLWR site.
6.2.3 Disposal concept

The disposal concept for the LLWR has evolved over its years of operation, as described previously. The site began accepting wastes in 1959 and these wastes were 'tumble tipped' in bags or drums into the trenches. The trenches were filled sequentially and covered with soil at the end of each day and periodically. From 1987 onwards, disposal operations were upgraded and remedial work was also carried out on the trenches and they were covered with an interim cap of soil, containing a plastic membrane to minimise the infiltration of water into the wastes (thus minimising the potential for activity to be mobilised).

From the late 1980s onwards, disposal operations were upgraded to modern standards and a vault system was installed (vaults 8 and 9). These vaults are made of concrete and allow the disposals of waste in steel containers that are filled with cement grout. Water infiltrating through the trenches and rain water run-off from these vaults are collected, sampled to check they are safe, and then discharged down a pipeline to the sea [55]. LLW Repository Ltd proposes to construct a final cap over the trenches and the vaults once all the disposals have been made. The cap is proposed to be three metres thick, excluding profiling material, and consist of a number of layers designed to deter intrusion by people, animals and plants, limit infiltration of rainwater into the wastes, and disperse any gas that accumulates during the lifetime of the facility. LLWR Repository Ltd also proposes to build a cut-off wall to limit the flow of groundwater into the wastes and help prevent contaminated water flowing out near the surface. This will extend an existing cut-off wall on the north and east sides of the trenches. The cut-off wall will be made of a mixture of clay, cement and blast furnace slag.

Figure 6-3 shows the combination of existing engineered features and those that are planned as part of the closure of the site.

**Figure 6-3 Key engineering features of the LLWR**
6.3 Safety case approach

6.3.1 Regulatory requirements

The Environment Agency has produced guidance on the requirements for authorisation (GRA) of a near-surface disposal facility [4]. The high-level principles and requirements are exactly the same as those for geological disposal [5]. The near-surface GRA [4] is a sister document to the GRA [5] for geological disposal. It states that an environmental safety case is a set of claims concerning the environmental safety of disposals of solid radioactive waste, substantiated by a structured collection of arguments and evidence. The environmental safety strategy should present a top level description of the fundamental approach taken to demonstrate the environmental safety of the disposal system. It should include a clear outline of the key environmental safety arguments and say how the major lines of reasoning and underpinning evidence support these arguments.

The near surface GRA also states a risk guidance level which is a level of radiological risk from a disposal facility which provides a numerical standard for assessing the environmental safety of the facility after the period of authorisation. The risk guidance level in the near surface GRA is stated as $10^{-6}$ to an individual per year, which is the same as the risk guidance level in the GRA for geological disposal.

However, in the near-surface GRA the risk guidance level does not apply to human intrusion after the period of authorisation (when the facility is closed). There is a separate requirement (requirement R7) stated in the GRA for the risk guidance level associated with human intrusion after the period of authorisation. This is not a requirement in the GRA for geological disposal. The near-surface GRA requirement states that [4];

"The developer/operator of a near-surface disposal facility should assess the potential consequences of human intrusion into the facility after the period of authorisation on the basis that it is likely to occur. The developer/operator should, however, consider and implement any practical measures that might reduce the chance of its happening. The assessed effective dose to any person during and after the assumed intrusion should not exceed a dose guidance level in the range of around 3 mSv/year to around 20 mSv/year. Values towards the lower end of this range are applicable to assessed exposures continuing over a period of years (prolonged exposures), while values towards the upper end of the range are applicable to assessed exposures that are only short term (transitory exposures)."

The structure of the ESC is discussed later in Section 6.4.

6.3.2 Safety arguments

Due to the near-surface nature of the facility, there is more emphasis on the engineered barriers and relatively little reliance is placed on the natural barriers.

The ESC for the LLWR is set out as a set of safety arguments under four main headings:

1. Management and dialogue
2. System characterisation and understanding
3. Optimisation and Site Development Plan (SDP)
4. Assessment

The main report (see Section 6.4 for details of other reports) of the ESC sets out the safety arguments under the four main headings above and labelled as such. The report also refers back to the near-surface GRA [4] for each of these labelled safety arguments. The key safety arguments presented under the topic of management and dialogue include the management system and safety culture, organisation of the ESC project, dialogue with
both the environmental regulator and their stakeholders, and finally peer review. The key safety arguments under characterisation and understanding include the wastes, the near field, the geology and the hydrogeology, the coastal and the surface environment and monitoring.

Under the topic of optimisation LLWR Ltd considers remediation of past waste disposals, optimisation of future disposals, site engineering, waste emplacement strategies, management of run-off and leachate and management during closure. The assessment topics include both qualitative (understanding of safety functions) and quantitative considerations (analysis and modelling). There is also consideration of uncertainty management, radiation doses during the Period of Authorisation and risks in the long term, non-radiological impacts, impacts on non-human biota, Waste Acceptance Criteria and radiological capacity, and extended capacity.

To some extent this structure of topics mirrors the topics within the GRA [4].

6.3.3 Safety case approach and modelling

In the safety case documentation the high level safety arguments and qualitative lines of reasoning are discussed first and then underpinned with quantitative assessments. The objective of the LTRA is to present a detailed quantitative assessment of the radiological performance of the LLWR. However, it states that it is useful to first set out a qualitative description of the functions and the expected evolution of the LLWR and its environment based on the understanding developed during current and previous assessments.

A brief example of the qualitative approach used is the discussion on migration in groundwater. LLW Repository Ltd provides a discussion on how the cap and cut-off walls ensure any contaminant movement from the trenches and vaults is downwards in unsaturated conditions to the water table. Contaminants moving down to, and in, groundwater flowing in the drift and sandstones beneath the site may be sorbed, especially to clay minerals, but the primary safety function for those mobile contaminants that are of greatest concern is the dilution provided in groundwater flowing beneath the site.

The quantitative assessment of the groundwater pathway is considered by LLW Repository Ltd to provide an appropriate representation of the LLWR facility and the movement of radionuclides through the various components of the facility.

