Nuclear
Decommissioning
Authority

ASSESSMENT REPORT

Generic Design Assessment: Disposability Assessment for Wastes and Spent Fuel arising from Operation of the UK EPR
Part 1: Main Report

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EXECUTIVE SUMMARY

Introduction

The 2008 White Paper on Nuclear Power[1], together with the preceding consultation[2], established the process of Generic Design Assessment (GDA), whereby industry-preferred designs of new nuclear power stations would be assessed by regulators in a pre-licensing process. Amongst the parties requesting assessment under the GDA process is a collaborative venture between Electricitié de France (EdF) and Areva NP (EdF/Areva), which is seeking an initial endorsement of the UK EPR design.

An important aspect of the GDA process is the consideration of the disposability of the higher activity solid radioactive wastes and spent fuel that would be generated through reactor operation. Consequently, regulators have indicated that requesting parties should obtain and provide a view from the Nuclear Decommissioning Authority (NDA) (as the authoritative source in the UK for providing such advice) on the disposability in a Geological Disposal Facility of any proposed arisings of higher activity wastes or spent fuel[3].

In accordance with regulatory guidance, EdF/Areva has requested that the Radioactive Waste Management Directorate (RWMD) of NDA provide advice on the disposability of the higher activity wastes and spent fuel expected to arise from the operation of an EPR. The reported assessment of the disposability of the higher activity wastes and spent fuel from the EPR is based on information on wastes and spent fuel, and proposals for waste packaging supplied by EdF/Areva, supplemented as necessary by relevant information available to RWMD.

This GDA Assessment Report presents the results of the disposability assessment undertaken by RWMD, together with comprehensive details of the wastes and spent fuel, and their characteristics, including measures taken by RWMD to supplement the information provided by EdF/Areva.

The GDA Disposability Assessment process comprises three main components: a review to confirm waste and spent fuel properties; an assessment of the compatibility of the proposed disposal packages with concepts for geological disposal; identification of the main outstanding uncertainties, and associated research and development needs relating to the future disposal of the wastes and spent fuel.

It is recognised that at this early stage in reactor licensing and development of operating regimes, packaging proposals are necessarily outline in nature, however, this Disposability Assessment has led to the production of a comprehensive and detailed data set describing the higher activity wastes and spent fuel to be generated from operation and decommissioning of an EPR. At a later stage in the licensing process for new reactors, RWMD would expect to assess more specific and detailed proposals through the existing Letter of Compliance process for endorsement of waste packaging proposals[4].

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[1] Meeting the Energy Challenge, *A White Paper on Nuclear Power*, Cm 7296, January 2008.

[2] The Future of Nuclear Power, *The Role of Nuclear Power in a Low Carbon UK Economy*, Consultation Document, URN 07/970, May 2007.

[3] Environment Agency, *Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs*, January 2007.

[4] NDA RWMD, *Guide to the Letter of Compliance Assessment Process*, NDA Document WPS/650, March 2008.

Nature of the Higher Activity Wastes and Spent Fuel

EdF/Areva has provided information on the higher activity wastes and spent fuel expected to arise from an EPR operating for 60 years with a maximum fuel assembly average irradiation (burn-up) of 65 GWd/tU. In line with the White Paper[1], spent fuel from a new nuclear power programme is assumed to be managed by direct disposal after a period of interim storage.

Three general categories of higher activity wastes and spent fuel are identified in this report: intermediate-level waste (ILW) arising from reactor operation, ILW arising from reactor decommissioning, and spent fuel. EdF/Areva has provided information for the following six types of operational waste that could potentially be classified as ILW:

- Ion exchange resins;
- Spent cartridge filters (ILW) – higher activity filters from the reactor primary circuit;
- Spent cartridge filters (LLW and ILW)[5] – other designs of filter, typically with lower activity;
- Operational wastes with a dose rate >2mSv/hr – those general operational wastes that would be categorised as ILW, as determined by dose-rate;
- Wet sludges;
- Evaporator concentrates.

EdF/Areva has indicated that the decommissioning ILW should be assumed to comprise the more highly activated steel components that make up the reactor vessel and its internals, and information has been assessed accordingly. In practice, decommissioning wastes will comprise a mix of ILW and LLW. Further development of decommissioning plans in the future will provide an improved understanding of the expected quantities of ILW, although that detail is not required for this GDA Disposability Assessment.

As indicated above, information on spent fuel has been supplied by EdF/Areva based on an assumed maximum fuel assembly average burn-up of 65 GWd/tU. It has been conservatively assumed that all spent fuel would achieve this burn-up. In practice, it is likely that this value represents the maximum of a range of burn-up values for individual fuel assemblies.

Proposals for Packaging

EdF/Areva has put forward proposals for the packaging of operational ILW based on operational experience for existing designs of Pressurised Water Reactors (PWR) in France. These proposals are based on the use of reinforced concrete casks as waste containers. This packaging option is denoted the "Reference Case".

The concrete casks proposed in the Reference Case packaging option have not been considered by RWMD in previous disposability assessments. These containers are therefore currently denoted non-standard. Furthermore, such casks might not be adopted by all future operators of the EPR. Consequently EdF/Areva has proposed two variant cases for the packaging of operational ILW from the EPR, based on the use of UK standard containers, and cast-iron casks as used in Germany for the packaging of certain light water reactor (LWR) wastes. As with the concrete casks, the cast-iron casks are presently considered to be non-standard in the UK.

The three packaging options for operational ILW may be summarised as follows:

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[5] Some items included in this waste stream might be able to be categorised as LLW but, conservatively, all wastes within this stream are being considered as potentially requiring disposal as higher activity waste.

- Reference Case – use of reinforced concrete casks as used in France for the packaging of similar operational wastes from PWRs;
- Variant Case 1 – use of stainless steel 500 litre Drums consistent with RWMD standards and specifications;
- Variant Case 2 – use of cast-iron casks as used in Germany for the packaging of similar operational wastes from LWRs.

The proposals for the packaging of decommissioning ILW are based on the use of larger waste containers consistent with RWMD standards and specifications (the containers designated are the $3m^3$ Box and 4 metre Box), with no variants being proposed.

The GDA Disposability Assessment has assumed that the spent fuel assemblies will be packaged in a robust disposal canister for disposal. For the purposes of this assessment, the spent fuel disposal canister is assumed to be manufactured from either copper or steel, with the fuel assemblies loaded into a cast-iron inner vessel. For consistency with previous assessments of the disposal of spent fuel undertaken by RWMD, it has been assumed that each disposal canister would contain up to four spent fuel assemblies. It is further assumed that the spent fuel would be delivered to the disposal facility packaged in the disposal canisters.

Radionuclide Inventory of ILW and Spent Fuel

The information supplied by EdF/Areva on the radionuclide inventories of the identified wastes and spent fuel has been used to derive assessment inventories for the proposed disposal packages, including the variants for operational ILW. In some cases, to ensure a full coverage of potentially significant radionuclides, it has been necessary to supplement the information supplied by EdF/Areva using information available to RWMD. The assessment inventories are intended to characterise the range of disposal package inventories, taking account of uncertainties and the potential variability between packages. The assessment inventory defines a best-estimate (average) and bounding (maximum) inventory for a disposal package.

The uncertainties in the inventories arise from numerous sources, for example the reactor operating regime adopted, fuel burn-up, fuel irradiation history, possible fuel cladding failures and the disposal package loadings that will be achieved in practice. The GDA Disposability Assessment has used expert judgement to bound this uncertainty and thereby provide robust, conservative conclusions. It is anticipated that information on the inventories associated with the wastes and spent fuel will be refined as the design of the reactors and their operating regimes are developed further. RWMD would expect to consider such information, together with more refined packaging proposals, at an appropriate time in the future through the Letter of Compliance process.

Examples of opportunities for the refinement of data and removal of conservatisms include the assumptions relating to the incidence of fuel cladding failure (and the resultant activity associated with ILW ion exchange resins and filters), the pre-cursor concentrations for important activation products such as carbon-14 and chlorine-36 in the reactor and fuel assembly components, and the influence of the distribution of the fuel assembly burn-up.

It is particularly noted that the inventory associated with the spent fuel has been based on the conservative assumption that the maximum fuel assembly average burn-up of 65 GWd/tU applies uniformly to all fuel assemblies for disposal. In practice, the burn-up will vary with the operating history experienced by the assembly and the average burn-up of all assemblies would be less than 65 GWd/tU.

RWMD has concluded that the inventory data supplied by EdF/Areva, together with measures implemented by RWMD to supplement the data, has provided a comprehensive data set sufficient to provide confidence in the conclusions of the GDA Disposability Assessment.

The GDA Disposability Assessment has shown that the principal radionuclides present in the wastes and spent fuel are the same as those present in existing UK legacy wastes and spent fuel, and in particular, with the anticipated arisings from the existing PWR at Sizewell B. This conclusion reflects both the similarity of the designs of the EPR and of existing PWRs, and the expectation that similar operating regimes would be applied.

The adoption of a higher burn-up for the EPR, as compared to Sizewell B, is expected to result in increased concentrations of radionuclides in the spent fuel. Also, the longer operational life of the EPR (60 years as compared to 40 years anticipated for Sizewell B) increases the concentration of long-lived radionuclides in the decommissioning waste. The potential significance of such differences has been considered. The radionuclide inventory associated with the operational ILW will depend on operating decisions, for example the permitted radioactive loadings of ion exchange resins and filters, and therefore could be managed with the aim of meeting specific requirements for disposal.

Assessment of Proposed ILW Packages

The proposals for the packaging of ILW include outline descriptions of the means proposed for conditioning and immobilising the waste. Detailed descriptions and supporting evidence as to the performance of the proposed packages are not provided at this stage. This is consistent with expectations for the GDA Disposability Assessment. In future, RWMD would expect to work with potential reactor operators and provide assessment of fully-developed proposals through the Letter of Compliance process.

The Reference Case proposals, based on non-standard concrete casks, are not compliant with some aspects of existing RWMD standards for waste packages. Nevertheless, RWMD has judged that it should be feasible to develop design concepts for the transport of such packages to a Geological Disposal Facility, and for their subsequent handling and emplacement in disposal vaults. Further development of the proposed conditioning methods, using either a polymer or cement grout, would be required, but RWMD considers that, based on experience of similar wastes, suitable methods can be developed.

Although the concrete casks are licensed for the transport of wastes from existing PWRs in France, application of the EPR assessment inventory suggests that some packages from some streams containing operational ILW at the bounding inventory could exceed dose-rate limits permitted under current Transport Regulations. RWMD has judged that this issue may be addressed through future refinement of the assessment inventories, including provision of better data to remove pessimisms, consideration of an appropriate time for radioactive decay and/or development of the detailed packaging methods, such as provision of more shielding in the packages.

The proposal under Variant Case 1 to use RWMD standard waste containers provides compliance with many aspects of the existing standards and specifications. Furthermore, the requirement for such packages to be transported in a reusable shielded transport over-pack eliminates potential challenges to the dose-rate limits set out in the IAEA Transport Regulations.

EdF/Areva has indicated that most of the operational ILW would not be directly conditioned into the 500 litre Drums under Variant Case 1. Instead, the wastes would be packed into smaller containers (200 litre drums) that would be grout enclosed within the 500 litre Drums. This does not represent common practice in the UK and although it represents the smallest overall volume of the three packaging options, more efficient use still could be made of the available volume. Nevertheless, the GDA Disposability Assessment has concluded that the necessary performance potentially would be available from such packages due to their robust nature. Furthermore, it is also noted that full immobilisation could be achieved through application of a conditioning process to the materials inside the 200 litre drum. An alternative option would be to directly loaded and condition the materials into 500 litre Drum, if such an option is adopted.

The Variant Case 2 proposals, based on non-standard fully sealed cast-iron casks, are not compliant with some aspects of existing RWMD standards for waste packages. Nevertheless, RWMD has judged that it should be feasible to develop design concepts for the transport of such packages to a Geological Disposal Facility, and for their subsequent handling and emplacement in disposal vaults. It is noted that such packages are currently approved for the packaging of ILW from light water reactors in Germany.

The Variant Case 2 proposals are similar to Variant Case 1 in that they would contain unimmobilised wastes. Again, it is anticipated that the robust nature of the containers alone potentially would provide the necessary performance. Further demonstration ultimately would be required of the means of treating the wastes prior to packaging. In particular drying to remove water to control the evolution of the wastes and prevent gas pressurisation. Nevertheless, it is judged that viable treatment processes are currently available.

The proposed decommissioning ILW packages comprise metal items conditioned in standard containers using a cement grout. These proposals conform to existing practices for similar wastes in the UK and are expected to produce packages that would be compliant with existing RWMD standards and specifications. The current bounding assessment inventory for the decommissioning ILW proposed to be packaged in 4 metre Boxes challenges some aspects of the Transport Regulations in relation to dose-rates but it is judged that this issue could be addressed by refining the assessment inventory, modifying the proposals to include additional shielding, allowing for radioactive decay and/or management of waste loading. Alternatively, employing containers that necessitate the use of a reusable shielded over-pack for transport (i.e. the $3m^3$ Box proposed for the remainder of the decommissioning ILW) would also address these challenges.

The assessment of long-term disposal system performance in the GDA Disposability Assessment has been undertaken by comparison with the assessment performed for legacy ILW. This was based on the assumed characteristics for a generic UK Geological Disposal Facility site. Since the properties of any selected site necessarily would need to be consistent with meeting the regulatory risk guidance level[6], based on the approach adopted for Letter of Compliance assessment, this assessment assumed a groundwater flow rate and return time to the accessible environment that would meet regulatory requirements when considering the inventory of legacy ILW. The additional radionuclide inventory associated with the ILW from an EPR represents only a small fraction of that of the legacy wastes, particularly for the majority of the radionuclides that determine risk in the long-term. Even considering the conservative approach to inventory assessment and recognising the potential for future optimisation of packaging proposals, the additional risk from the disposal of ILW from a single EPR in a site of the type described would be consistent with meeting the regulatory risk guidance level. The consideration of a fleet of six reactors does not alter this conclusion.

Overall, all three cases for the packaging of operational ILW and the proposals for the packaging of decommissioning ILW have been judged to be potentially viable. While further development needs have been identified, including ultimately the need to demonstrate the expected performance of the packages, these would represent requirements for future assessment under the Letter of Compliance process.

The number and type of new build reactors that may be constructed in the UK is currently not defined. Therefore, the GDA Disposability Assessment has evaluated the implications of a single EPR and, to illustrate the potential implications of constructing a fleet of such reactors, consideration also has been given to a fleet of six EPR reactors. This corresponds to a generating capacity of about 10 GW(e), equivalent to the capacity of the existing nuclear reactors in the UK expected to cease operations in the next 20 years.

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[6] Environment Agency and Northern Ireland Environment Agency, *Geological Disposal Facilities on Land for Solid Radioactive Wastes: Guidance on Requirements for Authorisation*, February 2009.

The potential impact of the disposal of EPR operational and decommissioning ILW on the size of a Geological Disposal Facility has been assessed. Although the impact has some dependence on the packaging variant considered for operational ILW, it has been concluded that in all cases the necessary increase in the 'footprint area' is small, corresponding to less than approximately 60m of disposal vault length for each EPR. This represents approximately 1% of the area required for the legacy ILW, per reactor, and less than 10% for the illustrative fleet of six EPR reactors. This is in line with previous estimates for potential new build reactor designs[7].

Assessment of Spent Fuel Packages

EdF/Areva has indicated that the GDA Disposability Assessment for the EPR should assume that the reactor would operate with uranium dioxide fuel 5% enriched in uranium-235 to achieve a maximum fuel assembly average burn-up of 65 GWd/tU. This burn-up is higher than that achieved at the existing PWR at Sizewell B.

In practice, the average burn-up for EPR spent fuel assemblies would be less than 65 GWd/tU and this maximum would represent the extreme of a distribution of burn-up values for individual fuel assemblies. However, in the absence of detailed information on the distribution of burn-up between fuel assemblies, for the purposes of the GDA Disposability Assessment it has been conservatively assumed that the value of 65 GWd/tU applies uniformly to them all.

Increased burn-up implies that the fuel is used more efficiently and that the volume of fuel to be disposed of will be smaller per unit of electricity produced. However increased irradiation leads to individual fuel assemblies with an increased concentration of fission products and higher actinides, leading in turn to assemblies with higher thermal output and dose-rate. This difference is recognised as an important consideration in the assessment of spent fuel from the EPR.

The GDA Disposability Assessment for the EPR has assumed that spent fuel would be over-packed for disposal. Under this concept, spent fuel would be sealed inside durable, corrosion-resistant disposal canisters manufactured from suitable materials, which would provide long-term containment for the radionuclide inventory. Although the canister material remains to be confirmed, the assessment has considered the potential performance of both copper and steel canisters. In both cases, it is assumed that a cast-iron inner vessel is used to hold and locate the spent fuel assemblies, and in the case of the copper canister would provide mechanical strength as well. Over-packing of spent fuel in robust containers for disposal is a technology that is being developed in several overseas' disposal programmes.

Current RWMD generic disposal studies for spent fuel define a temperature criterion for the acceptable heat output from a disposal canister. In order to ensure that the performance of the bentonite buffer material to be placed around the canister in the disposal environment is not damaged by excessive temperatures, a temperature limit of 100°C is applied to the inner bentonite buffer surface. Based on a canister containing four EPR fuel assemblies, each with the maximum burn-up of 65 GWd/tU and adopting the canister spacing used in existing concept designs, it would require of order of 100 years for the activity, and hence heat output, of the EPR fuel to decay sufficiently to meet this temperature criterion.

It is acknowledged that the cooling period specified above is greater than would be required for existing PWR fuel to meet the same criterion and RWMD proposes to explore how this period can be reduced. This may be achieved for instance through refinement of the assessment inventory (for example by considering a more realistic distribution of burn-up), by reducing the fuel loading in a canister, or by consideration of alternative disposal concepts.

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[7] United Kingdom Nirex Limited, *The Gate Process: Preliminary Analysis of Radioactive Waste Implications Associated with New Build Reactors*, Nirex Technical Note Ref: 528386, February 2007.

The sensitivity of the cooling period to fuel burn-up has been investigated by consideration of an alternative fuel inventory based on an assembly irradiation of 50 GWd/tU. For this alternative scenario it is estimated that the cooling time required will reduce to the order of 75 years to meet the same temperature criterion.

RWMD planning for the transport of packaged spent fuel to a Geological Disposal Facility and the subsequent emplacement of containers is at an early stage of development. Consequently, although the EPR spent fuel may influence the necessary arrangements, for example through the need for additional shielding, it is judged that sufficient flexibility exists in the current concept to allow suitable arrangements to be developed.

The GDA Disposability Assessment has considered how spent fuel packages would evolve in the very long term post-disposal, recognising that radionuclides would be released only subsequent to a breach in a disposal canister. A limited sensitivity analysis has been performed, examining two different canister materials (copper and steel) and testing the influence of the assumed corrosion properties.

Subsequent to any canister failure, the radionuclides associated with the spent fuel would be able to leach into groundwater. The rate at which radionuclides are leached, in combination with the assumed properties of the host rock, the behaviour of individual radionuclides and exposure routes are then used to assess the potential risk to humans.

The leaching of radionuclides from spent fuel is characterised by an initial 'instant release fraction' (IRF), and then by a more general dissolution rate. The IRF is the fraction of the inventory of more mobile radionuclides that is assumed to be readily released upon contact with groundwater and is influenced by the properties of the spent fuel. In the case of higher burn-up fuel, the increased irradiation of the EPR fuel would increase the IRF as compared to that for lower burn-up fuel. Generally available information[8] on the potential performance of higher burn-up fuel has been used to provide a suitable IRF for assessment.

The assessment of long-term disposal system performance in the GDA Disposability Assessment has been based on the assumed characteristics of a generic UK Geological Disposal Facility site. Since the properties of any selected site necessarily would need to be consistent with meeting the regulatory risk guidance level, this assessment assumed the same site characteristics as assumed for the existing RWMD generic assessment. On the basis of the information provided and what are expected to be conservative calculations of canister performance, it is estimated that the spent fuel from a fleet of six EPR reactors would give rise to an estimated risk below the risk guidance level based on these geological conditions and the existing safety case arguments.

The risks calculated for the disposal of spent fuel reflect the assumed performance of the proposed packaging methods. The sensitivity analysis demonstrated that while the calculated risk would be influenced by assumptions about the canister materials, for the assumed characteristics of the canisters and the disposal site, risks always remained below the regulatory guidance level, regardless of any impact that the high burn-up experienced by the fuel assemblies would have on the IRF.

RWMD recognises that the performance of disposal canisters will be an important element of a safety case for the disposal of spent fuel. Consequently, it is anticipated that RWMD will continue to develop canister designs, with the intention of substantiating current assumptions and optimising the designs.

The potential impact of the disposal of EPR spent fuel on the size of a Geological Disposal Facility has been assessed. The assumed operating scenario for an EPR (60 years operation) gives rise to an estimated 900 disposal canisters, requiring an area of

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[8] Nagra Technical Report, *Estimates of the Instant Release Fraction for UO_2 and MOX fuel at t = 0*, Nagra TR 04-08, November 2004.

approximately 0.15 km^2 for the associated disposal tunnels. A fleet of six such reactors would require an area of approximately 0.9 km^2, excluding associated service facilities. This represents approximately 8% of the area required for legacy HLW and spent fuel per EPR reactor, and approximately 50% for the illustrative fleet of six EPR reactors. This is in line with previous estimates for potential new build reactor designs[7].

RWMD is currently developing a Generic Disposal System Safety Case covering the Baseline Inventory of waste and wastes that may potentially arise in the future as set out in the Managing Radioactive Waste Safely White Paper[9]. RWMD is also considering an upper bound inventory reflecting the uncertainty around the Baseline Inventory, including the potential for wastes and spent fuel to arise from a new nuclear build power programme. This will provide information on the disposability of the various categories of waste in a single 'co-located' facility. It is planned that the Generic Disposal System Safety Case will be published in September 2010 to support the Geological Disposal Facility site selection and assessment process. This will provide a baseline for the ongoing provision of advice on the disposability of wastes, including for future interactions on EPR waste and spent fuel.

Conclusions

RWMD has undertaken a GDA Disposability Assessment for the higher activity wastes and spent fuel expected to arise from the operation of an EPR. This assessment has been based on information on the nature of operational and decommissioning ILW, and spent fuel, and proposals for the packaging of these wastes, supplied to RWMD by EdF/Areva. This information has been used to assess the implications of the disposal of the proposed ILW packages and spent fuel disposal packages against the waste package standards and specifications developed by RWMD and the supporting safety assessments for a Geological Disposal Facility. The safety of transport operations, handling and emplacement at a Geological Disposal Facility, and the longer-term performance of the system have been considered, together with the implications for the size and design of a Geological Disposal Facility.

RWMD has concluded that sufficient information has been provided by EdF/Areva to produce valid and justifiable conclusions under the GDA Disposability Assessment. RWMD has concluded that ILW and spent fuel from operation and decommissioning of an EPR should be compatible with plans for transport and geological disposal of higher activity wastes and spent fuel. It is expected that these conclusions eventually would be supported and substantiated by future refinements of the assumed radionuclide inventories of the higher activity wastes and spent fuel, complemented by the development of more detailed proposals for the packaging of the wastes and spent fuel and better understanding of the expected performance of the waste packages. At such later stages, RWMD would expect to assess, and potentially endorse, more specific and detailed proposals through the established Letter of Compliance process for assessment of waste packaging proposals.

On the basis of the GDA Disposability Assessment for the EPR, RWMD has concluded that, compared with legacy wastes and existing spent fuel, no new issues arise that challenge the fundamental disposability of the wastes and spent fuel expected to arise from operation of such a reactor. This conclusion is supported by the similarity of the wastes to those expected to arise from the existing PWR at Sizewell B. Given a disposal site with suitable characteristics, the wastes and spent fuel from the EPR are expected to be disposable.

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[9] *Managing Radioactive Waste Safely: A Framework for Implementing Geological Disposal*, Cm 7386, June 2008.

TABLE OF CONTENTS

1 INTRODUCTION

1.1 Background

The 2008 White Paper on Nuclear Power [1], together with the preceding consultation [2], established the process of Generic Design Assessment (GDA), whereby industry-preferred designs of new nuclear power stations would be assessed by regulators in a pre-licensing process. Amongst the parties requesting assessment under the GDA process is a collaborative venture between Electricitié de France (EdF) and Areva NP (EdF/Areva), which is seeking an initial endorsement of the UK EPR design.

An important aspect of the GDA process is the consideration of the disposability of the higher activity solid radioactive wastes and spent fuel that would be generated through reactor operation. Consequently, regulators have indicated that requesting parties[10] should obtain and provide a view from the Nuclear Decommissioning Authority (NDA) (as the authoritative source in the UK in providing such advice) on the disposability in a geological disposal facility (GDF) of any proposed arisings of higher activity wastes or spent fuel [3].

In accordance with regulatory guidance, EdF/Areva has requested that the Radioactive Waste Management Directorate (RWMD) of NDA provide advice on the disposability of the higher activity wastes and spent fuel expected to arise from the operation of an EPR. The reported assessment of the disposability of the higher activity wastes and spent fuel from the EPR is based on information on wastes and proposals for waste packaging supplied by EdF/Areva, supplemented as necessary by relevant information available to RWMD.

Comprehensive details of the information supplied to RWMD by EdF/Areva, measures taken by RWMD to supplement this information, assessment methods and the detailed conclusions of this GDA Disposability Assessment are presented in this Assessment Report. This report is presented in two parts. This document is Part 1 and is the Main Report. Part 2 provides data summary sheets and inventory estimates for the proposed disposal packages. The principal conclusions and summary of the work undertaken by RWMD within the GDA Disposability Assessment are also presented in a separate summary level Disposability Report [4].

The GDA Disposability Assessment process comprises three main components: a review to confirm the waste properties; an assessment of the compatibility of the proposed waste packages with concepts for geological disposal of higher activity wastes and spent fuel; identification of the main outstanding uncertainties and associated research and development needs relating to the future disposal of the wastes.

It is recognised that at this early stage in the GDA process, waste packaging proposals are necessarily outline in nature. At a later stage in the licensing process for new reactors, RWMD would expect to assess more specific and detailed proposals through the existing Letter of Compliance assessment process [5].

[10] Requests for a Generic Design Assessment will normally originate from a reactor vendor. However, requests may also be initiated by vendor/operator partnerships. Consequently, the term ‘Requesting Party’ is used within the GDA process to identify the organisation seeking the GDA and to distinguish it from a nuclear site licence applicant.

The assessment has been undertaken in response to the purchase order from EdF/Areva dated 13 September 2008 (Purchase Order C451C80560, RWMD Document Reference #8675561) and is based upon the information set out in the submitted documents. The assessment has been performed in accordance with the terms and conditions of the Transport and Packaging Contract between EdF/Areva and NDA, dated 27 September 2008.

1.2 Objectives

The purpose of this GDA Disposability Assessment is to undertake assessment of the disposability of those higher activity wastes and spent fuel expected to be generated from operation of an EPR. The assessment has been commissioned by EdF/Areva to support its submission to regulators under the GDA process. The scope of the GDA Disposability Assessment has followed that set out and agreed with regulators in the protocol issued by RWMD in 2008 [6].

It is recognised that the nature and quantities of wastes, and the methods used to manage them following their generation, are subject to uncertainty at this stage of the process. Such uncertainties arise from the procedures that will be adopted in operating an EPR, and the processes and methods used to treat, condition and package wastes following their generation. Appropriate assumptions have been developed and applied in this GDA Disposability Assessment and are made explicit in this Assessment Report.

Therefore, the objective of the study is not to provide an endorsement of any particular packaging proposals, but to:

- provide a view on the disposability of higher activity wastes and radioactive materials (intermediate-level waste (ILW) and spent fuel) arising from operation and decommissioning of an EPR;

- comment on initial proposals by EdF/Areva for conditioning and packaging of ILW and spent fuel.

In the White Paper on Nuclear Power [1], the Government stated that despite some differences in characteristics, waste and spent fuel from new nuclear build would not raise such different technical issues as to require a different technical solution in comparison with nuclear waste from legacy programmes. A supplementary objective of the GDA Disposability Assessment is to confirm that the proposed wastes and spent fuel from an EPR present no technical issues compared to legacy wastes that would require a different technical solution. This has been undertaken by comparing the expected characteristics of the proposed wastes and spent fuel against the known characteristics of legacy wastes and spent fuel.

In addition, the White Paper flagged the importance of being able to give as much clarity as possible to communities considering hosting a GDF on the likely increases in both the volume and the level of radioactivity of the disposal inventory over and above that identified for legacy wastes and materials, that would arise from disposal of waste and spent fuel from new nuclear power stations. Therefore, a further supplementary objective of the GDA Disposability Assessment is to provide information on potential waste and spent fuel volumes and characteristics which would be of relevance to stakeholders of a GDF project. In fulfilling this objective RWMD has presented additional information for a fleet of EPR reactors noting that the actual impact on the UK's waste inventory as a result of new nuclear power stations would depend on the mix of reactor types and size of construction programme.

This document describes the GDA Disposability Assessment for the EPR and presents the results of the assessment. In particular, the report describes the higher activity wastes and spent fuel expected to be generated through operation and decommissioning of the EPR, describes options for conditioning and packaging these materials and identifies issues and further information requirements from the perspective of transport and disposal, which would need to be addressed in the future.

1.3 Scope

The GDA Disposability Assessment considers three types of waste and materials:

- ILW arising from reactor operations (operational ILW);
- ILW arising from the decommissioning of the reactor and associated plant (decommissioning ILW);
- spent fuel arising from reactor operation.

Wastes being dealt with through alternative routes, e.g. low-level waste (LLW) and/or very low-level waste (VLLW) are not considered within the scope of this Disposability Assessment.

In line with the White Paper [1], spent fuel from a new nuclear power programme is assumed to be managed by direct disposal after a period of interim storage.

The GDA Disposability Assessment considers as its baseline, the ILW and spent fuel arising from the operation and decommissioning of a single EPR, as described in Section 3. However, the disposal implications of a fleet of reactors are also considered where appropriate. The number of reactors that will be built and operated in the UK is subject to uncertainty. For the purposes of this report, the analysis has been based on operation of six EPRs, which would provide generating capacity of approximately 10 GW(e) (six EPRs would produce approximately 9.6 GW(e)). This assumption is made purely to facilitate comparison with legacy wastes and spent fuel and to consider disposability implications of a reasonably sized fleet, and does not indicate the size of any expected EPR reactor programme.

1.4 Document Structure

This GDA Assessment Report for the EPR is structured as follows:

- Section 2 provides a summary of the approach taken in the GDA Disposability Assessment, in particular describing the specifications against which EdF/Areva proposals have been assessed and the assessment methodology applied;
- Section 3 provides an overview of the EPR, the assumptions regarding operation of an EPR used in the GDA Disposability Assessment and summarises the inventory, packaging proposals, disposal package numbers and disposal package characteristics for EPR ILW and spent fuel;
- Section 4 describes the assessment of EPR operational and decommissioning ILW;
- Section 5 describes the assessment of EPR spent fuel;
- Section 6 presents the conclusions;
- Appendix A provides a summary of the Letter of Compliance process;

- Appendix B lists issues identified during the assessment that would need to be addressed by plant operators in future Letter of Compliance interactions.

2 APPROACH TO GDA DISPOSABILITY ASSESSMENT

2.1 Assessment Context

The GDA Disposability Assessment for the EPR has considered the conditioning and packaging proposals put forward by EdF/Areva. These packaging proposals have been assessed in relation to their compatibility with RWMD's existing specifications. These specifications include Waste Package Specifications [7] and [8], which consider disposal to a GDF based on Disposal System Specifications provided in [9] and [10].

The reference geological disposal concept for ILW used in the provision of disposability advice (Figure 1) envisages conditioning and packaging of ILW in standardised, highly-engineered stainless steel or concrete containers. The waste packages would be emplaced in disposal vaults constructed at depth in a suitable geological environment. When it is time to ultimately close the facility, a cement-based backfill would be placed around the disposed waste packages and this will act as a chemical barrier, sorbing and reducing the solubility of key radionuclides. The geological barrier would provide a long groundwater travel time and dilution and dispersion for those radionuclides that do not decay in-situ within the engineered barriers.

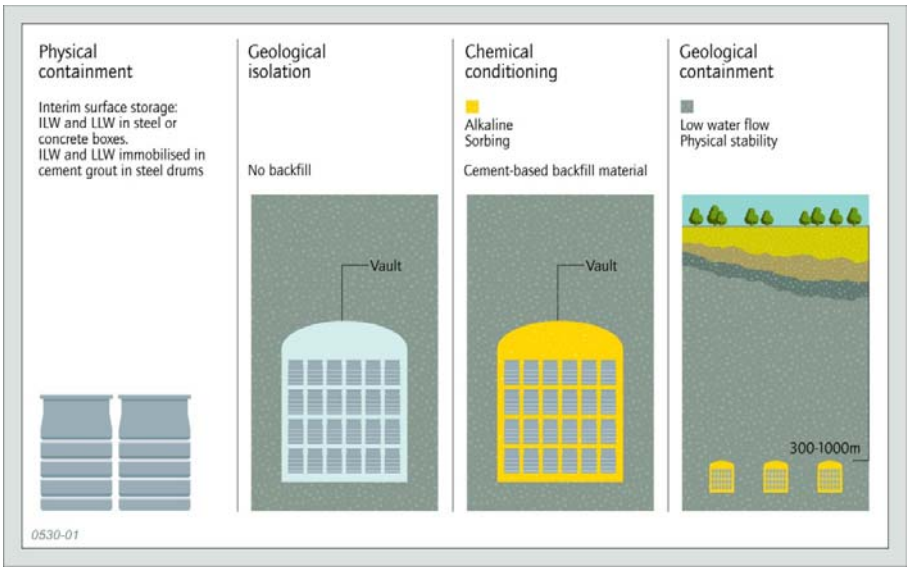

Figure 1 Concept for the disposal of ILW

A reference disposal concept for spent fuel is also used in the provision of disposability advice [11]. Under this concept, spent fuel would be over-packed into durable, corrosion-resistant disposal canisters manufactured from suitable materials, which would provide long-term containment for the radionuclides contained within the spent fuel. Although the canister material remains to be confirmed, the assessment has considered the potential performance of copper and steel canisters. In both cases, it is assumed that cast-iron inner vessel is used to hold and locate the spent fuel assemblies, and in the case of the copper canister would provide mechanical strength as well. These canisters would be emplaced in disposal holes lined with a buffer made from compacted bentonite, which swells following contact with water (Figure 2). This reference concept is based on the KBS-3V concept developed by SKB for disposal of spent fuel in Sweden [12].