As a site already exists for the LLWR, calculations can be compared with measured concentrations e.g. in the trench leachate for the groundwater pathway.

In support of the ESC and its supporting Long Term Radiological Assessment (LTRA) [56], LLW Repository Ltd has developed a thorough, scientifically based understanding of the evolution and performance of the existing disposal facility and its planned development in terms of the:

- potential radiological and non-radiological hazards;
- safety functions, e.g. isolation of the waste, containment of contaminants and attenuation of releases; and
- the FEPs that provide, promote or reduce those functions.

The well-established approach to long-term assessment of radioactive waste disposal facilities is followed in the 2011 LLWR ESC, as described by the NEA [57] and by the IAEA [58, 59]. This includes activities that are interlinked:

1. Acquisition of information and data, and development of understanding of the disposal system and the uncertainties present, including those related to waste form, site characteristics, engineered structures and their evolution;
2. Identification of the pathways potentially leading to the transfer of radionuclides from the repository to humans and the environment, and refinement/definition of those pathways taking account of system-specific understanding and data;

3. Definition of the broad scenarios for the long-term evolution of the disposal system and its environment, and identification of the cases that should be addressed taking account of the model capacity;

4. Developing and testing of conceptual and mathematical models of the behaviour of the disposal system and its components, and implementation in quality assured numerical models and codes;

5. Application of all of the above in a structured and comprehensive analysis to yield the required assessment of the relevant uncertainties; and

6. Evaluation of the robustness of the analysis and its results, and identification of key uncertainties that have bearing on the interpretation of the results and conclusions that should be drawn from the assessment.

Modelling hierarchy
The 2011 ESC describes different types of models in a hierarchical structure. LLW Repository Ltd states that an assessment model is the assembly of FEPs and interactions that are treated in a given case or set of assessment cases. The conceptual model is the descriptive presentation of the model and the numerical model is the model as implemented by given equations or software.

LLW Repository Ltd states that the first requirement for a conceptual model is that it includes the relevant FEPs and incorporates a correct understanding of the important interactions at an appropriate scale. This is a matter of sufficient information, scientific understanding and judgement. Development of conceptual models may be iterative and generally includes stages of discussion and review. Scoping calculations and results from model testing may also reveal where attention should be focused to improve a model, and focus it to the assessment in hand.

Key assessment software codes used in the 2011 LTRA are GoldSim, which has been used within the assessments of the groundwater, gas, coastal erosion and human intrusion pathways, CONNECTFLOW\textsuperscript{16} and GRM\textsuperscript{17}, which have been used for underpinning detailed modelling of groundwater flow and near-field biogeochemistry, respectively.

In the 2011 ESC, LLW Repository Ltd presented a probabilistic risk analysis for the groundwater pathway for radionuclides released via a water abstraction well. This is because the groundwater well is the limiting biosphere release path for the groundwater pathway, the pathway is amenable to probabilistic analysis, and the probability of a well is intrinsic to the analysis. The results are presented as annual risks for comparison with the GRA annual risk guidance level.

The probabilistic risk analysis was preceded by a suite of deterministic calculations to separately investigate the impacts of alternative near-field models, assumed forward inventory, different biosphere release paths, and alternative data assumptions e.g. for solubility limits and sorption parameters. Results from deterministic calculations were

\textsuperscript{16} www.connectflow.com

\textsuperscript{17} The Generic Repository Model (GRM) is a reactive-transport model, coupling biogeochemical processes with advective groundwater transport. It was developed by British Nuclear Fuels Limited (BNFL), who previously has responsibility for the LLWR, for studies of the biogeochemical evolution of the LLWR trenches and vaults.
presented as conditional risks, for consistency with the results of the probabilistic risk analysis that follows.

Results were calculated for defined deterministic cases for the gas and the coastal erosion pathways. This is because in these cases there are uncertainties that are not readily quantified at the present state of knowledge over:

- forward inventory waste characteristics, and carbon-14 release and gas generation processes, for the gas pathway; and
- future global CO₂ emissions and sea-level rise and local erosion under future sea-level rise conditions, for coastal erosion.

Results for these pathways are presented primarily as annual doses to emphasise that the cases relate to conditions or evolutions, the probabilities of which are not quantified. However, the cases that are considered most representative or ‘central’ and therefore most appropriate to compare with an annual dose corresponding to the risk guidance level (20 μSv) are identified.

In the human intrusion pathway calculations LLW Repository Ltd calculates doses for comparison with the dose guidance range of around 3 mSv y⁻¹ to around 20 mSv y⁻¹, as specified in the GRA [4]. Dose is calculated from the event for short-term exposures such as received by people directly engaged in the event, and annual doses to people exposed over longer times to environmental contamination arising from the event.

**Scenarios and calculation cases**

LLW Repository Ltd defines scenarios as:

‘Scenarios are the broad alternative future events, circumstances, conditions or their evolution, that are characteristically different and provide a framework for analysis that is useful in providing an understanding of the environmental safety of the disposal system.’

The 2011 ESC suggests that there may be relatively few scenarios, however there may be a large number of calculation cases required to adequately explore all aspects and uncertainties within a scenario. A calculation case is defined as;

‘A calculation case is a specified combination of events, circumstances, conditions or their evolution, including specification of model boundary conditions and data, which represents a particular realisation of the disposal system, its evolution and radionuclide or contaminant release path of concern.’

Alternative calculation cases are defined to investigate the effect of a particular assumption or uncertainty, or combinations of assumptions, e.g. near-field evolution, geosphere pathway, characteristics of potentially exposed groups.

An example of a calculation case is the reference case for the groundwater pathway deterministic calculation case, which is defined as that combining:

- the expected natural evolution scenario;
- the reference inventory; and
- the near-field model ‘C’ (which considers radionuclides in the saturated zone immediately available for dissolution and release of carbon-14 determined from the Generic Repository Model (GRM)).