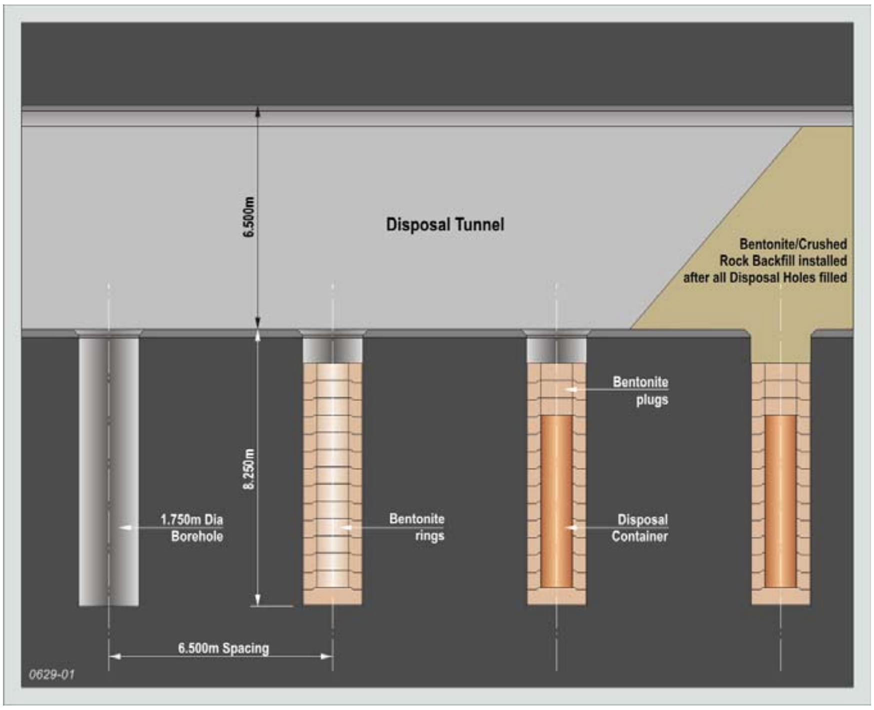

Figure 2 Concept for the disposal of spent fuel illustrating the disposal holes and emplacement of disposal canisters

2.2 Assessment Approach and Constraints

2.2.1 Approach followed for GDA Disposability Assessment

The GDA Disposability Assessment of the EPR was based on a protocol [5] agreed with the Environment Agency and the Nuclear Installations Inspectorate (NII), and was managed as a structured project using management procedures controlled under the RWMD Management System. These management procedures were based on those applied to assessments undertaken under the existing Letter of Compliance (LoC) process used by RWMD to provide guidance to plant operators on conditioning and packaging of wastes. An overview of the LoC Process is provided in Appendix A.

Assessment of the general disposability of the waste was based on work typically undertaken in the first stages of the LoC process including an independent review of the radionuclide and physical/chemical inventory of the ILW and spent fuel, and of the proposed package types and package numbers.

Conclusions have been drawn regarding the suitability of EdF/Areva proposals through comparison of information on EPR ILW and spent fuel with legacy wastes as follows:

- the key radionuclides and the quantities expected to arise as ILW and spent fuel have been compared to key radionuclides and their quantities in legacy wastes;

- the properties of proposed waste packages have been compared to the properties of UK standard packages, and initial views developed on further information requirements and issues that may need to be addressed in future LoC interactions.

Subsequent stages of the assessment considered the proposed waste packages and assessed performance using the approaches, safety assessments and "toolkits" developed for the LoC process. The application of the toolkits results in calculation of a series of quantitative performance measures, for example:

- estimates of dose rates, gas generation, loss or dispersal of radioactive contents (containment) under normal and accident conditions, and heat output during transport operations;
- estimates of risks to workers and the public owing to postulated accidents that release radioactivity from waste packages as a result of impact events and fires;
- for spent fuel, estimates of risks to humans from migration of radionuclides to the biosphere following closure of the GDF, with risks considered for the groundwater pathway.

The packaging proposals provided by EdF/Areva are preliminary in nature, and therefore, the results obtained through this assessment should be taken as indicative. Detailed specifications for some of the materials to be used in the EPR were not available to RWMD, and, therefore the assessment inventory has been supplemented by additional information based on assumptions regarding material composition made by RWMD. Where this has been the case, RWMD has adopted conservative or pessimistic assumptions and made this clear within the report.

2.2.2 GDA Disposability Assessment structure

The GDA Disposability Assessment was arranged in three stages, and with the work to be undertaken in each stage described in specific work instructions:

- Nature and Quantity Assessment;
- Disposal Facility Design Assessment;
- Safety, Environmental and Security Assessments.

Typical LoC assessments would also consider Data Recording and Quality Management System (QMS) issues. However, these were not considered in the GDA Disposability Assessment for the EPR at this stage and would need to be considered in any future LoC interactions.

The work undertaken in each stage is discussed below.

Nature and Quantity Assessment

The first stage in the process was a Nature and Quantity Assessment. For ILW, separate consideration was given to the wastes and "wasteform[11]". For spent fuel, separate consideration was given to the characteristics of the spent fuel assemblies and the disposal package characteristics. Work under this stage used information supplied by EdF/Areva, supplemented by additional information generated by RWMD. In particular, knowledge of the characteristics of radioactive waste and spent fuel arising at the Sizewell B pressurised water reactor (PWR) was used to add value to the GDA Disposability Assessment.

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[11] The wasteform is the term applied to the solid waste product following conditioning for long-term storage and disposal.

The Nature and Quantity evaluation was used to collate data on the operational and decommissioning ILW, and the spent fuel from the EPR, and to define reference cases for evaluation during the GDA Disposability Assessment. In particular, the objective of the Nature and Quantity evaluation was to establish a suitably detailed understanding of the radionuclide inventory, composition and quantity of ILW and spent fuel, and included:

- peer review of the submitted information;
- identification of any deficiencies and/or inconsistencies in the information;
- confirmation of waste volumes and waste package volumes for disposal.

The Nature and Quantity evaluation is presented in Section 3. This describes the characteristics of the ILW packages and spent fuel disposal packages and provides the basis for later stages of the assessment.

The Wasteform evaluation included:

- collation of information on proposed conditioning and packaging methods for ILW;
- development of an understanding of organic materials content, potential for gas generation and chemo-toxic content for ILW;
- description of geometry, material properties, and physical and chemical nature of the spent fuel.

The Wasteform evaluations for ILW and spent fuel are presented in Sections 4.1 and 5.2 respectively.

Disposal Facility Design Assessment

The second stage in the process was a Disposal Facility Design assessment. This stage comprised a Waste Package Performance evaluation and a Disposal Facility Design Impact evaluation.

The Waste Package Performance evaluation considered performance of waste packages under impact and fire accidents relevant to possible accident scenarios in transport of waste packages to a GDF and operations at a GDF, including estimation of release fractions for a range of standard impact and fire scenarios. In the GDA Disposability Assessment for the EPR, ILW package and spent fuel disposal canister release fractions have been developed for the ILW streams and spent fuel, and packaging scenarios proposed by EdF/Areva.

The Waste Package Performance evaluations for ILW and spent fuel are presented in Sections 4.1 and 5.2 respectively.

The Disposal Facility Design evaluation considered the implications on the design of a GDF. The evaluation considered the following:

- the footprint area needed to accommodate the ILW and spent fuel, in both a standalone facility and in a disposal facility also incorporating legacy wastes and spent fuel;
- compatibility of packaging assumptions with existing design assumptions;

- identification of unique or distinguishing features of the ILW and spent fuel and/or proposed ILW packages and spent fuel disposal packages;
- significance of potential variability in the proposed ILW packages and spent fuel disposal packages;
- consideration of the impact of new conditioning and packaging techniques.

The Disposal Facility Design evaluations for ILW and spent fuel are presented in Sections 4.2 and 5.3 respectively.

Safety, Environmental and Security Assessments

In the third stage of the process Safety, Environmental and Security assessments were undertaken. This included a Transport Safety assessment, Operational Safety assessment, Post-closure Safety assessment, Environmental evaluation, and a Security evaluation. The Safety, Environmental and Security Assessments considered the compatibility of operational and decommissioning ILW, and spent fuel from the EPR with existing assessments of RWMD reference disposal concepts. These assessments provide the basis for judging the potential disposability of EPR wastes and spent fuel

The Transport Safety assessment considered the logistics, regulatory compliance and risk of transport operations, with specific consideration of dose, gas generation, containment and heat output under normal and accident conditions. The Transport Safety assessment considered a set of bounding and representative waste streams, which were selected by RWMD based on the radioactivity of the waste packages and the type of container used for packaging. In addition, for waste packages assumed to be transported as Industrial Packages, the waste characteristics of all relevant waste streams were compared to international criteria for specification of low-specific activity material. The Transport Safety assessments for ILW and spent fuel are presented in Sections 4.2 and 5.3 respectively, which discuss issues related to the design and operation of the disposal system.

The Operational Safety assessment considered radiological dose and risk to workers and the public as a result of GDF operations. This included consideration of accidents, effects of gas generation and criticality. As with the Transport Safety assessment, the Operational Safety assessment considered a set of bounding and representative waste streams, which were selected by RWMD based on the radioactivity of the waste packages and the type of container used for packaging. The Operational Safety assessments for ILW and spent fuel are presented in Sections 4.2 and 5.3 respectively.

The Post-closure Safety assessment considered potential radiological impacts to humans and the environment in the long-term. Consideration was given to the groundwater and gas pathways, human intrusion and criticality, and environmental impacts due to chemotoxic species contained in the waste. The Post-closure Safety assessment for ILW was undertaken by comparison of each ILW stream with a similar ILW stream from Sizewell B. A similar comparison was made for spent fuel. In addition, post-closure safety for spent fuel was also assessed by quantitative calculation of risks to humans through the groundwater pathway. The Post-closure Safety assessments for ILW and spent fuel are presented in Sections 4.3 and 5.4 respectively.

The Environmental evaluation considered material usage in the GDF and commented on implications for non-radiological environmental impacts. The Environmental evaluation for ILW and spent fuel are presented in Sections 4.2 and 5.3 respectively.

The Security evaluation included estimation of the quantity of Nuclear Material (NM) contained in the ILW and spent fuel, determination of the likely security categorisation of the proposed ILW packages and spent fuel packages, and commentary on requirements for accountancy of the use of Nuclear Material. The Security evaluations for ILW and spent fuel are presented in Sections 4.2 and 5.3 respectively.

3 EPR OPERATION, WASTES, PACKAGING PROPOSALS AND WASTE PACKAGE CHARACTERISTICS

This section provides a summary of the information used in the GDA Disposability Assessment for the EPR. RWMD used the information supplied by EdF/Areva, supplemented as necessary by information available to RWMD, to provide a comprehensive dataset of information covering waste package numbers, inventories and characteristics when conditioned and packaged.

This section contains the following information:

- summary description of an EPR (Section 3.1);
- assumptions regarding the operation of an EPR (Section 3.2);
- description of the higher activity radioactive waste streams and spent fuel that will be generated through operation and decommissioning of an EPR (the ‘assessment inventory’), including volumes, assumptions regarding the packaging of these wastes and estimates of waste package numbers and their characteristics (Section 3.3 and Section 3.4).

In order to place the description of EPR wastes in context, the expected ILW and spent fuel arisings are compared to the reported arisings from Sizewell B PWR [13,14].

The implications of the waste volumes, package numbers and activities presented in this section are discussed in Sections 4 and 5.

3.1 Summary of EPR Design and Operation

The EPR is an evolutionary PWR design with a rated thermal power of 4500 MW and an electrical power output of approximately 1600-1660 MW(e), depending on site-specific factors.

The EPR evolutionary design is based on experience from operation of Light Water Reactors (LWR) worldwide, primarily those incorporating the most recent technologies: the N4 and KONVOI reactors currently in operation in France and Germany respectively. The primary system design, loop configuration, and main components are similar to those of currently operating PWRs.

In PWRs such as the EPR, ordinary (light) water is utilised to remove the heat produced inside the reactor core by thermal nuclear fission. This water also ‘thermalises’ or moderates, neutrons in a manner necessary to sustain the nuclear fission reaction. The heat produced inside the reactor core is transferred to the turbine through the steam generators. Only heat is exchanged between the reactor cooling circuit (primary circuit) and the steam circuit used to feed the turbine (secondary circuit). No exchange of cooling water takes place.

The EPR design is furnished with a four-loop, pressurised water ‘reactor coolant system’ composed of a reactor vessel that contains the fuel assemblies, a pressuriser including control systems to maintain system pressure, one reactor coolant pump per loop, one steam generator per loop, associated piping, and related control and protection systems (Figure 3). This equipment would be standardised for all EPRs.

In the reactor coolant system, the primary cooling water is pumped through the reactor core and the tubes inside the steam generators, in four parallel closed loops, by four reactor coolant pumps powered by electric motors. The reactor operating pressure and temperature are such that the cooling water does not boil in the primary circuit but remains in the liquid state, increasing its cooling effectiveness. A pressuriser, connected to one of the coolant loops is used to control the pressure in the reactor coolant system. Feed-water entering the secondary side of the steam generators absorbs the heat transferred from the primary side and evaporates to produce saturated steam. The steam is dried inside the steam generators then delivered to the turbine. After exiting the turbine, the steam is condensed and returned as feed water to the steam generators.

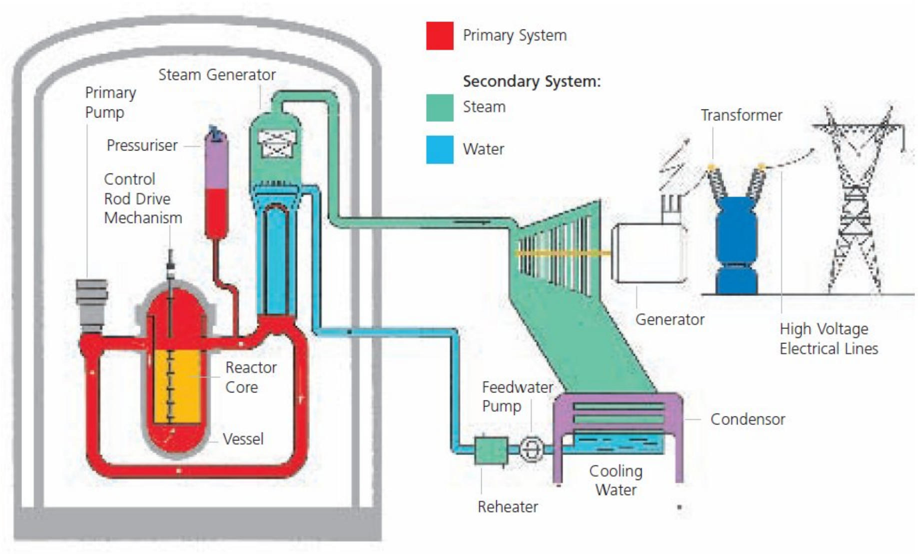

Figure 3 Principal components of an EPR. Figure reproduced from [15]

3.2 Assumptions

The GDA Disposability Assessment for the EPR was based on the following assumptions:

- The EPR would be operated for 60 years. During the operation of the reactor, fuel assemblies would be periodically rotated within the reactor core, and then removed and replaced with other fuel assemblies. Ninety spent fuel assemblies would be removed from the reactor every 18 months during planned shutdown periods and require storage.

- The date at which operation of power production from an EPR would commence in the UK is uncertain. Government and industry estimates suggest that the first reactor could be operational by 2017. In the GDA Disposability Assessment for the EPR, estimates of time-dependent properties, e.g. those related to radioactive decay, are assessed from the time of generation of the waste. In discussion of the implications for management of radioactive waste, RWMD has assumed a start date for a single reactor of 2020.

- Spent fuel characteristics have been determined on the assumption that the reactor would be operated to achieve a maximum fuel assembly irradiation (burn-up)[12] of

[12] The fuel assembly average irradiation (burn-up) represents the total irradiation associated with all the fissile material in an assembly divided by the initial mass of uranium in the assembly. It takes into account the variation in irradiation both axially along a fuel rod and the variation from one fuel rod to another. For simplicity, whenever fuel irradiation or burn-up is referred to in the remainder of the

65 GWd/tU. In the absence of data to the contrary, the GDA Disposability Assessment has assumed that all fuel will be irradiated to the maximum fuel assembly burn-up. This is a conservative approach and ensures that the conclusions from the assessment are bounding for a wide range of possible operational behaviours.

- The fuel used in the EPR would be manufactured from freshly mined uranium enriched to an initial U-235 content of 5%[13].

- It is assumed that ILW and spent fuel from the EPR will arrive at the GDF in a packaged state, ready for disposal.

3.3 ILW Streams, Packaging Assumptions, Waste Package Numbers and Characteristics

3.3.1 Operational ILW streams and packaging assumptions

EdF/Areva has indicated that six operational ILW streams could potentially arise from normal operation of an EPR:

- Ion exchange resins – organic resins that arise from the clean-up of primary circuit water and water from the Liquid Waste and spent fuel Pit Treatment Systems;

- Spent cartridge filters (ILW) – filters from the clean-up of primary circuit water and water from the Liquid Waste and spent fuel Pit Treatment Systems. The filters consist of a stainless steel support, with a glass fibre or organic filter media;

- Spent cartridge filters (LLW and ILW)[14] – filters, similar to, but typically smaller in size than spent cartridge filters (ILW);

- Operational wastes >2mSv/hr – a range of materials, including contaminated metal, plastics, cloth, glassware and rubble, arising from operations during planned shutdown periods (hence ‘operational wastes’);

- Wet sludges – sludges arising from cleaning the bottoms of liquid waste treatment tanks and various sumps;

- Evaporator concentrates – residues from the evaporation of waste water.

The raw waste volumes of these materials as determined by EdF/Areva are provided in Table 1.

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report what is meant is fuel assembly average irradiation or burn-up. Thus, the statement that the maximum fuel assembly burn-up is 65 GWd/tU means that the highest fuel assembly average burn-up will be 65 GWd/tU.

[13] Freshly-mined uranium may be contrasted with reprocessed uranium. The latter potentially contains significant quantities of U-236, which is a pre-cursor to Pu-238 and therefore can adversely affect the heat output of spent fuel. It is currently assumed that reprocessed uranium would not be used for manufacturing EPR fuel. Any change to this assumption would require further assessment.

[14] Some items included in this waste stream might be able to be categorised as LLW but, conservatively, all wastes within this stream are being considered as potentially requiring disposal as higher activity waste.

The GDA Disposability Assessment for the EPR considered three scenarios for conditioning and packaging of operational ILW arising over 60 years, referred to as the Reference Case, Variant Case 1 and Variant Case 2. Waste stream identifiers for each scenario are specified in Table 1.

Table 1 Total lifetime raw waste volumes for operational ILW from an EPR and identifiers used for different management scenarios

Waste Stream	Identifier Reference Case	Identifier Variant Case 1	Identifier Variant Case 2	Raw Waste Volume (m^3)
Ion exchange resin	EPR01	EPR11	EPR21	180
Spent cartridge filters ILW	EPR02	EPR12	EPR22	150
Spent cartridge filters (LLW and ILW)	EPR03	EPR13	EPR23	150
Operational waste >2mSv per hr	EPR04	EPR14	EPR24	60
Wet sludges	EPR05	EPR15	EPR25	60
Evaporator concentrates	NA (see text for explanation)	EPR16	EPR26	60

Reference Case

The Reference Case assumed that operational ILW would be conditioned and treated using the same procedures as applied during the operation of existing PWRs in France. The submission assumed that similar waste management practices could be integrated into UK regimes in an acceptable manner.

Two types of cylindrical pre-cast concrete casks, designated C1 and C4, were assumed in the Reference Case (Figure 4). Both of these casks can include internal mild steel shielding of flexible thickness (40-100 mm of shielding was assumed for the GDA Disposability Assessment) to provide shielding against different concentrations of gamma emitting radionuclides. The C1 Cask is 1.4 m in diameter, 1.3 m high, and has a 0.15 m thick concrete shield wall. The C4 Cask has the same dimensions apart from the diameter; it is 1.1 m in diameter. The C1 and C4 Casks are assumed to be used as Industrial Package Type 2 (IP-2) transport packages as defined by IAEA Transport Regulations.

In the Reference Case scenario, the operational ILW would be immobilised using epoxy resin (EPR01), or cement grout (EPR02, EPR03, EPR04 and EPR05). The range of wastes comprising EPR04 would be placed into plastic bags and compressed to reduce volume before grouting. The EdF/Areva submission assumed that, in the Reference Case, evaporator concentrates would be incinerated leaving no radioactive residue, which is the current practice in France. The Reference Case identifier for Evaporator Concentrates in Table 1 is therefore ‘NA’. The absence of radioactive residue following this practice will need to be confirmed in future.

Figure 4 Illustration of the C1 and C4 concrete casks proposed for packaging of operational ILW in the EPR Reference Case

Variant Case 1

For Variant Case 1, it is assumed that EPR operational ILW (waste streams EPR11-EPR16) will be packaged in 200 litre Drums, which would subsequently be placed in UK standard stainless steel 500 litre Drums with an annular grout lining cast into place during packaging and assumed to be 100 mm thick (Figure 5). Evaporator concentrates are assumed to be packaged and disposed of rather than incinerated. Waste streams EPR11 (Ion exchange resins), EPR14 (Operational wastes), EPR15 (Wet sludges) and EPR16 (Evaporator concentrates) would be dried and packaged directly in the 200 litre drums (i.e. without immobilisation). Waste streams EPR12 (Spent cartridge filters, ILW) and EPR13 (Spent cartridge filters, LLW and ILW) would be grouted into the 200 litre drums. For transport, the 500 litre Drums would be placed, in groups of four, inside a Standard Waste Transport Container with 285 mm of shielding (SWTC-285) [16] for transport as Type B transport packages. The SWTC-285 has been designed with a cavity size and shape suitable for transport of four 500 litre Drums using a handling frame.

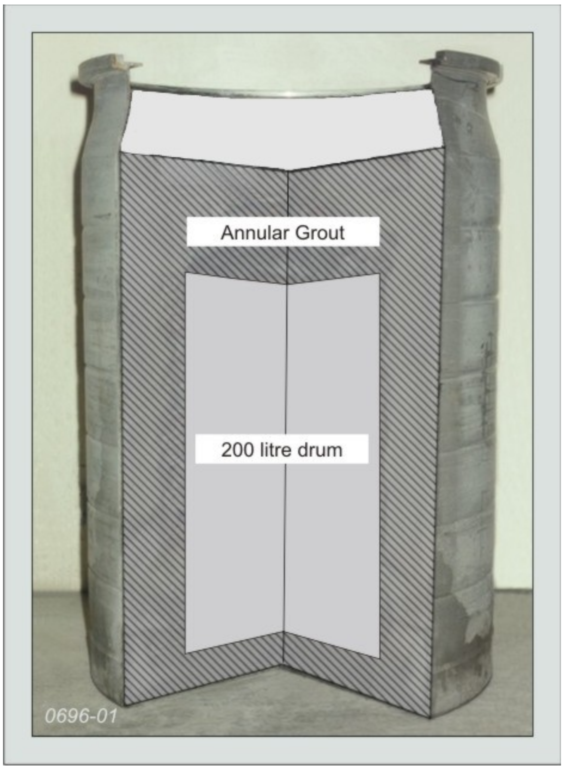

Figure 5 Illustration of 200 litre drum cement grouted within a 500 litre Drum as proposed for packaging of operational ILW in the EPR Variant Case 1

Variant Case 2

For Variant Case 2, the waste is assumed to be packaged in cylindrical cast-iron casks. Containers of this type, for example MOSAIK Casks, are approved and are currently used for the packaging of operational waste in Germany (Figure 6). These casks are made from Ductile Cast Iron, and have dimensions of 1.06m (diameter) by 1.5m (height) and have walls, base and lid thicknesses of 0.16m. The cast-iron casks may be used as either Industrial Package Type 2 (IP-2) or Type B[15] transport container, the latter requiring suitable over-packing arrangements to ensure appropriate performance. For reasons of efficiency, this assessment has assumed that Type B arrangements would be used to ensure optimum waste loading. It is recognised further development work would be required to confirm the appropriateness of this assumption.

The EdF/Areva submission assumed that waste streams EPR21 (Ion exchange resins), EPR24 (Operational wastes), EPR25 (Wet sludges) and EPR26 (Evaporator concentrates) would be dried and packaged directly in the cast-iron casks without conditioning. No assumptions were provided for the packaging of EPR22 (Spent cartridge filters, ILW) and EPR23 (Spent cartridge filters, LLW and ILW), and RWMD assumed that both types of Spent cartridge filters would be conditioned in the same manner as for Variant Case 1, i.e. conditioned by grouting within the cast-iron casks.

[15] IP-2 and Type B – transport package categories defined by IAEA Transport Regulations

Figure 6 Illustration of a cast-iron cask as proposed for packaging of operational ILW in the EPR Variant Case 2

3.3.2 Decommissioning ILW streams and packaging assumptions

The reference decommissioning assumption advised by EdF/Areva is that transport of decommissioning waste occurs 40 years after reactor shutdown. Inventory calculations have been undertaken in line with this assumption. With such a delay, EdF/Areva has assumed that even the highest specific activity bioshield concrete will have decayed to LLW, that any resins from a final decontamination of the primary circuit will also be LLW, and that these materials will be suitable for disposal to a LLW repository.

Although it is asserted that all concrete would be LLW after 40 years storage, this remains to be proven. Nevertheless, given the compact nature of an EPR, RWMD estimates that the volume of any such ILW concrete is unlikely to exceed 100m^3, and would be unlikely, therefore, to raise significant issues for disposability.

All other ILW produced prior to Stage 3 decommissioning would be managed as operational ILW and, for the purposes of this assessment, has been assumed to be encompassed by the operational ILW described above. This would include any wastes generated during early decommissioning, i.e. immediately after the reactor shut-down (Stage 1), and prior to Care and Maintenance (Stage 2).

Decommissioning ILW would consist of three waste streams (there are no variant packaging assumptions for decommissioning ILW) and would be packaged as follows:

- EPR06 (Reactor Vessel), which consists of ferritic steel associated with the mid-height section of the pressure vessel and from the vessel cladding. The pressure vessel steel will be in the form of thick (~0.2m) curved steel plate, possibly with its stainless steel cladding, typically a few mm thickness, still attached. These wastes would be grouted into 4 metre Boxes with a 100 mm concrete wall (Figure 7) and would be transported as IP-2 transport packages.

- EPR07 (Upper and Lower Reactor Internals), which consists of low cobalt stainless steel in the form of plates with thickness of the order of 0.01m. These wastes would be grouted into 3m^3 Boxes (Figure 7); and would be transported in a standard waste transport container (SWTC) as Type B transport packages.
- EPR08 (Lower Reactor Internals including Heavy Shield), which consists of two similar grades of stainless steel: a low cobalt grade steel used for all the components closest to the core that receive the highest irradiation, and a grade containing higher concentrations of cobalt used for the more distant components. These steels are expected to have plate-like structures with thickness of the order of 0.01m. These wastes would be grouted into 3m^3 Boxes (Figure 7); and would be transported in a SWTC as Type B transport packages.

Raw waste volumes for decommissioning ILW are presented in Table 2.

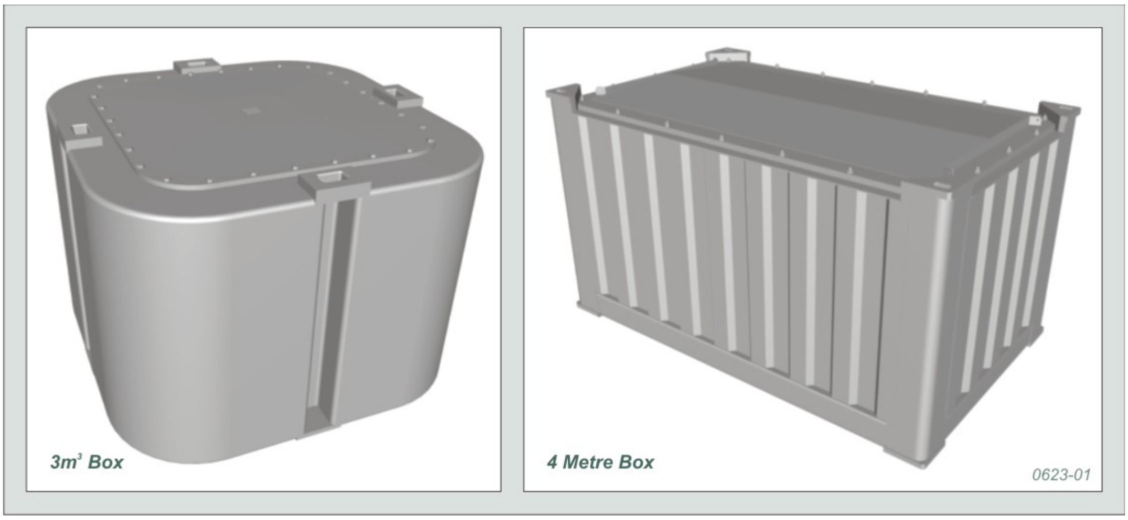

Figure 7 Illustration of a 3m^3 Box and a 4metre Box as proposed for packaging of decommissioning ILW from the EPR

Table 2 Raw waste volumes for decommissioning ILW from an EPR and identifiers used

Waste Stream	Waste Stream Identifier	Raw Waste Volume (m^3)
Reactor Vessel	EPR06	23
Upper and Lower Reactor Internals	EPR07	10
Lower Reactor Internals Including Heavy Shield	EPR08	18

3.3.3 ILW package numbers and characteristics

The information supplied by EdF/Areva on the radionuclide inventories of the identified wastes has been used to derive assessment inventories for the various proposed waste packages, including variants for operational ILW (see Table 3 for supplied scaling factors used by EdF/Areva to derive radionuclide inventories for operational wastes). In some cases, to ensure full coverage of potentially significant radionuclides, it has been necessary to supplement the information supplied by EdF/Areva with information available to RWMD. The assessment inventories are intended to characterise the range of waste package inventories, taking account of uncertainties and variability between packages.

In support of this GDA Disposability Assessment, the assessment inventory defined:

- best estimate (average) waste package inventory. This inventory when taken with the number of waste packages defines the total inventory associate with the waste stream. This is particularly relevant to the post-closure assessment and some aspects of operational safety assessment;

- bounding (maximum) waste package inventory. This is used for transport safety and certain aspects of the operational safety assessment where individual waste packages are considered.

The EPR ILW waste package radionuclide-related parameters and waste quantities (package numbers and total packaged volume) are given in Table 4 to Table 6 and Table 9. Radionuclide related parameters (e.g. dose rate) are calculated at the time of arising (i.e. zero-decayed for operational ILW and 40 year decayed for decommissioning ILW appropriate to the assumed times if transport). The fissile content of waste is not included in the summary tables as it is estimated to be well below the 15g fissile exception level for non-fissile transport packages.

Nature and Quantity of Operational ILW

For operational ILW (Table 4 to Table 6), information on the raw waste volumes, package types, package numbers and radionuclide content have been derived from consideration of information on waste packages from existing PWRs provided by EdF/Areva and enhanced as described below by RWMD. The radionuclide related data presented in the body of Tables 4 to Table 6 relate to average waste package inventories. In addition footnotes to these tables present approximate multiplying factors that relate maximum to average waste package properties. These factors come from a statistical analysis of the measured activity content of waste packages arising from European PWRs. For waste streams EPR01 to EPR04 the number of packages on which the statistical analysis was performed varied between 257 and 1308 and hence there is good confidence that the characteristics of the maximum packages are well founded.

The volumes of waste are based on experience of operating PWRs in France and Germany. Information from 2001-2003 was complemented by an analysis of 50,000 waste packages produced in 2005-2007 from 25 reactors (20 reactors operating at 1,300 MW(e) and five reactors operating at 1,500 MW(e)). The waste from these PWRs is considered by EdF/Areva to be representative of the waste from the EPR.

The waste package numbers developed by EdF/Areva are based on waste loadings used during existing operations. EdF/Areva's assumptions regarding waste loadings were accepted and applied during the N&Q evaluation.

Information provided by EdF/Areva on radionuclide concentrations in the waste was based on different approaches for different radionuclides:

- estimates of the concentrations of short-lived beta/gamma emitting radionuclides were based on measurements by gamma spectrometry of waste from existing PWRs;
- concentrations of long-lived beta, beta/gamma and alpha emitting radionuclides were estimated using scaling factors linked to "easy-to-measure" radionuclides (Co-60 and Cs-137), which is a standard practice applied in the nuclear power industry [17]).

The information on the scaling factors provided by EdF/Areva (Table 3 below) showed that an extensive set of the long-lived fission and activation products of relevance to post-closure safety was being considered (grey cells in Table 3 signify that the scaling factors are obtained from literature). In addition a scaling factor relating Co-60 to total long-lived alpha activity was also presented that provided a means to ensure that all the actinides of significance to post-closure safety could also be assessed. Also evident from the data provided by EDF/Areva is that key shorter-lived radionuclides of significance to operational and transport safety such as Co-60, Sr-90 and Cs-137 are also covered.