LLW Repository Limited has a different approach to defining the assessment timescales as the site is located near to the sea in an area of coastal erosion. The ESC draws upon both qualitative and quantitative evidence including modelling studies, and concludes that the coastal site for the LLWR facility is likely to begin to be eroded on a timescale of a few hundred to a few thousand years, with consequent disruption of the repository; with erosion
of the vaults and trenches being complete within one to a few thousands of years. Based on this, the ESC includes two scenarios: the first is the 'Expected Natural Evolution' scenario which looks at releases on the completion of closure engineering, at the end of management control, 100 years after present and 500 years after present. The second scenario is a 'Delayed Coastal Erosion' scenario where impacts are considered for up to 10,000 years after present on the assumption that the site will still not be completely eroded after 10,000 years.

Doses are also calculated for other timescales and the rationale for these timescales in presented in Table 1.

Table 6-1 Timescales for examining contributions to dose

<table>
<thead>
<tr>
<th>Year</th>
<th>Approximate Time</th>
<th>Rationale</th>
</tr>
</thead>
<tbody>
<tr>
<td>2180</td>
<td>170</td>
<td>About 100 years after last disposals; earliest envisaged end of Period of Authorisation</td>
</tr>
<tr>
<td>2310</td>
<td>300 y</td>
<td>Earliest assessed time for contact of site by coastal erosion</td>
</tr>
<tr>
<td>3000</td>
<td>1000 y</td>
<td>Reference time for contact of site by coastal erosion</td>
</tr>
<tr>
<td>5000</td>
<td>3000 y</td>
<td>Latest assessed time for contact of site by coastal erosion</td>
</tr>
<tr>
<td>12000</td>
<td>10,000 y</td>
<td>If the site is not eroded as expected</td>
</tr>
</tbody>
</table>

Improvements in 2011 ESC quantitative modelling

The 2011 LLWR ESC provides the following improvements relative to the 2008 assessment\(^\text{18}\) with respect to the quantitative assessment of the groundwater pathway:

- A revised approach to the assessment of risks following the sinking of a well is employed. In this approach the probability that a well is present in the area of contaminated groundwater plume is taken into account.

- The kinetic release of carbon-14 from graphite and steel is taken into account, since a significant proportion of the disposed carbon-14 inventory is present within these wasteforms. This is done by making use of detailed calculations undertaken for carbon-14 within the GRM.

- A much more detailed representation of the water flows through the repository has been adopted. A compartment flow model has been developed that calculates the key water flows through the repository, and the temporal evolution of the flows, as a function of the evolution of the parameters that determine the flows.

- The saturated and unsaturated regions of the trenches and vaults are each represented separately. An approach has been developed to represent the release of wastes from the unsaturated regions into solution, and also to accommodate the transfer of radionuclides between the saturated and unsaturated regions as the water heights in the trenches and vaults change.

It is accepted that uncertainty is present about the release and transport of radionuclides from the near field. This is addressed by undertaking a probabilistic assessment calculation case, as well as a number of deterministic cases.

\(^{18}\) In 2008 further screening calculations were undertaken for the groundwater pathway to take account of the updated inventory at that time (the 2004 UK Radioactive Waste Inventory), and using a more cautious groundwater model that caused several shorter-lived radionuclides to be included.
In the probabilistic approach, probability density functions are assigned to various flow and transport parameters. The probability density functions for the different parameters were derived mostly by expert elicitation. Table 2 provides a description of the parameters and their derivation. The results of the probabilistic calculation case then provide an indication of the range of risks that could arise under parametric uncertainty.

Table 6-2 Parameter types assigned PDFs and their derivation [56]

<table>
<thead>
<tr>
<th>Parameter Type</th>
<th>Method of deriving PDFs</th>
</tr>
</thead>
</table>
| Radionuclide inventories             | C-14, Cl-36 and I-129 in the vaults – defined on the basis of examination of the waste streams that give rise to these radionuclides in the forward inventory.  
All other radionuclides – log-triangular PDFs with Case A inventory as the peak, and minimum and maximum values once order of magnitude on either side. |
| Near field sorption coefficients     | Tc and U on grout and soil – PDFs defined by expert elicitation (log-triangular PDFs derived).  
All other radionuclides – log-triangular PDFs with best-estimate values as the peak, and minimum and maximum values one order of magnitude on either side. |
| B2 and B3 sorption coefficients      | Log-triangular PDFs in which the maximum and minimum are one order of magnitude either side of the best-estimate values used in the deterministic calculations. |
| Unsaturated release parameters       | PDF according to elicited values (more information in [56]).                                                                                          |
| Well probability (density)           | PDF of annual probability of a well between the LLWR and the coast defined by expert elicitation as a log-triangular distribution (0.03, 0.1, 3) per km². |
| B3 water flow rates                  | PDF specified as a ‘flow factor’ that multiplies the flow model rates. Based on the range of hydrogeological parameters that provided a good match to observed data in the 2010 Hydrogeological Model the flow factor PDF is log-triangular (0.333, 1, 2). |
| B3 flow Peclet number                | PDF specified by judgement as log-triangular (5, 10, 20). This choice retains the characteristic of B3 as a unit in which radionuclide transport is advection-dominated. |
| Infiltration through the cap         | PDF according to elicited values, (see [56]).                                                                                                        |
| Hydraulic conductivities of the     | PDF according to elicited values, (see [56]) and matching to 2010 Hydrogeological Model.                                                              |
| components of the LLWR              |                                                                                                                                                      |

**Treatment of uncertainty**

LLW Repository Ltd states the following goal for treating uncertainty;

*"Our goal within the Long Term Radiological Assessment (LTRA) is to present qualitative understanding of the key uncertainties, and qualitative illustrations of the effects of*
quantifiable uncertainties, such as to inform both our own and the Environment Agency’s judgement as to whether our assessment results are consistent with the risk and dose guidance levels. In addition we identify priorities for further work by which key uncertainties might be reduced."

LLW Repository Ltd assessed alternative inventory cases that take account of uncertainty over future nuclear build, waste volume arisings, and use of alternative disposal routes for some categories of LLW.