Table 3 Scaling factors provided by EdF/Areva

Ion Exchange Resins			Water Filters & Others		
Nuclides	**Key**	**SF 1999**	**Nuclides**	**Key**	**SF 1999**
Be-10	Co-60	**2 E-07**	Be10	Co-60	**2 E-07**
C-14	Co-60	**1.8 E-02**	C-14	Co-60	**1.1 E-02**
Cl-36	Co-60	**1 E-05**	Cl-36	Co-60	**1 E-06**
Ca-41	Co-60	**5 E-06**	Ca-41	Co-60	**5 E-06**
Fe-55	Co-60	**1.4 E-01**	Fe-55	Co-60	**2.1 E+00**
Ni-59	Co-60	**1.1 E-03**	Ni-59	Co-60	**5.3 E-04**
Ni-63	Co-60	**1.4 E+00**	Ni-63	Co-60	**2.3 E-01**
Se-79	Cs-137	**4 E-06**	Se-79	Cs-137	**4 E-06**
Sr-90	Cs-137	**2.3 E-03**	Sr-90	Co-60	**2.6 E-02**
Mo-93	Co-60	**1 E-06**	Mo-93	Co-60	**1 E-06**
Zr-93	Co-60	**5 E-07**	Zr-93	Co-60	**5 E-05**
Nb-94	Co-60	**1.2 E-04**	Nb-94	Co-60	**1.3 E-04**
Tc-99	Cs-137	**1 E-05**	Tc-99	Cs-137	**4.2 E-04**
Pd-107	Cs-137	**1 E-07**	Pd-107	Cs-137	**1 E-05**
Ag-108m	Co-60	**1 E-03**	Ag-108m	Co-60	**1 E-03**
Sn-121^{m}	Cs-137	**2 E-05**	Sn-121m	Cs-137	**2 E-05**
Sn-126	Cs-137	**9 E-06**	Sn-126	Cs-137	**9 E-06**
I-129	Cs-137	**1 E-06**	I-129	Cs-137	**1 E-06**
Cs-135	Cs-137	**5 E-06**	Cs-135	Cs-137	**3 E-06**
Sm-151	Cs-137	**7 E-04**	Sm-151	Cs-137	**4 E-03**

The alpha activity of operational ILW from an EPR is expected to be low in an EPR, but would be affected by in-service fuel cladding failure. Should the cladding fail, actinides could contaminate the pressurised water circulating in the primary circuit. These actinides would be transferred to the resins and filters used to decontaminate the coolant.

Currently, EdF practice is to declare individual actinides or alpha emitting radionuclides in their operational waste only in the event that "serious fuel cladding failures" were found to have occurred. No data on the frequency of serious fuel cladding failures or the proportion of operational waste containing such alpha contamination were available. Hence, to ensure conservatism in the assessment inventories applied to all operational waste, the inventories of actinides and alpha emitting radionuclides in these wastes were ascribed values typical of those that would be seen during reactor operation with serious fuel cladding failures. In practice, only a small proportion of such wastes would be so contaminated.

The concentration of alpha emitters as the result of serious clad failure was estimated by RWMD based on the concentration of Co-60 and the ratio of total alpha activity to Co-60 activity measured in the primary circuit liquid of existing PWRs as discussed in [18]. The ratio is $3x10^{-3}$. The amounts of individual alpha emitting radionuclides were derived using a generic one-year cooled spent PWR fuel inventory associated with a 4.2% initially enriched fuel that had experienced a burn-up of 55 GWd/tU. The fuel inventory is different to the 65 GWd/tU maximum burn-up inventory used in spent fuel assessment. However, given the limited concentrations of actinides in these wastes, the difference in fuel inventory assumptions is not considered to be significant. This generic spent fuel inventory was also used to obtain activity estimates from beta/gamma emitting actinides such as Pu-241 and Am-242m.

Estimates of many waste package radionuclide properties (for example activity and heat output) were calculated using the DIQuest code developed by RWMD. The code is based on Microsoft Access 2002 and an SQL server. It has been developed and verified through a structured process [19, 20, 21] and its use is described in a comprehensive User Guide [22]. The code has been used in various ways in the EPR Disposability Assessment:

- waste stream activity data have been imported as a stock present at a defined cooling time which has then been decayed for a subsequent cooling period to generate a heat output or waste package dose rate versus cooling time function;
- waste stream activity data have been entered as a uniform arising over the 60 year operational lifetime of the reactor to allow a total waste stream inventory to be generated at a fixed number of years following final reactor shutdown;
- the DIQuest Nature and Quantity Summary Sheet report was run to generate the standard set of waste package radionuclide related parameters that are considered by RWMD in disposability assessments with the exception of dose rates, as presented in Section 2 of Part 2 of this report.

For the standard waste package types that have been developed in the UK, DIQuest is also able to estimate dose rates both outside bare packages and outside the transport containers used for the packages. This DIQuest capability was used to derive the dose rates associated with the Variant Case 1 operational and decommissioning waste packages. Hence all the radionuclide related properties including dose rates could be obtained directly from DIQuest for the Variant Case 1 operational and decommissioning waste packaging assumptions.

As noted in Section 3.3.1, EdF/Areva has proposed the use of containers that have not previously been considered in the UK that is the C1 and C4 concrete casks for the Reference Case and cast-iron casks for Variant Case 2. Estimation of package properties for the wastes assigned to these non-standard package types applied a two-stage process involving use of a surrogate DIQuest waste package with similar radiation shielding properties to the non-standard packages. The surrogate waste package used to represent both the Reference Case and Variant Case 2 waste packages was the 2 metre Box with 200 mm of internal concrete shielding.

In stage one of the process the activity content of each non-standard package (i.e. a different inventory for EPR01, EPR02 etc) was assigned to its surrogate waste package within DIQuest and a DIQuest N&Q Summary Sheet report was run. This yielded all the appropriate waste package radionuclide data but the dose rate information was relevant to the surrogate rather than the actual non-standard package.

In stage two the dose rate information relevant to the surrogate package was corrected using the results of detailed gamma shielding calculations performed with RANKERN for the non-standard packages. The calculations with RANKERN were undertaken for waste streams EPR01 and EPR02 and used to develop dose rate correction factors to be applied to the results for EPR03, EPR04 and EPR05 coming from stage one of the process. As the source geometry and total gamma attenuation through the walls of the cast-iron cask are similar to that of the C1 Concrete package equipped with a 100mm internal mild steel shield, the correction factors were also applied to waste streams from Variant Case 2.

The radionuclide concentrations were checked during the process of undertaking the calculations with DIQuest and it has been found that the waste stream activities of EPR14, EPR15, EPR24 and EPR25 given by EdF/Areva were typically 25 to 40% less than equivalent reference case stream for the Reference Case (EPR04 and EPR05). To be conservative, it has been assumed that the Reference Case data were correct and the waste package inventories for the above waste streams increased by the amount necessary to ensure that total waste stream activities were the same as given for the Reference Case.

In addition, the datasheets provided by EDF/Areva for EPR16 and EPR15 were identical to each other. Since it was necessary to make the 25% - 40% activity enhancement of EPR15, RWMD decided that an identical enhancement should be made to EPR16 to maintain the equality of the submission for EPR15 and EPR16. Since EPR26 is an alternative packaging arrangements for EPR16, RWMD also decided to apply the same enhancement factor to EPR25 and EPR26.

Table 4 EPR waste stream data: operational ILW Reference Case(1) (2)

Waste Stream	Package Type	Number of Packages	Total Packaged Waste Volume (m^3)	Average Package Alpha Activity (TBq)	Average Package Beta/ Gamma Activity (TBq)	Average Package A_2 Content	Average Package Heat Output (Watts)	Average Package Dose Rate at 1m from Package (mSv/hr)
EPR01	Concrete C1	450	900	4.00E-04	1.48E-01	3.92E-01	2.71E-02	3.94E-01
EPR02	Concrete C1	360	720	1.23E-03	4.81E-01	1.12E+00	7.87E-02	8.27E-01
EPR03	Concrete C4	600	741	2.40E-04	9.39E-02	2.18E-01	1.54E-02	1.66E-01
EPR04	Concrete C1	180	360	5.87E-05	2.07E-02	5.00E-02	3.13E-03	3.24E-02
EPR05	Concrete C1	240	480	1.19E-05	4.62E-03	1.02E-02	6.39E-04	6.19E-03
TOTALS		1830	3201					

Notes:

(1) The values are for average waste package inventories.

(2) Radionuclide data for the maximum package may be obtained as M times the average package data where approximately M=12 for EPR01 & EPR02, M=10 for EPR03 & EPR05 and M=7 for EPR04 (see discussion at the beginning of Section 3.3.3 for the basis of the M factors).

Table 5 EPR waste stream data: operational ILW Variant Case 1[(1)] [(2)]

Waste Stream	Package Type	Number of Packages	Total Packaged Waste Volume (m^3)	Average Package Alpha Activity (TBq)	Average Package Beta/ Gamma Activity (TBq)	Average Package A_2 Content	Average Package Heat Output (Watts)	Average Package Dose Rate at 1m from Package (mSv/hr)[(3)]
EPR11	500 litre Drum	948	550	1.90E-04	7.02E-02	1.86E-01	1.29E-02	7.03E-05
EPR12	500 litre Drum	360	209	1.23E-03	4.81E-01	1.12E+00	7.87E-02	3.60E-04
EPR13	500 litre Drum	600	348	2.40E-04	9.39E-02	2.18E-01	1.54E-02	7.02E-05
EPR14	500 litre Drum	300	174	3.52E-05	1.24E-02	3.00E-02	1.88E-03	1.43E-05
EPR15	500 litre Drum	316	183	9.06E-06	3.51E-03	7.74E-03	4.85E-04	2.93E-06
EPR16	500 litre Drum	316	183	9.06E-06	3.51E-03	7.74E-03	4.85E-04	2.93E-06
TOTALS		2840	1647					

Notes:

(1) The values are for average waste package inventories.

(2) Radionuclide data for the maximum package may be obtained as M times the average package data where approximately M=12 for EPR11 & EPR12, M=10 for EPR13, EPR15 & EPR16 and M=7 for EPR14 (see discussion at the beginning of Section 3.3.3 for the basis of the M factors)

(3) Dose rate 1m outside an SWTC-285 containing 4 x 500 litre Drums

Table 6 EPR waste stream data: operational ILW Variant Case 2[(1)] [(2)]

Waste Stream	Package Type	Number of Packages	Total Packaged Waste Volume (m^3)	Average Package Alpha Activity (TBq)	Average Package Beta/ Gamma Activity (TBq)	Average Package A_2 Content	Average Package Heat Output (Watts)	Average Package Dose Rate at 1m from Package (mSv/hr)
EPR21	Cast-iron casks	383	507	4.70E-04	1.74E-01	4.61E-01	3.19E-02	7.02E-02
EPR22	Cast-iron casks	360	477	1.23E-03	4.81E-01	1.12E+00	7.87E-02	9.35E-02
EPR23	Cast-iron casks	600	794	2.40E-04	9.39E-02	2.18E-01	1.54E-02	1.88E-02
EPR24	Cast-iron casks	128	169	8.28E-05	2.92E-02	7.05E-02	4.42E-03	1.29E-02
EPR25	Cast-iron casks	128	169	2.24E-05	8.69E-03	1.91E-02	1.20E-03	2.48E-03
EPR26	Cast-iron casks	128	169	2.24E-05	8.69E-03	1.91E-02	1.20E-03	2.48E-03
TOTALS		1727	2285					

Notes:

(1) The values are for average waste package inventories.

(2) Radionuclide data for the maximum package may be obtained as M times the average package data where approximately M=12 for EPR21 & EPR22, M=10 for EPR23, EPR25 & EPR26 and M=7 for EPR24 (see discussion at the beginning of Section 3.3.3 for the basis of the M factors).

Nature and Quantity of Decommissioning ILW

Estimates of the quantities and characteristics of decommissioning ILW (Table 9) have been developed based on modelling of the neutron flux, power history and material composition data for the core of an EPR reactor. The activation calculations used the highest total flux experienced by each component to derive its total inventory. Therefore, the inventories for EPR decommissioning ILW considered to be upper bound estimates (maximum package inventories).

The approach adopted by EdF/Areva to develop an estimate of the quantities and characteristics of EPR Decommissioning ILW is as follows:

- perform neutron transport studies (the computer code used was not specified in the submission) to generate three energy group neutron flux data within the primary circuit structures and surrounding bioshield
- use the neutron flux, power history and material composition data to derive specific activity data using an activation analysis tool known as DARWIN-PEPIN 2.1.1
- where appropriate to calculate dose rates outside proposed waste package systems with the code MERCURE 5.3

EdF/Areva did not include any Bioshield Concrete in its Decommissioning ILW waste streams. The chemical composition of the bioshield concrete analysed in the submission meant that after 40 years' decay it could all be declared as LLW. The composition of bioshield concrete is potentially variable depending upon the aggregate used in its manufacture. However, given the compact nature of a PWR, even if the concrete adjacent to the inner wall of the bioshield were found to be ILW the volume of this waste is unlikely to exceed of the order of $100m^3$. Therefore, the absence of any Bioshield Concrete in the Decommissioning ILW streams from the EPR is not viewed as a significant omission from the submission.

With regard to irradiation assumptions, the activation calculations performed by EdF/Areva for the EPR Decommissioning ILW used the highest total flux experienced by the component to derive its total inventory. For example, although the total neutron flux in the heavy reflector (EPR08) at mid core height varies between 3.9×10^{13} and 2.4×10^{14} $n/cm^2/s$, the total activity of this component was calculated on the basis that all the material was exposed to a flux of 2.4×10^{14} $n/cm^2/s$. Therefore, the estimates of activity for EPR Decommissioning ILW are upper bound estimates.

To properly correct for this conservatism it would be necessary to know the volumes of waste exposed to each flux level, a degree of detail that is not yet available to RWMD. Therefore, the datasheets that were generated for EPR Decommissioning ILW use the same value for average and maximum package inventories, which introduces conservatism into the assessment of these wastes.

Although the information provided by EdF/Areva was sufficient to establish waste classification and short-term radiological characteristics, such as dose rate and heat output, EdF/Areva's analysis did not consider the impact of many trace elements that may be present in the steel. Therefore, the quantities of many post-closure significant radionuclides were not included in the submission and the data were therefore enhanced by RWMD.

Of the three Decommissioning ILW streams, EPR08 is expected to have the highest activity because the material in this waste stream is exposed to the highest neutron flux. Therefore, a detailed enhancement of the radionuclide quantities in this waste stream was undertaken using FISPACT-2007, and quantities for EPR06 and EPR07 were approximated using a scaling factor developed from the analysis of EPR08.

The major element composition of the steel used in the heavy reflector (EPR08) and reported in the submission was extended by RWMD to include a full range of elements. The extended composition was based on the known composition of Type 304 stainless steel used in many PWRs in the US, which has a similar major element composition to that reported by EdF/Areva for the EPR components. For elements without information, crustal abundances were used. The assumed composition is provided in Table 7.

Table 7 Upper bound elemental concentration data applied for stainless steel type Z2 CN 19-10 + N2 in FISPACT activation calculations undertaken by RWMD for EPR08. Values shown with an orange background were taken from the submission document provided by EdF/Areva and values in yellow were assumed by RWMD

Elemental concentration expressed in weight fraction									
H	7.00E-05	K	6.43E-05	Kr	3.30E-06	Xe	3.95E-07	Hf	9.44E-07
Li	1.66E-06	Ca	4.02E-05	Rb	1.21E-05	Cs	4.80E-07	Ta	1.12E-06
Be	6.00E-04	Sc	1.80E-07	Sr	1.05E-05	Ba	7.75E-04	W	4.63E-04
B	7.45E-05	Ti	1.29E-03	Y	8.09E-06	La	7.02E-07	Re	1.75E-07
C	3.50E-04	V	8.77E-04	Zr	1.58E-05	Ce	7.74E-04	Os	4.45E-08
N	8.00E-04	Cr	2.00E-01	Nb	2.20E-04	Pr	5.50E-04	Ir	1.62E-06
O	1.50E-03	Mn	2.00E-02	Mo	5.34E-03	Nd	1.75E-06	Pt	5.00E-07
F	1.00E-03	Fe	7.25E-01	Ru	1.00E-07	Sm	4.02E-07	Au	5.00E-07
Ne	1.27E-05	Co	6.00E-04	Rh	2.56E-05	Eu	7.66E-08	Hg	5.00E-05
Na	2.99E-05	Ni	1.00E-01	Pd	1.03E-06	Gd	4.92E-07	Tl	6.00E-05
Mg	1.00E-03	Cu	1.00E-02	Ag	3.43E-06	Tb	4.72E-06	Pb	1.69E-04
Al	2.42E-04	Zn	1.46E-03	Cd	1.07E-05	Dy	5.60E-08	Bi	1.56E-07
Si	1.00E-02	Ga	4.27E-04	In	1.49E-07	Ho	3.78E-08	Th	6.09E-06
P	3.00E-04	Ge	7.00E-04	Sn	1.46E-04	Er	7.43E-07	U	3.09E-06
S	1.50E-04	As	7.97E-04	Sb	1.93E-05	Tm	3.20E-06		
Cl	6.80E-07	Se	4.46E-05	Te	2.00E-07	Yb	2.88E-06		
Ar	1.00E-03	Br	5.31E-06	I	3.00E-05	Lu	1.36E-06		

EdF/Areva provided neutron flux data in the following three broad energy groups in their submission:

- 1 x 10^{-5} to 0.4eV;
- 0.4eV to 1 MeV;
- 1 MeV to 20 MeV.

However, FISPACT requires detailed neutron flux spectra and RWMD therefore utilised existing 69-group WIMS flux data for PWR fuel. These data had previously been extended to cover the 10 to 20 MeV energy range. The approach was to scale the detailed WIMS flux spectrum in each of the above three broad energy groups such that the total flux and the relative flux in each of the three bins matched that reported for the heavy reflector in the submission document. The irradiation time applied in the calculations was 60 years and activities were requested at a 5 year cooling time.

On the basis of the limited set of elements included in EdF/Areva calculations (see data highlighted in orange in Table 7 above), it was expected that these calculations should have adequately estimated the total activities of C-14, Mn-53, Mn-54, Fe-55, Co-60, Ni-59 and Ni-63 in the heavy reflector. It was therefore considered that a comparison of the FISPACT and EdF/Areva calculated total activities of these radionuclides in the heavy reflector would provide a valuable check on the modelling approach. The results of this comparison are presented in Table 8. This shows that with the exception of Co-60, where the EdF/Areva activity was 67% greater than that predicted by FISPACT, that total activities predicted by FISPACT were all within 50% of those given in the submission document. This level of consistency was judged to be acceptable at this time given the simplifying assumptions used in the FISPACT modelling performed by RWMD.

Table 8 Comparison of the total activity of the Heavy Reflector (EPR08) at 5 years cooling as predicted using FISPACT-2007 activation calculations and as supplied in the EDF/Areva submission

	Total Activity (Bq)	Total Activity (Bq)	
	FISPACT 2007 Result	EdF/Areva Result	EdF/Areva / FISPACT 2007
C-14	6.07E+14	8.21E+14	1.35E+00
Mn-53	4.65E+09	6.44E+09	1.39E+00
Mn-54	3.39E+03	3.15E+03	9.28E-01
Fe-55	4.18E+13	6.27E+13	1.50E+00
Co-60	1.71E+15	2.86E+15	1.67E+00
Ni-59	5.86E+14	6.52E+14	1.11E+00
Ni-63	1.13E+17	1.55E+17	1.37E+00

To generate the final total inventories for waste stream EPR08 for each radionuclide, the highest total activity as given in either the submission or by the FISPACT calculations was adopted. To generate the maximum and average waste package inventories for 5-year cooled waste each $3m^3$ Box was assumed to contain 3 t of the activated steel. Since the total mass of steel within EPR08 was about 138 t, this meant that the total waste stream inventory was spread evenly over 46 waste packages.

To generate a data summary sheet appropriate to 40-year-cooled waste, the 5-year-cooled inventory data was loaded into DIQuest and a further 35 years cooling applied.

To provide an approximate enhancement of the total inventories in waste streams EPR06 and EPR07, an approximate scaling factor to apply to the 5-year-cooled FISPACT inventory data was derived. The scaling factors were based on the inter-stream ratios of total activities of selected key radionuclides, as given in the EdF/Areva submission data.

In the case of waste stream EPR06 which is dominated by ferritic steel, Fe-55 was chosen as the key radionuclide since the main production route for Fe-55 comes from Fe and the concentration of Fe in the steels from wastes streams EPR06 and EPR08 are similar. The scaling factor so derived for EPR06 was 4.34×10^{-4}; the FISPACT inventory data for EPR08 was multiplied by 4.34×10^{-4} to make it applicable to EPR06. Again, the general approach of choosing the higher of the total activities was adopted. However, because the FISPACT calculation had been made for stainless rather than ferritic steel it was considered most appropriate for stream EPR06 to adopt the submission total activities for C-14, Mn-53, Mn-

54, Fe-55, Co-60, Ni-59 and Ni-63 even though these were below the scaled FISPACT activity data.

To enhance the inventory data for stream EPR07 the key radionuclide chosen for scaling was Ni-63; in this case the scaling factor was 1.69 x 10^{-1}. In this case, because both EPR07 and EPR08 are stainless steel streams it was not considered necessary to rely solely on the submission data for C-14, Mn-53, Mn-54, Fe-55, Co-60, Ni-59 and Ni-63 and the higher activity from either the FISPACT calculations or the submission was used.

Table 9 EPR waste stream data: decommissioning ILW[(1)]

Waste Stream	Package Type	Number of Packages	Total Packaged Waste Volume (m^3)	Maximum Package Alpha Activity (TBq)	Maximum Package Beta/ Gamma Activity (TBq)	Maximum Package A_2 Content	Maximum Package Heat Output (Watts)	Maximum Package Dose Rate at 1m from Package (mSv/hr)[(2)]
EPR06	4 metre Box	10	215.00	1.21E-04	1.98E+00	3.29E-01	3.14E-02	1.45E-01
EPR07	$3m^3$ Box	25	82.50	1.88E-02	1.12E+03	8.38E+01	7.93E+00	4.41E-03
EPR08	$3m^3$ Box	46	151.80	6.04E-02	3.59E+03	3.33E+02	3.57E+01	2.32E-02
TOTALS		81	449.30					

Notes:

(1) The values are for maximum waste package inventories (average package data not available).

(2) For EPR07 & EPR08 1m dose rates relate to outside of an SWTC-285 containing 1 x $3m^3$ Box

Summary of ILW Package Numbers and Characteristics

The information in Table 4 to Table 6 and Table 9 are underpinned by a detailed evaluation of the radionuclide inventory of each of the waste streams, as presented in Section 2 of Part 2 of this report.

Overall, RWMD concluded that good information exists on the nature and quantities of operational ILW for the EPR based on extensive experience of operating PWRs. An issue for future consideration is building confidence in the frequency of fuel cladding failure leading to contamination of the primary circuit, filters and clean-up resins.

The EdF/Areva submission identified that the pressure vessel internals (EPR07 and EPR08) would be constructed from stainless steel with major element content similar to Type 304. EdF/Areva reports the nitrogen content of this grade of stainless steel to be 800 ppm, which is relatively high and leads to the estimate of 923 TBq of C-14 per reactor for EPR07 and EPR08. Given the potential significance of this radionuclide to post-closure safety, EdF/Areva should select the grade of stainless steel to be used in an EPR carefully, taking account of the nitrogen impurities in the steel.

3.3.4 Comparison of EPR ILW with Sizewell B ILW

In order to place the information on the radioactivity of the ILW that would arise from an EPR in context, a comparison has been made with ILW from Sizewell B, which is the pressurised water reactor operated in the UK by British Energy. The Sizewell B design net electrical power output is 1188 MW(e) [23]and an assumed operating life of 40 years, whereas the EPR's electrical power output is 1,600 MW(e) for an assumed operating life of 60 years. Information on the Sizewell B ILW inventory has been taken from the 2007 National radioactive Waste Inventory [24].

Decommissioning ILW is the dominant source of many radionuclides in the estimated inventory for EPR. The radionuclide with the highest total activity in both operational and decommissioning ILW (from EPR) is Ni-63, and it is estimated that there is approximately 10,000 times more of this radionuclide in the decommissioning ILW than in the operational ILW. Similar (slightly larger) factors apply to Ni-59 and Co-60. The C-14 content of the EPR decommissioning waste at 923 TBq is about 1,000 times that in the operational waste. The inventories assigned to the decommissioning waste streams are upper bound values whereas those assigned to the operational wastes are central values. However, RWMD is of the view that the conservatism associated with the decommissioning waste inventory is unlikely to be more than a factor of 10, so that the decommissioning waste will still be the most important source of radionuclide activities.

The activity of EPR stainless steel decommissioning ILW (streams EPR07 and EPR08) is compared with the activity of the equivalent Sizewell B PWR waste [24] (2007 National Inventory stream 3S306) in Table 10. The basis for Table 10 is as follows:

- radionuclide activities have been estimated for 40 years after reactor shutdown;
- the activity data have been normalised to the total electrical output of the two reactors (Sizewell B – 1.18 GW(e) for 40 years, EPR 1.6 GW(e) for 60 years), this allows a like-for-like comparison of the radionuclide inventories between the two types of reactors, and highlights any differences that would result from the design of the reactor or the operational practices (e.g. intensity of neutron flux);
- all the EPR normalised activity values (measured in TBq per GW(e).yr) were reduced by a factor of three as EDF/Areva had used peak neutron fluxes to calculate radionuclide activities, whereas the Sizewell B data had used average neutron flux data in the activation calculations;
- the radionuclides considered in Table 10 are the top 10 most active in the EPR wastes for which estimates were also available for the Sizewell B PWR wastes;
- the cell colouration displayed in the sixth column of Table 10 is used to indicate the closeness of the agreement that presents the ratio of EPR to Sizewell B normalised activities as follows: green 0.33 to 3, yellow 0.1 to 0.33 & 3 to 10, pink <0.1 & > 10.

Table 10 Comparison of radionuclide activities for Stainless Steel decommissioning ILW from an EPR with Equivalent ILW stream from Sizewell B PWR (3S306)

Nuclide	Sizewell B (3S306) (TBq)	EPR (EPR07 and EPR08) (TBq)	Sizewell B (3S306) (TBq per GW(e).yr)	EPR (EPR07 and EPR08) (TBq per GW(e).yr)	(EPR07 + EPR08) / (3S306)*
Ni-63	3.35E+04	1.80E+05	7.09E-01	6.26E-01	8.82E-01
H-3	8.77E+01	6.68E+03	1.86E-03	2.32E-02	1.25E+01
Co-60	8.04E+02	3.15E+03	1.70E-02	1.09E-02	6.43E-01
Ni-59	3.23E+02	9.34E+02	6.85E-03	3.24E-03	4.73E-01
C-14	1.21E+02	9.23E+02	2.56E-03	3.21E-03	1.25E+00
Nb-93m	3.84E+02	6.95E+02	8.13E-03	2.41E-03	2.97E-01
Mo-93	1.21E+00	2.34E+02	2.56E-05	8.13E-04	3.18E+01
Fe-55	1.64E+02	7.87E+01	3.48E-03	2.73E-04	7.85E-02
Nb-94	4.04E+00	1.95E+01	8.56E-05	6.76E-05	7.90E-01
Tc-99	1.21E-01	4.11E+00	2.57E-06	1.43E-05	5.55E+00

* Ratio of (EPR07+EPR08) to Sizewell B (3S306) each normalised to TBq per GW(e).yr

As can be seen from Table 10, with the exception of H-3, the activities of the five radionuclides with the highest activities are similar (within a factor of three) and the activity of the radionuclide with the sixth highest activity, Nb-93m, is only just outside that range. Like H-3 the total activity of Mo-93 and Tc-99 is considerably higher in the EPR stainless steel wastes than that from Sizewell B. This can be explained by the application of conservative upper bound trace element concentrations in the RWMD inventory enhancement work.

The practices used in operating an EPR are subject to development, for example the timing of outages and the materials used to treat water in the cooling circuits, and, therefore, the volumes and activities of wastes are only estimates at this stage. For ILW, the most active waste streams are those from decommissioning, and estimates of decommissioning ILW from an EPR are primarily affected by assumptions regarding the neutron flux in the reactor and the composition of steel used in reactor internals.

In conclusion, radionuclide activity from EPR ILW is dominated by radionuclides within the decommissioning waste streams. Comparison with reported activities in similar wastes and normalised to facilitate a like-for-like comparison, shows that radionuclide activity in EPR decommissioning waste streams is comparable with that for Sizewell B.

3.4 Description of Spent Fuel, Packaging Assumptions, Waste Package Numbers and Characteristics

3.4.1 Description of spent fuel

The core of an EPR consists of 241 fuel assemblies providing a controlled fission reaction and a heat source for electrical power production. Each fuel assembly is formed by a 17×17 array of Zircaloy M5 tubes, made up of 265 fuel rods and 24 guide thimbles, as illustrated in Figure 8.

The rods are held in bundles by 11 spacer grids distributed at roughly uniform intervals up the 4.6m free height of the rods. The rods are fixed top and bottom into stainless steel nozzles that provide both structural integrity and direct coolant flow up the assembly. The total height of the assembly excluding the upper hold-down springs is 4.805m. The 24 guide thimbles are joined to the grids and the top and bottom nozzles. The guide thimbles are the locations for the rod cluster control assemblies (RCCAs – the control rods), the neutron source rods, or the in-core instrumentation. Guide thimbles that do not contain one of these components are fitted with plugs to limit the bypass flow. The grid assemblies consist of an 'egg-crate' arrangement of interlocked straps. The straps contain spring fingers (made from Inconel 718) and dimples for fuel rod support, as well as coolant mixing vanes.

The EPR fuel assembly and fuel rod are illustrated in Figure 8 and some additional dimensional information is provided in Table 11.

The fuel rods consist of uranium dioxide (UO_2) pellets stacked in a Zircaloy M5 cladding tube plugged and seal welded to encapsulate the fuel. Zircaloy M5 is a development of Zircaloy-4, which has been used previously for fuel rod cladding; the new alloy provides for greater radiation and chemical stability (i.e. corrosion-resistance in reactor water) to allow for higher burn-up in the reactor. Zircaloy M5 contains approximately 98.5% zirconium, with approximately 1.0% niobium, and trace iron and oxygen.

The stack of UO_2 pellets extends over a height of 4.2m known as the active height of the fuel. Above and below the UO_2 stack are the upper and lower fission gas plenums designed to accommodate any volatile fission products released during the irradiation process. An Inconel (believed to be Grade 718) spring is present in the upper plenum to maintain the dimensional integrity of the UO_2 stack, at the bottom of which is placed a thermal insulation pellet (believed to be made from alumina, Al_2O_3).

In some fuel rods, consumable neutron absorber ("burnable poison"), in which the fuel pellets are coated with neutron absorbing boron compound or gadolinium oxide (Gd_2O_3), is mixed with the UO_2 which contributes to controlling excess reactivity during the fuel cycle.

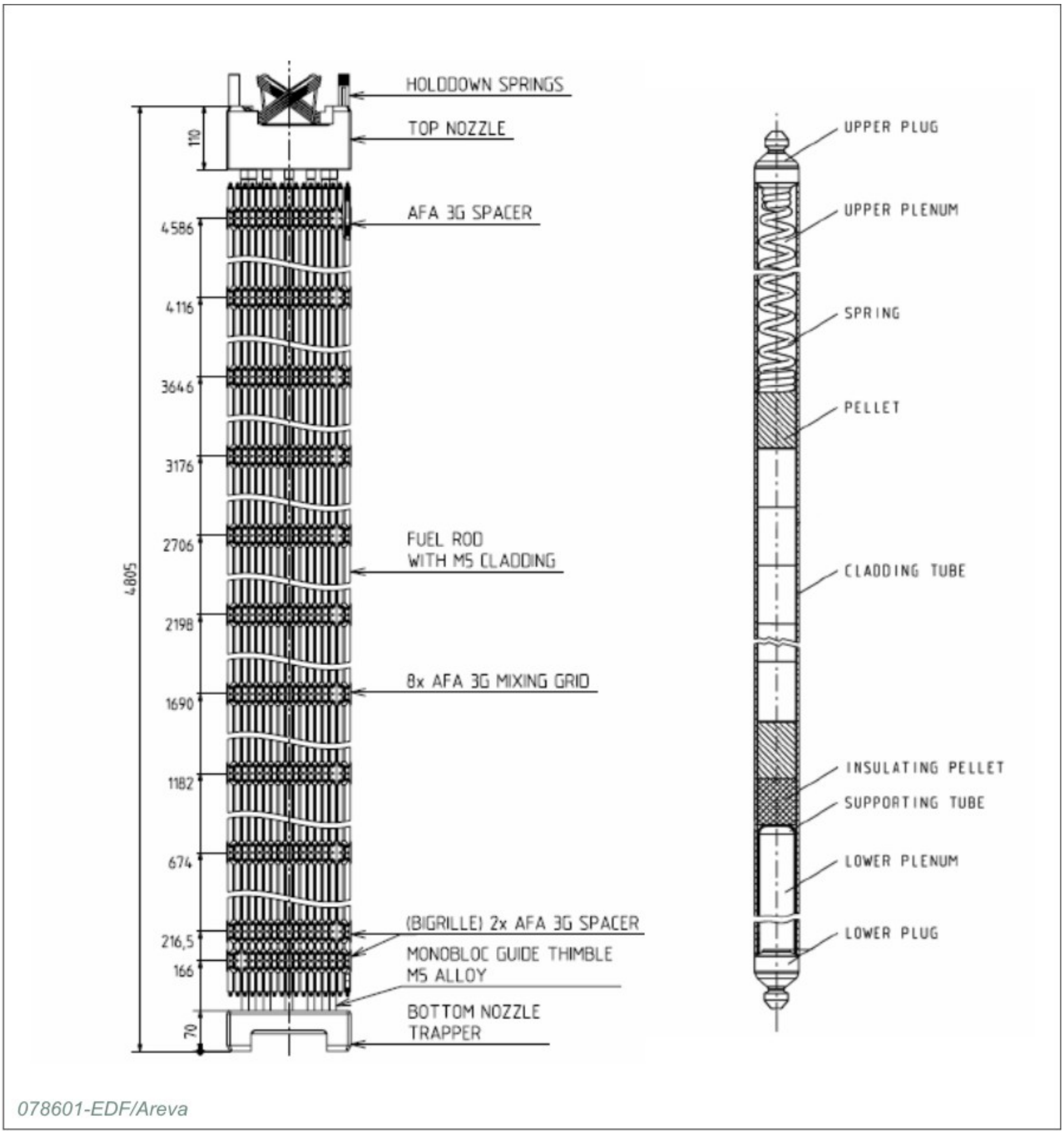

Figure 8 Components of an EPR fuel assembly (left) and a single EPR fuel rod (right)

Table 11 Dimensional information for EPR fuel assemblies and rods

Fuel Assembly	
External maximum section (mm x mm)	214 × 214
Maximum length (mm)	4859.5
Active length (mm) (Average, at 20 °C)	4200
Overall weight (kg)	779.8
Uranium mass (kg)	527.5
Fuel Rod	
Number of fuel rods	265
Fuel rod outer diameter (mm)	9.5
Cladding thickness (mm)	0.57
Pin pitch (mm)	12.6

3.4.2 Spent fuel packaging assumptions

The disposal concept adopted by RWMD and used within this assessment for spent fuel assumes that fuel assemblies will be loaded into a robust disposal canister. To accommodate the EPR design of fuel, the disposal canister would be required to be 5.2m in length (Figure 9). This is a development of the canister envisaged for legacy fuel from Sizewell B PWR and is approximately 0.6 m longer. The reference assumption is for four spent fuel assemblies to be packaged in each canister.