For the assessment in support of the 2011 ESC (e.g. the LTRA) the Expected Natural Evolution scenario that encompasses the broad expectation for the evolution of the disposed wastes, engineered barriers and natural environment was defined. This is based on the characterisation and understanding of the wastes, engineered barriers, hydrogeology of the site and environmental processes and their impacts. Uncertainties within this broad scenario are treated by alternative assessment cases, appropriate to each of the pathways.

A Delayed Coastal Erosion scenario, extending to 10,000 years was also assessed, to explore the proposition that the site will not be eroded within this time frame. Human intrusion events are a special set of cases that are assessed by taking account of the broadly expected evolution of the site, and for the scenario in which the site exists over a longer time.

In the LLWR 2011 ESC uncertainties have arisen from:

- Uncertainties concerning past and future waste disposal operations, e.g. the radiological inventory, future engineering and waste management choices;
- Uncertainties in the characteristics of the waste disposal facility and its environment, e.g. due to measurement uncertainty, uncertainty in interpretation, applicability of literature values, and variability of parameters in time and space;
- Uncertainties about the long-term evolution of the waste disposal facility and its environment and about the future events that may have an impact on the disposal facility and its environment; and,
- Uncertainties concerning future human actions and behaviour.

LLW Repository Ltd states that some uncertainties will remain unresolved, but that this does not prevent an assessment of the safety of the LLWR taking account of the uncertainties.

The classification of uncertainties adopted is conventional in radioactive waste disposal assessment, focusing on the mode of treatment in the safety assessment, thus:

- Scenario uncertainty – definition of a scenario, or scenarios, sufficiently broad to represent the possible evolutions of the disposal facility and its environment;
- Model uncertainty – models, including alternative model assumptions that adequately represent the FEPs that are important to radionuclide release, transport and exposure to humans; and
- Parameter uncertainty – variation of model parameter values within their realistic or possible ranges.

Within the LTRA these types of uncertainty are addressed as follows:

**Scenario uncertainty:**

- Identification of the pathways leading to the transfer of radionuclides from the repository to humans and the environment and characterisation of the uncertainties for each pathway; and
• Identification of a broadly expected scenario for the future evolution of the natural environment and a less likely alternative scenario, and identification of assessment cases within each.

Model uncertainty:

• Development of sound conceptual models of the pathways, including considering alternative assumptions, and implementation of the models in quality assured codes;
• Use of these models to analyse and assess each of the pathways including taking account of alternative assumptions and parameter uncertainty; and,
• Creation of a FEP and uncertainty tracking system (within Excel), which indicates how uncertainties have been addressed within the models and cases analysed in the LTRA.

Parameter uncertainty:

• In most cases parameter values were taken from standard sources or established by recorded judgements. For key parameters that are important to the assessment of the groundwater pathway formal elicitations were undertaken [60].

In the 2011 LLWR ESC a FEP list was developed and used to trace how each FEP and its related uncertainties were treated, based on the conceptual models actually envisaged in the Long Term Radiological Assessment. The FEP and uncertainty tracking system is implemented as a Microsoft Excel™ spreadsheet including screens and macros to assist in the management of the FEPs and associated uncertainties. It includes consideration of how each FEP is treated and a means by which significant uncertainties can be identified, rated according to expert judgement on the importance to sub-system performance.

Optimised site development plan

To establish the capacity of the LLWR for disposal of LLW, radiological impacts are assessed assuming development of the facility according to an optimised Site Development Plan (SDP), accommodating currently disposed and emplaced waste, plus future arisings of LLW in the UK.

The SDP is LLW Repository Ltd’s current view of how the LLWR should be developed and provides the basis for quantitative modelling and assessments of environmental safety. The 2011 ESC considers a reference case in which an area covering the northern 40 hectares of the site is developed for the disposal of LLW (Reference Disposal Area, RDA) and a variant case in which a further adjacent 11 hectares is developed for further disposals of LLW. The SDP would allow the site to continue to operate as the primary destination for the disposal of LLW in the UK until about 2080 (for the Reference Disposal Area), depending on assumptions about future LLW waste arisings and the application of waste segregation, diversion and treatment.

6.4 Safety case presentation

LLW Repository Ltd presents three main levels of report. Figure 6-4 shows the hierarchical structure of the ESC documentation.
The high-level safety arguments are set out in the level 1 report and comprise arguments concerning the development and safety of the Site Development Plan. The level 1 report (approximately 150 pages long) focuses on the arguments in principle, referring to the more detailed and quantitative evidence presented in the level 2 reports. The main features and findings of the supporting reports are presented, demonstrating that the SDP is optimised, and that the assessed safety is consistent with the regulatory guidance over the lifetime of the facility, including after closure.

The level 2 reports present the evidence that underpins the safety arguments, including descriptions of the management framework, system understanding (including site history and description, inventory, engineering and design, near field, hydrogeology, site evolution and monitoring), optimisation, assessments and proposed conditions for acceptance of waste. There are six level 2 assessments reports (with 16 level 2 reports in total), as follows:

1. Environmental Safety During the Period of Authorisation
2. Assessment of Long-term Radiological Impacts
3. Assessment of Non-Radiological Impacts
4. Assessment of Impacts on Non-human Biota
5. Waste Acceptance
6. Assessment of an Extended Disposal Area

These reports are essentially standalone reports; however, they do refer to each other where appropriate. The level 3 or supporting documents provide details of model formulations and data used in specific topic areas.

LLW Repository Ltd decided to develop and present the ESC according to the approach to arriving at the proposal for safe, optimised development of the LLWR. LLW Repository Ltd then shows the arguments and evidence advanced to satisfy the requirements of the GRA, providing a table which lists the GRA requirements, indicating how the safety case arguments correspond to each of the requirements.
In addition there is also a non-technical summary of the ESC [55], which has been produced to help a wider range of stakeholders understand its nature, conclusions and implications.

6.5 Conclusions and confidence

At a high level the safety case is that:

- LLW Repository Ltd has worked within a sound management framework and firm safety culture, while encouraging dialogue with stakeholders.