It is assumed that spent fuel will be packaged for disposal (sometimes referred to as encapsulation) before being dispatched to the GDF. For transport the packaged spent fuel would need to be shielded and contained in a reusable shielded transport over-pack. For the purposes of assessment, this is assumed to be accomplished by use of a Disposal Canister Transport Container (DCTC) which has been developed to a preliminary design stage by RWMD. The DCTC provides two layers of shielding material:

- immediately adjacent to the canister is a stainless steel gamma shield with thicknesses of 140mm in the radial direction and 50mm at the ends of the canister;
- surrounding the stainless steel gamma shield is a 50mm thick neutron shield made of a high neutron capture material such as ‘Kobesh’.

Although the quantitative analyses conducted in the GDA Disposability Assessment for the EPR are based on certain disposal concept assumptions, the implications of alternative disposal concepts also have been considered.

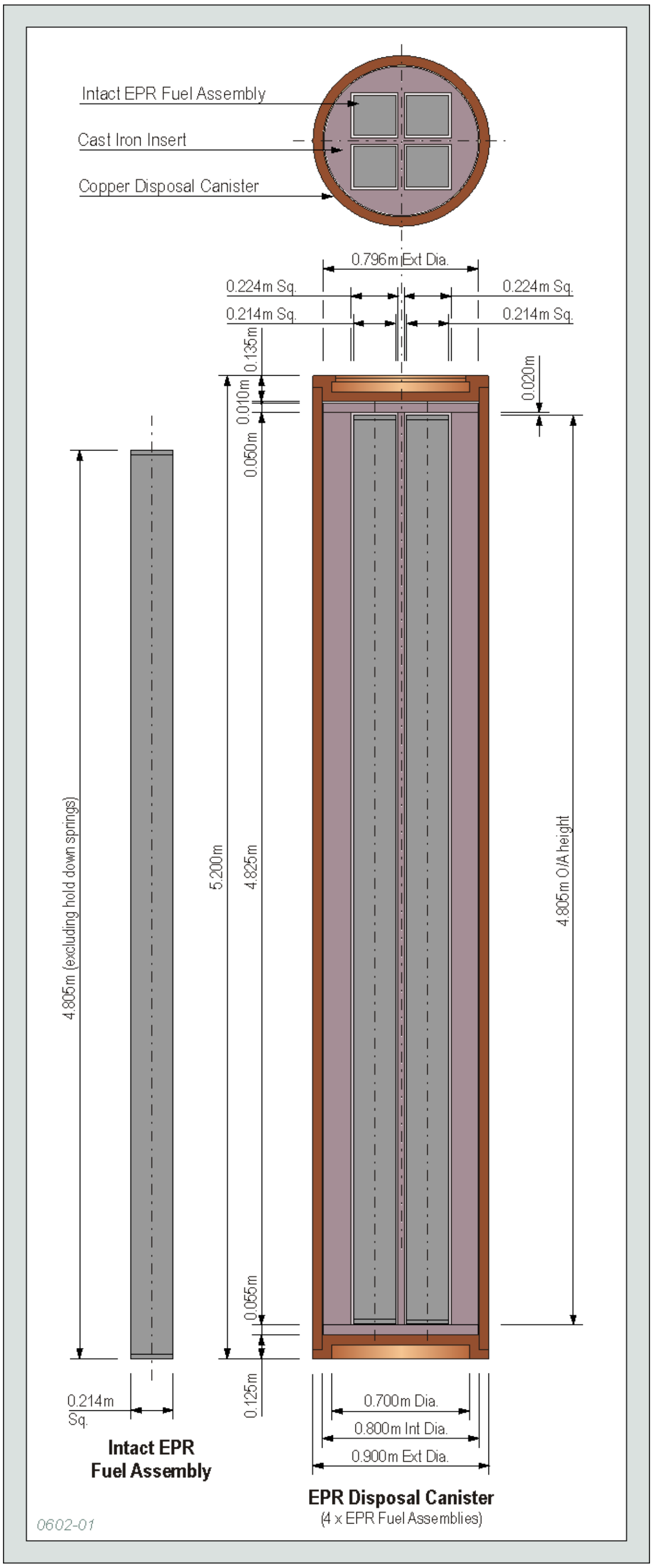

Figure 9 Illustration of an EPR spent fuel disposal canister

3.4.3 Spent fuel package numbers and characteristics

The GDA Disposability Assessment for the EPR assumes that 90 fuel assemblies will be generated every 18 months of reactor operation, which, for an assumption of 60 years operation, results in a total of 3,600 assemblies requiring disposal, i.e. 900 canisters.

The RCCAs described in Section 3.4.1 were not included in the initial disposal inventory supplied by EdF/Areva. Although these items may have high specific activity, they will not be of large volume, and, therefore, are not expected to affect disposability of wastes from an EPR. These components could be managed as ILW or, given their dimensions, packaged as a complete unit with their associated fuel assembly. The RCCAs are longer than the spent fuel, but can be reduced in size by removing the end supports. In any future submission under the LoC process, the operator should provide further information on proposals for the management of RCCAs.

The dimensions of one fuel assembly are 0.214m x 0.214m x 4.805m (Figure 9), so the raw waste volume associated with 3,600 fuel assemblies is 792 m^3. Regarding packaged volume, the envelope volume of a canister capable of accommodating four fuel assemblies is 3.33 m^3, and the packaged volume of the waste consisting of 900 canisters is therefore 2,997 m^3.

EdF/Areva provided radionuclide inventory data for one-year-cooled spent fuel that had been irradiated to 65 GWd/tU [25]. This dataset was generated by the ORIGEN-S [26] inventory code which is internationally considered to be the most reliable for the generation of PWR spent fuel inventory data. The calculations considered the UO_2, Zircaloy M5 cladding, guide thimbles and grids and the Inconel 718 springs present within the fuelled zone of the assembly. At the cooling times of interest in a disposability study (greater than about 50 years) the radionuclide content of the components, such as the nozzles, located outside the fuelled zone are of negligible importance.

For the calculations EdF/Areva investigated three potential power histories. The history that was adopted and considered to be most challenging from the waste management perspective, consisted of four short cycles with constant high specific power. Each of the four cycles consisted of a 388.7 day irradiation at a specific power of 41.81 MW/tU followed by a 10 day decay period. The starting composition of the UO_2 used in the calculations is provided in Table 12. Alternative fuel cycle assumptions potentially would reduce the challenges, but a sensitivity study to examine the magnitude of any possible reductions has not been undertaken at this time.

Table 12 Starting composition of the UO_2 used in EdF/Areva calculations of spent fuel radioactivity.

Isotope / Element	ORIGEN input Concentration (g/tU)

The starting compositions and amounts of the Zircaloy M5 and Inconel 718 are given in Table 13 and Table 14. By expressing these compositions in terms of g per tU, Table 13 and Table 14 define the amounts of Zircaloy M5 and Inconel 718 assumed present within the

EdF/Areva have also provided separate one-year cooled spent fuel inventory data for the actinide, fission product and light element content of the UO_2 and the light element content of the structural materials [25]. These one-year-cooled inventory components have been combined and imported into DIQuest so that various waste package and radionuclide related parameters such as heat output can be derived (Table 15).

The component mass estimates for an EPR fuel assembly are provided in Table 13. This table includes a small additional quantity of Zircaloy M5 used to balance the individual component masses with the total spent fuel assembly mass. Table 14 presents the mass data for each fuel assembly and each canister (for illustrative purposes, copper has been assumed as the material of manufacture in this case), summed for each material type.

Package data are summarised in Table 15. The information in Table 15 is underpinned by a detailed evaluation of the radionuclide inventory. This is presented in Section 3 of Part 2 of this report. In compiling the package data it was necessary to define a cooling period which would form a baseline for package characteristics such as activity, heat loading and dose rate. RWMD initially assumed that spent fuel would require cooling for an interim period of about 90 years before disposal and this period was adopted as the basis for the characteristics listed in Table 15. In later stages of the assessment RWMD undertook heat transfer calculations to determine how much cooling would be appropriate before emplacement in a GDF (this is described in Section 5.1).

Table 13 Estimates of component mass for an EPR fuel assembly

Component of fuel assembly	Material	Mass per assembly (kg)
UO_2	UO_2	5.98E+02
Cladding, grids & guide tubes within active region	Zircaloy M5	1.46E+02
Cladding, grids & guide tubes outside active region	Zircaloy M5	1.13E+01
Upper & Lower plug for fuel pin	Zircaloy M5	1.29E+00
Additional Zircaloy M5 mass	Zircaloy M5	3.43E+00
Inconel 718 grid spring within active zone	Inconel 718	6.60E-01
Top nozzle spring	Inconel 718	1.30E+00
Plenum springs	Inconel 718	2.40E+00
Top & Bottom Nozzle	AISI 304 L St Steel	1.46E+01
Alumina Insulating pellets	Al_2O_3	5.95E-01
Total		**7.80E+02**

Table 14 Material mass breakdown for an EPR fuel assembly and for a copper canister (assuming four assemblies per canister)

Material	Mass per Assembly (kg)	Mass per Canister (kg)
UO_2	5.98E+02	2.39E+03
Zircaloy M5	1.62E+02	6.48E+02
AISI 304 L Stainless Steel	1.50E+01	6.00E+01
Inconel 718	4.00E+00	1.60E+01
Al_2O_3	1.00E+00	4.00E+00
Total	**7.80E+02**	**3.12E+03**

Table 15 EPR waste stream data: spent fuel[(1)]

Waste Stream	Package Type	Number of Packages	Total Packaged Waste Volume (m^3)	Maximum Package Alpha Activity (TBq)	Maximum Package Total Beta/ Gamma Activity (TBq)	Maximum Package A_2 Content	Maximum Package Heat Output (Watts)	Maximum Package Dose Rate at 1 m from Transport Container (mSv/hr)	Maximum Package Total Fissile Content (g) {U233+ U235+ Pu239+ Pu241}
EPR09	Disposal Canister	900	2997.00	1.03E+03	3.59E+03	1.02E+06	1.43E+03	1.20E-01	2.67E+04

Notes:

(1) The values are for maximum waste package inventories (a single set of pessimistic assumptions were used to derive the inventory data so average package data are not available) after 90 years cooling.

Although EdF/Areva is designing and planning for burn-up of fuel to 65 GWd/tU, this is the maximum burn-up that a fuel assembly would experience. The average burn-up across all fuel assemblies in the core will be somewhat lower than this and will be determined by the fuel management regime implemented by the operator. At this stage of the assessment EdF/Areva has not been able to provide further information on average irradiation. To give an idea as to the potential difference between average and maximum burn-up, RWMD has estimated average irradiation as follows.

The lifetime thermal energy production for an EPR at a load factor of 93% would be 9.17E+04 GWd. These 3,600 EPR fuel assemblies would contain 1,899 tU. Therefore, assuming that 3,600 fuel assemblies are generated over the lifetime of a reactor implies that the average burn-up of the assemblies is 48.3 GWd/tU. In calculating the total spent fuel inventory for the post-closure performance assessments, it was assumed that all 3,600 spent fuel assemblies had been irradiated to 65 GWd/tU, rather than 48.3 GWd/tU. This is clearly conservative although the conservatism only amounts to about a factor of 1.3 for most of the post-closure significant radionuclides. This average burn-up value has been used to illustrate the impact on required spent fuel interim storage period in Section 5.1.

3.4.4 Comparison of EPR spent fuel with Sizewell B PWR spent fuel

Fuel used to generate heat in an EPR would be expected to experience higher burn-ups than existing commercial reactors in the UK, for example the PWR at Sizewell B. Higher burn-ups result in efficiency savings for the operator. For a similar quantity of electricity produced, an EPR would create a smaller volume of spent fuel.

For example, an EPR operating for 60 years at 1.6 GW(e) would produce 3,600 spent fuel assemblies; this is equivalent to 37.5 spent fuel assemblies for every GW(e) year. In comparison, assuming the PWR at Sizewell B operates for 40 years at 1.188 GW(e) and produces 2,228 spent fuel assemblies [27], 46.9 spent fuel assemblies would be produced for every GW(e) year. Thus the efficiency gains can be seen, however it should be noted that this does lead to a higher concentration of activity in EPR spent fuel assemblies in comparison to Sizewell B PWR spent fuel assemblies.

Table 16 provides a comparison of the radionuclide inventories for the most significant post-closure radionuclides in spent fuel from an EPR, with radionuclide inventories for spent fuel from Sizewell B PWR. The comparison is based on the inventory of radionuclides estimated to be present in one spent fuel canister at 90 years cooling[16]. The data for the Sizewell B PWR are derived from the Low Burn-up PWR data presented in [28], the fission product and actinide data from which were used in a previous assessment of the implications associated with new build reactors undertaken by Nirex [29].

The only comparison of EPR and Sizewell B spent fuel inventories that could readily be made involves EPR's maximum fuel assembly average burn-up inventory with the batch average fuel burn-up inventory associated with Sizewell B, as reported in [23]. It is recognised that it would have been more appropriate to compare either the two maximum fuel assembly average burn-up cases or two batch average fuel burn-up inventories. However, relevant information was not available for such comparison at the time of this assessment. Since the burn-up assumed for EPR spent fuel is about twice that assumed for the Sizewell B spent fuel, for many radionuclides the ratio of EPR to Sizewell B fuel activities is about two, as shown in Table 16. Ratios a little below and above two reflect non-linearity

[16] 90 years was selected at the outset of this assessment to provide a reasonable approximation of the amount of cooling time expected before disposal. A more considered view is covered in Section 5.1.

effects that arise from, for example, the higher proportion of fissions coming from Pu-239 in the higher burn-up fuel. A few of the activity ratios are outside the range that might be expected from the different burn-ups and these, perhaps unexpected differences are attributable to five separate causes which are discussed below. Yellow, pink, blue and green shadings have been used in Table 16 to identify the causes of the apparently anomalous activity ratios.