- LLW Repository Ltd has characterised and established a sufficient understanding of the LLWR site and facility, and its evolution, relevant to its environmental safety.

- On which basis, LLW Repository Ltd carried out a comprehensive evaluation of options to arrive at an optimised SDP for the LLWR.

- LLW Repository Ltd has assessed the environmental safety of the SDP, showing that impacts are appropriately low and consistent with regulatory guidance. Using these assessments, LLW Repository Ltd has determined the radiological capacity of the facility and conditions under which waste may be safely accepted and disposed.

LLW Repository Ltd states that the ESC sets out the environmental safety arguments and supporting evidence that they believe justify the continued use of the LLWR to dispose of LLW and ideally only that requiring disposal in vaults.

The conclusions of the LTRA first set out LLWR Ltd’s position on the core deficiencies of the 2002 Post Closure Radiological Safety Assessment as identified by the Environment Agency, and then confirm that the Environment Agency’s recommendations based on the 2008 Performance Assessment Update have been addressed. The assessed radiation risks and doses and their consistency with the GRA Requirement R6 and R7 are then summarised and the key uncertainties that could most affect these assessments are discussed.
7 Lessons learned for RWMD

This chapter discusses some key aspects in the development and production of the safety cases that were judged to be of particular interest to RWMD when studying the five safety cases. The aim here is to identify and understand commonalities and differences in approach and then to consider these in the context of RWMD’s programme with a view to identifying any potential learning points for the on-going development of RWMD’s Environmental Safety Case. RWMD has published an overall strategy for the development of its ESC [61] that describes how the current generic ESC will be maintained and developed in parallel with a separate strand of work that will start to develop a site-specific ESC once a site or sites are suggested for consideration for hosting a geological disposal facility. As the safety cases studied in this report all relate to specific geological environments, the lessons learned from this study are likely to be most relevant to the development of the site-specific ESC, but there may also be some points worth considering for the development of the generic ESC, particularly concerning its structure and presentation.

7.1 Role and use of FEPs and safety functions

FEPs are a key building block of all the safety cases studied – it is essential to understand and appropriately represent those features, events and processes that are relevant to the performance of a disposal facility. The main challenge is to demonstrate confidence that all significant FEPs have been considered. This is not an easy task, but considerable confidence can be taken from the fact that over the last decade or so, as waste management organisations have been publishing their FEP lists, for example as part of the NEA FEP database [62], there have been no identifications of significant new FEPs. Although FEPs continue to be the backbone of a safety case as they form the basis of all conceptual model development, the NEA has noted [63] that several organisations are now placing more focus on identifying safety functions based on the scientific and technical knowledge base that the disposal project has accumulated. These safety functions are then used to identify performance assessment scenarios that are relevant to exploring the robustness of the safety functions. The NEA FEP database, however, continues to be a useful tool and is referred to in each of the five safety cases studied in this report. The NEA FEP database is generally referenced to demonstrate that all relevant FEPs have been addressed in a safety case.

SR-Site has perhaps led the way internationally in developing the concept and use of safety functions. The identification and processing of FEPs still remains the first step in the safety assessment approach, as it is only from a detailed understanding of the system features and processes, that all safety functions can be identified [6, p. 248]. SR-Site has well defined safety functions and has, for many of the functions, quantified safety function indicator criteria that determine whether or not each safety function is maintained. For example, the primary safety function is the integrity of the copper canister, so the thickness of the copper is a safety function indicator and its associated safety function indicator criterion is that the thickness of the copper must be greater than zero over the whole canister surface. Other safety function indicator criteria, for example, define the maximum hydraulic conductivity of the bentonite buffer ($10^{-12}$ m/s) and the minimum swelling pressure of the buffer (1 MPa). Such quantification of the required performance of safety functions is possible because of the essential role of the integrity of the copper canister in the SR-Site safety case and the performance of other system components (such as the bentonite buffer) can be defined in terms of their capacity to protect the copper canister.

In other safety cases, where the safety concept may rely on a more complex interaction of safety functions, it is less straightforward to quantify the required safety function performance. For example, in the French Dossier Argile, Andra defines expected safety functions based on a detailed consideration of relevant disposal system phenomena over
both spatial and temporal scales. The focus of the safety analysis is reasoned arguments (both quantitative and qualitative), based on scientific evidence, to support the presented phenomenological understanding of the disposal system evolution.

The two American safety assessments (for WIPP and Yucca Mountain) both start with the identification and screening of FEPs. FEPs were initially identified from internationally accepted FEP databases (the NEA FEP database or its precursors) and then screened to remove irrelevant FEPs and those that were judged to be beneficial to safety. The US regulations also enabled screening of FEPs with a sufficiently low consequence (an annual probability of occurrence of less than $10^{-8}$) or which related to activities excluded from the regulations (for example the required human intrusion scenarios are explicitly defined in the regulations so other intrusion FEPs can be ignored). This is an example where the relatively explicit regulations enabled simplification of the assessment basis. The focus of these US assessments was then to define classes of scenarios to satisfy their respective regulatory requirements and undertake detailed probabilistic analyses for comparison with the regulatory safety criteria. Safety functions did not therefore have an explicit role in defining the assessment approach in these two safety cases.

The Long Term Radiological Assessment of the LLWR ESC considered the required high-level safety functions (namely, the isolation of the wastes, containment of the contaminants and attenuation of any releases) and the FEPs that provide, promote or reduce these safety functions. The well-established approach to long-term assessment of radioactive waste disposal facilities is followed in the 2011 LLWR ESC, as described by the NEA [57] and by the IAEA [58, 59].

At the current generic stage, RWMD needs to ensure that it considers appropriate FEPs and safety functions that are sufficiently generic to encompass the range of illustrative potential geological environments and disposal concepts; this is likely to lead to a greater focus on qualitative reasoning, perhaps supported by illustrative scoping calculations. As RWMD starts to develop a site-specific ESC, it will be possible to add greater definition to the safety functions, which will be reflected in a corresponding greater definition of the safety assessment within the safety case.

### 7.2 Role of mathematical analyses and modelling strategy

In each of the safety cases studied, mathematical modelling is a key tool in the demonstration of safety. Such modelling is used to develop and demonstrate understanding of a site or physical process, to decide whether a process needs to be taken into account explicitly in performance assessment, and provide the method of calculating quantitative risks or doses for comparison with national regulations.

Each of the five safety cases studied included probabilistic calculations, and detailed consideration of variability and sources of uncertainty in the models. Details of modelling are generally included in the lower tier reports of a safety case or in appendices, with only the key results presented in the top-level documentation. US DOE, SKB and LLW Repository Ltd have also developed simple ‘insight’ models to supplement their more complex performance assessment models.

In each of the examples studied, the scenarios and conceptual models used in performance assessment are developed using a systematic, documented process of FEP analysis and / or the consideration of the safety functions of the disposal concept, as noted in Section 7.1 above. This is consistent with the RWMD approach to model development in which conceptual models are developed from a FEP analysis process and then developed into mathematical and software models.

RWMD also recognises that models are developed at different levels of detail and used in different ways within a safety case. RWMD’s modelling strategy [61] involves developing a hierarchy of models, where a performance assessment model which represents the whole
system is supported by component and process level models which represent specific aspects of the system in more detail. Other than in the Yucca Mountain Licence application, this terminology is not used by other waste management organisations, however, it could be argued that this approach has been adopted implicitly.

Adopting RWMD’s terminology, the details of how component and process level models are linked together to form the system-level model varies considerably between the examples studied. For example, the performance assessment models used in SR-Site are linked to the detailed hydrogeological models in a relatively simple manner by passing an output file from the hydrogeological model to the system level model. By contrast, in the WIPP CCA and CRA, calculations of multiphase flow coupled with mechanical effects were undertaken within the system level model used for risk assessment. The difference in approach is likely to be a consequence of the different physical processes identified as being of significance in the different disposal concepts: coupled processes may be more important in the halite environments at WIPP compared to the granitic environment at Forsmark. The Yucca Mountain licence application discusses the issues around abstracting the details of component and process level models into a system level model; DOE used a mixture of approaches including statistical or mathematical abstractions, look-up tables, equations representing response surfaces, probability distributions, linear transfer functions, or reductions of model dimensionality.

SR-Site refers to analytical models and RWMD has also developed ‘insight’ models, for both ILW [64] and HLW [65] disposal systems, based on analytical understanding of the physical and chemical processes that determine the expected peak radiological releases of each radionuclide from different disposal systems. It is anticipated that these relatively simple models will continue to play an important role in developing understanding of system performance, testing sensitivity to different parameters and concepts and providing a check for the results from more complex models.

7.3 Register of uncertainties


“...[a GDF implementer should take] adequate account of all uncertainties that have a significant effect on the environmental safety case. This will mean establishing and maintaining a register of significant uncertainties...”

The treatment of uncertainties is an important aspect for all waste management organisations; however, given the GRA quote above, it is particularly relevant to note that Andra explicitly refers to an “inventory of uncertainties” within Dossier 2005 Argile. This is detailed in the Safety Evaluation [16] which presents a summary of the main conclusions from Andra’s assessment of uncertainty. There are several elements to Andra’s approach in this area. First, Andra determines the main uncertainties. These are identified on the basis of possible interactions between the disposal facility and the surrounding environment which leads to the identification of a list of possible events that are, in principle, unfavourable for the facility’s safety assessment. The evolutionary analysis of disposal scenarios and the description of the conceptual models’ formal framework, in which the conceptual models are established, are both used to help make the identification of uncertainties more systematic.

Once the uncertainties are identified, a systematic study is undertaken which looks at:

- which components are affected by which uncertainties, including the effects caused by one component acting on another;
- which performance aspects can impact the safety functions and the time period over which the safety function may be affected;
• a qualitative assessment of the likelihood of these effects; and
• whether combinations of uncertainties could lead to effects that deviate significantly from the normal evolution.

The aim of the analysis is to carefully identify the importance of each uncertainty on the disposal system’s overall behaviour. The uncertainties are divided into groups depending on their origin (for example inventory, external events, likelihood of design defects, etc.). Each is then assessed in the context of the total system to rank the uncertainty in terms of its potential significance to overall safety. Some uncertainties were excluded during the qualitative analysis, as they were judged to have no, or very little, effect on safety in any scenario considered. The possible ranges of variations of the parameters are identified, together with the system components likely to cause variations in overall behaviour.

This analysis therefore generates a prioritised list of uncertainties, guided by the manner in which they could affect the safety functions. If the impact is covered within the normal evolution scenario the uncertainty need not be considered further. If this is not the case, there may be a basis in which to represent this uncertainty in the alternative evolution scenarios and/or through sensitivity studies.

In the 2011 LLWR ESC, a FEP list was developed and used to trace how each FEP and its related uncertainties were treated, based on the conceptual models actually envisaged in the Long Term Radiological Assessment. The FEP and uncertainty tracking system is implemented as a Microsoft Excel™ spreadsheet including screens and macros to assist in the management of the FEPs and associated uncertainties. It includes consideration of how each FEP is treated and a means by which significant uncertainties can be identified and rated according to expert judgement regarding their importance to sub-system performance.

It is interesting to note that the international peer review of SR-Can [66], i.e. the assessment that preceded SR-Site [62], suggested that SKB should consider compiling a register of uncertainties. The Swedish regulators, however, in their review of the SR-Can assessment, jointly decided after consideration not to require such a register of uncertainties because no obvious advantage was identified to justify the required effort. Rather, the regulators pointed to the importance of justifying and explaining methods to handle different uncertainties in the different phases of the safety assessment, and that it is clear where in the safety report the different uncertainty analyses are documented.

Given the long timescales that are addressed within a post-closure safety case, it is inevitable that there will be a considerable number of uncertainties, even for relatively advanced safety cases. The crucial point is that many of these uncertainties will not be significant in terms of the overall safety. This is recognised in the GRA, as quoted above, in that only a register of significant uncertainties is required. Of course, the safety case may have to explain why some uncertainties are not judged to be significant. This is likely to require appropriate reasoned argument and/or sensitivity analyses in order to justify the relative significance of outstanding uncertainties. Maintaining such a register of significant uncertainties will also be beneficial in prioritising the disposal facility research programme, by focusing research on reducing those uncertainties that have the greatest impact on confidence in the safety case, and thus demonstrating a needs-driven approach to research.

7.4 Assessment timeframes

Assessment timeframes area generally constrained by specific national regulations.

Swedish regulations indicate that the timescale of a safety assessment for spent nuclear fuel should be one million years after closure – with a detailed risk analysis required for the first 1,000 years and a quantitative risk analysis up to approximately 100,000 years. Beyond 100,000 years quantitative risk calculations are not considered meaningful, but the
safety case should still demonstrate that releases from the engineered and geological barriers are limited and delayed as far as reasonably possible, using calculated risk as one of several indicators.

SR-Site actually presents numerical safety calculations up to one million years and thereafter provides qualitative reasoning (including discussion of natural analogues) to provide confidence in continued safety.

When considering assessment timescales, Andra is guided by the Basic Safety Rule. In keeping with international practice, Dossier 2005 Argile extends performance assessment calculations against the Basic Safety Rule recommended dose limit to a period of one million years. Although Andra’s approach to assessment remains consistent over all timescales, it is acknowledged that different system components and processes are more or less relevant depending on the timeframe of interest.

US federal regulations define the performance assessment timescale for assessments of the WIPP facility to be 10,000 years. This is therefore the period modelled within the numerical performance assessment which underpins the WIPP CCA.

For the Yucca Mountain licence application, initial regulations required demonstration that the mean annual dose up to 10,000 years post-closure was no more than 0.15 mSv. Subsequently, an individual protection standard applying up to one million years post-closure was introduced, during which period the mean annual dose should not exceed 1 mSv. The Yucca Mountain performance assessment therefore includes a probabilistic dose assessment up to one million years.

LLWR Repository Limited has a slightly different approach to defining the assessment timescales as it is a near-surface facility and the site is located near to the sea in an area of coastal erosion, meaning that different timescales are appropriate compared to deep geological facilities.

The ESC draws upon both qualitative and quantitative evidence including modelling studies, and concludes that the coastal site for the LLWR facility is likely to begin to be eroded on a timescale of a few hundred to a few thousand years, with consequent disruption of the repository, with erosion of the vaults and trenches being complete within one to a few thousands of years.

Based on this, the LLWR ESC presents two scenarios; the first is the expected evolution scenario which calculates releases on the completion of closure engineering, at the end of management control, 100 years after present and 500 years after present (based on the start of coastal erosion). The second scenario is a delayed coastal erosion scenario where impacts are considered for up to 10,000 years after present with the assumption (considered to be unlikely) that the site will still not be completely eroded after 10,000 years. Several other timescales are studied for assessing the contributions to dose and the rationale for choosing these timescales is also provided.

### 7.5 Use of ‘Site Descriptive Models’

SKB uses the phrase ‘Site Descriptive Model’ (or SDM) to refer to the collation of evidence-based conceptual understanding of the GDF site and its expected evolution. This is an important concept in SR-Site and the development of the SDM underpins all the safety analyses.

Andra does not refer specifically to a site descriptive model but has set up a broader system that captures all knowledge about the geological disposal system known as PARS (phenomenological analysis of repository situations). This system describes the phenomena (thermal, mechanical, hydraulic, chemical, and radiological) and their couplings throughout the geological disposal facility evolution and specifies the phases of this evolution from its construction up to 1 million years. This is broken into a number of
spatial scales relevant to components of the system including the geological setting. The knowledge captured within PARS is not constrained by its use in safety assessments. However, the safety assessments draw upon this knowledge base in order to produce safety arguments and develop a conceptual model of the system.

Neither of the US safety cases studied refers to a ‘site descriptive model’. Rather, the WIPP performance assessment model is based on a series of conceptual release pathways for undisturbed and disturbed release scenarios. These scenarios describe the significant processes identified through the FEP analysis and the release pathways feature the Salado formation together with overlying units. The Yucca Mountain Licence application used a series of detailed process models to underpin the performance assessment; these included climate analysis, an infiltration model, models of the unsaturated and saturated zones, near-field chemistry, thermal hydrology and the EBS [50]. The groundwater regime is of less importance for the US safety cases (which both relate to essentially ‘dry’ sites) and hence the role of a site descriptive model could be considered less relevant.

LLW Repository Ltd also does not use the specific terminology of a ‘site descriptive model’ in the 2011 ESC. However, LLWR Ltd has developed a detailed characterisation and understanding of the LLWR (including historical data) which is presented in the supporting reports of the 2011 ESC.

7.6 Roles of engineered and natural barriers

The five safety cases considered place different emphases on the engineered and natural barriers. In SR-Site the long-term integrity of the copper canister provides the main safety function; and other engineered barriers, in particular the buffer, have safety functions that provide protection to the copper canister. The natural geological barrier has an important role in providing a stable environment for the engineered barriers, in particular the canister. It also provides a secondary, retarding safety function, should the containment be breached.

In Dossier Argile, it is the clay geological environment that provides the main safety function, working in conjunction with the engineered barriers.

In the WIPP safety case there are few engineered barriers which contribute to post-closure safety (just the sacks of magnesium oxide) and the sealing properties of the salt and lack of groundwater provide the main safety functions. The waste inventory is, of course, carefully controlled to make it suitable for disposal in this environment (for example, the WIPP does not accept liquid waste). However, it is interesting to consider that in the WIPP safety case, the identified exposure routes all involve disruption of the geological barrier, e.g. by mining or drilling.

The Yucca Mountain Project proposed high integrity waste containers and even considered the use of titanium drip shields to prevent even the expected low levels of groundwater contacting the waste packages, i.e. a strong reliance on engineered barriers, whilst also recognising the desirable safety function on the unsaturated geological barrier.

Due to the near-surface nature of the LLWR facility, there is naturally more emphasis on the engineered barriers and active safety management and relatively little reliance is placed on the natural barriers. LLW Repository Ltd proposes to construct a final cap over the trenches and the vaults once all the disposals have been made. The cap is proposed to be three metres thick, excluding profiling material, and consist of a number of layers designed to deter intrusion by people, animals and plants, limit infiltration of rainwater into the wastes, and disperse any gas that accumulates during the lifetime of the facility. The plans also propose to build a cut-off wall in the ground around the trenches and vaults under the edge of the cap. This will extend an existing cut-off wall on the north and east sides of the trenches. The cut-off wall will be made of a mixture of clay, cement and blast...
furnace slag. It will limit the flow of groundwater into the wastes and help prevent contaminated water flowing out near the surface.

7.7 Structure and presentation

The safety cases considered within this document show significant variations in both structure and presentation. Some, for example LLWR, adopt a specific multi-level approach, with the level and detail of technical content determined by the level of the document. For example, a high-level non-technical document may provide an overview for a general audience, while a ‘level 3’ document may provide a detailed scientific discussion. Others, for example WIPP, provide relatively detailed information throughout, with chapter introductions used to provide overview and set context. This approach is likely to be sensible if the document is intended primarily for a scientific audience well versed in the field (for example a regulator). Safety cases with the multi-level approach tend to issue numerous documents (e.g. overview reports, status reports etc.), whilst organisations adopting a single-level approach tend to issue fewer, but lengthier, documents (albeit with appendices and attachments).

In some safety cases (for example WIPP and Yucca Mountain), visual aids are relatively sparse and primarily of a technical nature, reflecting the regulatory audiences. This differs from safety cases intended for wider audiences, which tend to also include more colourful illustrations. Andra’s use of thumbnail graphics within the text, with a full-size plate at the end of the document is interesting as it enables both continuity of the text and easy reference to the illustration. A sample page containing these thumbnails is illustrated in Figure 7-1 and RWMD may wish to consider this style of presentation in future publications. The Yucca Mountain safety case includes all figures at the end of the document, thus maintaining the text flow, but potentially hindering the reader when referring to the figures. In short, as may be expected, safety cases tend to be written differently depending on the intended reader. A safety case written exclusively for regulators is different from one written for multiple audiences, especially if these readers are less versed in the field and literature.

The length of a safety case also varies significantly, with the complete licence application for SKB covering more than 10,000 pages and the WIPP CCA extending to 72,000 pages. With documents of such length, the ability to find the desired information is a key requirement for all stakeholders. In this regard it is interesting to note that DOE has attempted to use technology to make the WIPP safety case more useable and accessible. In particular DOE commissioned the production of two tools which are based on the original 72,000 page 1996 CCA:

- The Folio software was used to deliver an electronic version of the submission and all appendices, in a searchable and hyperlinked Infobase format. This approach brings an extremely powerful means of delivering information and allows very sophisticated searching (including, for example, proximity searches).
- The Toolbook software was used to produce the Performance Assessment Support System (PASS) V2.0. This database-like system allows the user to search FEPs, conceptual model parameterisation, computer model development and assumptions for an area of interest.

Although both products are somewhat dated now, and technology has moved on, both were found to be extremely useful in reviewing and navigating around the documents for the purpose of this report. It would be beneficial for RWMD to investigate whether modern technologies in the field of e-publishing may help the production or distribution of its safety case.
Figure 7-1 Use of thumbnail graphics in Andra’s Dossier 2005 Argile (reproduced from [22] © Andra 2005)

- C1 + C2 vitrified waste

Figure 5.5-22 SEN – million year model - Doses at the Saulx outlet of the Oxfordian – Reference packages (C1+C2)

- C3 + C4 vitrified waste

Figure 5.5-23 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – Reference packages (C3+C4)

- Reference packages B1x

Figure 5.5-24 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – Reference packages (B1x)
Another useful learning point for RWMD from the WIPP safety case is that the 1996 and 2004 submissions followed a chapter based format, which did not specifically link to the sections of the appropriate federal regulations (40 CFR 194). Instead, a regulatory crosswalk document was included to provide a bridge between the chapter format and the regulatory requirements addressed by each chapter. In response to the original submission, the EPA, acting as regulator, published its review in the form of documents, termed Compliance Application Review Documents (CARDs), which followed the structure of the regulations. As a result of this, in the 2009 submission, DOE restructured the safety case to follow the structure of the regulations more closely “to facilitate the EPA and stakeholder reviews of the application”. This was, of course, a significant task. It would be sensible for RWMD to review the layout of its safety case and ensure this is consistent with the expectations of its audience, particularly the regulators.

7.8 Summary

It has been very beneficial for the development of RWMD’s Post-closure Safety Group and hence for the on-going development of RWMD’s post-closure safety case to study these substantial international safety cases; to explore the different safety arguments employed and understand the different approaches to developing and structuring a safety case in a variety of contexts. Each safety case has its own specific challenges and requirements, but there is much that can be learned from understanding how a safety case has been developed and compiled to address those requirements; and this learning can be translated to inform the on-going development of RWMD’s own ESC.

The greatest challenge for the RWMD ESC is that it is currently generic [2], yet needs to provide an appropriate basis against which to assess ongoing proposals for the conditioning and packaging of UK wastes, such that there is confidence the waste packages will be suitable for disposal in whatever geological environment and facility design are eventually chosen. This is the reason that illustrative numerical calculations for post-closure releases are presented in the generic ESC [3]. Moving forward, it may be appropriate for RWMD to consider expanding a range of more qualitative and reasoned safety arguments that could be applied to specific geological environments. These safety arguments could perhaps be supported by insight modelling or simple scoping calculations. Such an approach may be helpful in providing an initial view of the likely post-closure safety issues at a particular site. This would then of course subsequently need to be developed and refined in the light of the understanding gained from the eventual site characterisation programme.
8 References


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[34] R. Iman and W. Conover, “A Distribution-Free Approach to Inducing Rank Correlation