Table 16 Comparison of radionuclide activities for spent fuel from an EPR with spent fuel from Sizewell B

Radionuclide	Sizewell B SF (TBq per Canister)	EPR SF (TBq per Canister)	Ratio of EPR:Sizewell B
C-14	6.45E-02	3.11E-01	4.8
Cl-36	8.31E-04	1.57E-02	19
Ni-59	9.08E-04	3.63E-02	40
Se-79	3.18E-02	1.01E-02	0.32
Sr-90	6.75E+02	1.27E+03	1.9
Tc-99	1.03E+00	1.89E+00	1.8
Sn-126	5.67E-02	8.59E-02	1.5
I-129	2.39E-03	4.81E-03	2.0
Cs-135	3.02E-02	7.22E-02	2.4
Cs-137	1.02E+03	2.06E+03	2.0
U-233	1.23E-05	2.91E-05	2.4
U-234	1.33E-01	2.31E-01	1.7
U-235	1.53E-03	1.05E-03	0.69
U-236	2.15E-02	3.67E-02	1.7
U-238	2.46E-02	2.36E-02	1.0
Np-237	3.28E-02	6.94E-02	2.1
Pu-238	9.09E+01	3.91E+02	4.3
Pu-239	2.50E+01	3.10E+01	1.2
Pu-240	3.61E+01	6.03E+01	1.7
Pu-241	1.23E+02	2.15E+02	1.7
Pu-242	1.24E-01	3.90E-01	3.2
Am-241	2.83E+02	4.97E+02	1.8
Am-242m	7.32E-01	8.21E-01	1.1
Am-243	1.14E+00	6.26E+00	5.5

Yellow cells: C-14, Cl-36 and Ni-59. These radionuclides arise mainly as activation products of trace impurities or in the case of Ni-59, from trace impurities and the small amount of a nickel alloy (Inconel 718) used for grid springs. The stable elements responsible for these activation products are: nitrogen for C-14; chlorine for Cl-36; nickel for Ni-59. In general, EdF/Areva has adopted more conservative specification limit values for the trace impurities in their spent fuel inventory calculations than has been adopted by RWMD in previous studies of PWR fuel. This has led to EPR inventories that are more than the factor of two greater than those coming from the Sizewell B calculations (identical impurity levels would have resulted in EPR inventories being about twice the Sizewell B inventories because of the two-fold higher irradiation). For example, for its calculations EdF/Areva adopted chlorine concentrations of approximately 25ppm and 20ppm for the UO_2 and Zircaloy M5 cladding respectively, whilst the Sizewell B calculations used approximately 5ppm chlorine for the UO_2 and neglected the chlorine content of the cladding. Based on an extensive Cl-36 research project conducted by Nirex in the 1990's the chlorine concentrations adopted for the

Sizewell B calculations are considered more justifiable (i.e. the upper bound chlorine concentration for LWR UO_2 and Zircaloy were assessed to be approximately 5ppm and 1.7ppm respectively [30],[31]).

The large (factor of 40) activity ratio calculated for Ni-59 arises from the extra activity induced in the nickel-rich Inconel 718 grid springs of the EPR assembly. The calculations performed for the Sizewell B fuel did not include any Inconel fuel structural component.

Pink cells: Se-79. Differences in the estimated activities of Se-79 are associated with changes to data on the fission yield and half-life of this radionuclide, and these parameters have been revised in recently published nuclear data libraries. For a given fission yield in terms of number of atoms, the associated activity is inversely proportional to half-life. The estimated activity of Se-79 for an EPR used a half-life for the radionuclide of about 3.3E+05 years. However, the Sizewell B estimates used a Se-79 half-life of 6.5E+04 years, and the difference in Se-79 activity presented in Table 16 is in accord with the difference in half-lives and burn-ups associated with the two with the two spent fuel calculations used to develop the estimates.

Blue cells: U-235. The lower activity of U-235 present in the EPR spent fuel is relatively straightforward to explain, it is merely a feature of the higher burn-up experienced by the EPR spent fuel. Since U-235 is the main fissile isotope in the fuel to achieve a higher burn-up, more U-235 must be consumed. Fission of Pu-239 and Pu-241 complicates the detailed fissile mass balance but extra consumption of U-235 in high burn-up fuels is expected.

Green cells: Pu-238, Pu-242 and Am-243. A number of higher mass actinides are produced by multi-step activation reactions. A characteristic of such reactions is that they produce an increase in activity above the linear dependence found for most fission products and low mass actinides. For example, Pu-238 is produced by the activation of Np-237 which in turn is produced from the irradiation of both U-236 and U-238. This is an example of a simple two step activation reaction for which the activity of the product (Pu-238) increases as the second power of burn-up. Thus a two-fold increase in burn-up results in a four-fold increase in Pu-238 activity. In other actinide build-up chains, such as those involving Pu-239, Pu-240 and Pu-241, saturation and decay effects complicate the position. Hence, the increase in Pu-242 and Am-243 activity is not as fast as would be anticipated by the number of activation steps required for their production. However, the above-linear increase of Pu-242 and Am-243 activity with burn-up is still fundamentally down to the fact that they are produced by multi-step activation reactions.

Given the pessimisms associated with the per canister inventories, it can be concluded that the radionuclide characteristics of spent fuel from an EPR are consistent with those from Sizewell B PWR.

4 ASSESSMENT OF EPR OPERATIONAL AND DECOMMISSIONING ILW

In this section we discuss the assessment of EdF/Areva's packaging proposals for ILW against RWMD's waste package specification [7] and disposal system specification [9] discussed in Section 2.1. The approach used follows that described in Section 2.2. The assessment is reported in four sections:

- Section 4.1 describes the assessment of the packages proposed by EdF/Areva, including consideration of proposed waste containers (Section 4.1.1), wasteforms (Section 4.1.2) and predicted waste package performance (Section 4.1.3);

- Section 4.2 describes consideration of the impact of EdF/Areva's waste packaging proposals on operation of the disposal system, including engineering design impact (Section 4.2.1), safety during the transport of waste to the GDF – transport safety (Section 4.2.2), safety during the receipt, handling and emplacement of waste in the GDF – operational safety (Section 4.2.3), environmental issues (Section 4.2.4), and security and safeguards implications (Section 4.2.5);

- Section 4.3 describes the assessment of the impact of EdF/Areva's waste packaging proposals on long-term safety following closure of the GDF;

- Section 4.4 provides a statement regarding the overall disposability of ILW from an EPR and identifies the basis for this statement.

For each component of the assessment, the context is discussed (i.e. the required performance), and the results and the implications of the assessment are provided. Issues identified under each component of the assessment are listed in Appendix B. These would be required to be addressed in future LoC assessment process by operators if any of the outlined packaging proposals were to be pursued.

4.1 Waste Package Characteristics and Performance

4.1.1 Waste container characteristics

The Generic Waste Package Specification (GWPS) [7] is the primary means by which RWMD defines the required characteristics and key features of ILW waste packages. The specification introduces the concept of "standard" waste packages to give confidence that the waste packages will be able to be safely and efficiently transported to the GDF and on receipt, be able to be handled and emplaced using standard equipment.

C1 and C4 concrete casks and cast-iron casks are not used currently in the UK as disposal packages and would be classed as "non-standard". This is not in itself a major impediment as non-standard containers can be assessed, and if disposability requirements are met, they may be subsequently adopted and introduced into the standard range. RWMD would expect a detailed assessment of the container and packages to be undertaken using the LoC assessment process in the future during development of the waste packaging processes.

The GWPS specifies the following characteristics for standard waste containers:

- dimensions within a defined envelope;

- standardised lifting features;

- gross mass not exceeding [package specific limit];

- defined unique identifier format and location;
- physical containment provided by container body, lid and sealing system;
- standardised stacking characteristics;
- filtered venting where necessary.

The above waste container "standard" criteria have been used as a check-list for the review of the different waste container types proposed in the EdF/Areva submission. It would not be productive to apply this in a mechanistic way to known non-standard containers, but applied sensibly it has enabled some issues that would require future follow up to be identified.

The results of the assessment are reported in Table 17, and the most significant points discussed below. It should be noted that a key factor influencing long-term behaviour of all waste containers will be the environment in which the completed waste package is stored following manufacture. RWMD has issued generic guidance on appropriate storage environments (see [32] and other guidance listed in Appendix A). RWMD would follow-up on storage conditions with operators under a future LoC assessment.

Operational ILW - Reference Case

C1 and C4 concrete casks are currently non-standard. RWMD has no information or experience of the use of this type of concrete cask, although has experience of use of a concrete box for packaging decommissioning wastes from the Windscale AGR. Assessment of the proposals has identified various uncertainties which would need to be followed-up in the future under the LoC process. The fact that these containers are licensed for use in France gives some confidence that these uncertainties may not lead to insurmountable issues.

Information provided by EdF/Areva on the casks does not include a description of their material composition beyond being described as "concrete". For example, there is no information on the constituents of the concrete, on any additives (e.g. superplasticiser) or on the form of any reinforcement. Further, more detailed, information on the nature and performance of the casks would need to be considered for a future LoC submission.

Chemical superplasticisers are commonplace in the construction industry, as aids to improve the workability of concrete. In radioactive waste management their use is best avoided where possible, or where they fulfil a justified need, their chemical constituents and long-term effect (e.g. on radionuclide solubility) should be understood. Regarding reinforcement, RWMD assumes that these containers would be reinforced, possibly with the inclusion of metal or polypropylene fibres. Details of the construction and constituents would be needed for a future LoC assessment. RWMD would also need to understand the long-term durability of such containers as part of such an assessment.

It should be noted that since concrete casks are currently non-standard, adoption of their use and endorsement through the LoC process would result in an increase in the number of package types that need to be handled at the GDF. This would increase handling complexity at the GDF and thus increase costs and may have other implications for operability. For example, RWMD is not aware of the extent of stackability of the concrete casks and whether their adoption would require changes to the design of disposal vaults and the related effects such as impact on the repository footprint.

Variant Case 1

Options utilising standard containers for which RWMD has experience, such as the standard stainless steel 500-litre Drum waste container, will impose the least burden of proof on future operators. RWMD will need to assess specific designs in future LoC assessments to confirm that the container criteria will be met.

Variant Case 2

Cast-iron casks are currently non-standard containers. RWMD has no information or experience of their use although they are being investigated by UK operators for packaging ILW on some of the existing sites. The fact that these containers are licensed for use in Germany gives some confidence that these uncertainties may not lead to insurmountable issues. RWMD is at an early stage of interaction with the operators and any information provided for that assessment is protected under commercial arrangements and has not been used within this GDA assessment.

Cast-iron casks are not vented and in the event of gases being generated inside (from wasteform degradation mechanisms – see Table 17) the container cavity may pressurise. This depends on a number of factors including the gases themselves, the potential for recombination reactions and the performance of the sealing system. Further information on this would be required in future LoC submissions. The other issue identified is the need to understand the long-term durability of the container and its components. In principle cast-iron casks should be durable and long-lasting. However, for a future LoC submission this will need to be demonstrated, including the role of the protective paint finish and the necessary lifetime of the lid seal.

Since cast-iron casks are currently non-standard, adoption of their use and endorsement through the LoC process would result in an increase in the number of package types that need to be handled at the GDF. This would increase handling complexity at the GDF and thus increase costs and may have other implications for operability.

Decommissioning Wastes

Options utilising standard containers for which RWMD has experience will impose the least burden of proof on future operators. Therefore, the $3m^3$ Box/4 metre Box options are unlikely to raise any waste container incompatibility issues. RWMD will need to assess specific designs in future LoC assessments to confirm that the container criteria will be met.

Table 17 Waste container criteria: Operational and decommissioning packaging options

Waste Container	Concrete cask	500 litre drum (as specified in WPS300)	Cast-iron cask	$3m^3$ Box (as specified in WPS 310 or WPS 315)	4 metre Box (as specified in WPS 330)
Dimensions within a defined envelope	Non-standard but within the envelope of a 4m box	Standard, as specified.	Non-standard but within the envelope of a SWTC	Standard, as specified.	Standard, as specified.
Standardised lifting features	Non-standard. Modified or new lifting arrangements can be developed, but may have cost and operability implications for the GDF.	Standard. Lifting feature is the drum body lid flange with a dedicated drum grab that engages with the flange. Note that 500-litre drums will typically be lifted in stillages at the GDF.	Non-standard. Modified or new lifting arrangements can be developed, but may have cost and operability implications for the GDF	Standard. Twistlock fittings on top face of container	Standard. Twistlock fittings on top face of container
Gross mass within specified mass limit	Non-standard, but maximum of 6.7 tonnes is within the acceptable mass limit for a 4m Box (for comparison).	Estimate 1.25 tonnes average. Within specified mass limit of 2 tonnes	Non-standard. Maximum of 6.7 tonnes is within the acceptable mass limit for a 4m Box (for comparison)	Estimate 9.3 tonnes average. Within specified mass limit of 12 tonnes	Contents can be controlled to meet specified package mass limit of 65 tonnes.
Defined identifier format and location	Non-standard container, assume specified format will be met. Positions will need to be agreed in the future.	Standard. Alpha-numeric identifier in machine readable format in four positions on drum body lid flange	Non-standard container, assume specified format will be met. Positions will need to be agreed in the future	Standard. Alpha-numeric identifier in machine readable format in four positions on box body	Standard. Alpha-numeric identifier in machine readable format in four positions on box body

Note: Documents denoted WPS 300, WPS 310 etc., are documents from the RWMD document suite known as Waste Package Specification and Guidance Documentation (WPSGD)

Waste Container	Concrete cask	500 litre drum (as specified in WPS300)	Cast-iron cask	3m^3 Box (as specified in WPS 310 or WPS 315)	4 metre Box (as specified in WPS 330)
Physical containment provided by container body, lid and sealing system	Non-standard. Further information will be required for a future LoC assessment regarding effectiveness of the concrete and lid sealing to provide both short-term and long-term physical containment.	Standard. Stainless steel containment system with bolted or welded lid. Lid sealing sufficient to retain particulates	Non-standard. Cast-iron containment system, protected with paint system on outer surface. Lid sealing system with elastomer seal. Potential for similar longevity to standards, depending on thickness and performance of cast iron, and contribution from paint. Further information will be required for a future LoC assessment regarding longevity of containment and effectiveness of seals	Standard. Stainless steel containment system with bolted or welded lid. Lid sealing sufficient to retain particulates	Standard. Stainless steel containment system with bolted lid incorporating a testable elastomer seal. Sealing to meet IAEA Transport Regulations
Standardised stacking characteristics	Extent of stackability not known by RWMD.	Standard. Will be consolidated into 4-drum stillages stacked 7 high in the GDF.	Extent of stackability not known by RWMD.	Standard. Integral stacking posts, to be designed for 7 high stacking.	Integral stacking posts, to be designed for 6 high stacking
Filtered venting where necessary	Concrete cask likely to be sufficiently permeable (note: wasteform permeability requirements considered in Table 18)	Filtered vent can be fitted where necessary	No vent. It will need to be established in a future LoC assessment that gas venting is not required	Filtered vent can be fitted where necessary	Filtered vent can be fitted where necessary, but will need to be established that physical containment meets IAEA Transport Regulations

4.1.2 Wasteform

The production of a wasteform is the currently accepted common practice by which the original 'raw' waste is conditioned and rendered into a passively safe form, so wasteform design can have a significant influence on waste package performance under both normal and accident conditions. A range of parameters can affect the quality of the wasteform, and thus its acceptability. The principal parameters considered under the wasteform assessment are based on those defined in the GWPS [7], as follows:

- *physical immobilisation*: the wasteform shall be designed to immobilise radionuclides and toxic materials so as to ensure appropriate waste package performance during all phases of waste management. For many wastes, this immobilisation requires the use of an encapsulating matrix;

- *mechanical and physical properties:* the wasteform shall be designed to provide the mechanical and physical properties necessary to ensure appropriate performance of the waste package during all phases of waste management;

- *chemical containment:* the wasteform shall not be incompatible with the chemical containment of radionuclides and hazardous materials;

- *hazardous materials:* the wasteform shall not contain hazardous materials, or have the potential to generate such materials, unless the conditioning of such materials or items makes them safe. The means by which any of these materials is made safe shall be demonstrable for all phases of waste management;

- *gas generation:* gases generated by the wasteform shall not compromise the ability of the waste package to meet the GWPS;

- *wasteform evolution:* changes in the characteristics of the wasteform as it evolves shall not result in degradation that will compromise the ability of the waste package to meet the GWPS.

The Reference Case and variant proposals for packaging of operational ILW include outline descriptions of the means of conditioning and immobilising activity associated with the waste. Detailed arguments and supporting evidence as to the properties of the proposed wasteforms have not been presented by EdF/Areva, consistent with expectations for this stage of the GDA Disposability Assessment. In future, RWMD would expect to work with potential reactor operators to achieve fully-developed proposals through the Letter of Compliance process.

The Wasteform evaluation considered the criteria listed above on a waste-stream by waste-stream basis [33]. The results of the evaluation are reported in Tables 18 to 21. The key points are summarised below:

Operational Waste - Reference Case

The wastes arising from operation of the EPR are similar to, or not greatly dissimilar from, existing ILW, and therefore any wasteform performance issues will arise from the detail of specific wasteform design. The Reference Case conditioning proposal is to produce solid waste products within the concrete casks, by use of epoxy resin or cement grout to infiltrate or encapsulate the wastes.

The key points of the wasteform evolution [33] are summarised in Table 18 for the Reference Case.

Table 18 Wasteform characteristics: Operational ILW - Reference Case, C1/C4 casks

Waste stream	Ion exchange resin	Filters, sludges	Operational waste
Conditioning proposal	Conditioned with epoxy resin within welded inner steel shielding container	Conditioned with cement grout	Compacted and cement grouted
Physical immobilisation	Measured volume of spent ion exchange resin is placed in the concrete container and infiltrated with epoxy resin to form solid product. Based on experience appropriate immobilisation is likely to be achievable	Waste is mixed or infiltrated with cement grout to form solid product. Extent of infiltration into filter elements not known, but based on experience appropriate methods of immobilising filters could be developed (see RWMD guidance on filters [34]).	Waste is placed in plastic bag and compacted before grouting. Waste is entombed and activity is unlikely to be immobilised intimately. Based on experience, sufficient containment may be achievable from the grout annulus and waste container.
Mechanical/ physical properties	Experience suggests that typical grouts and epoxy resins are both likely to have acceptable strength and mass transport properties	Wastes closely resemble other wastes that have been considered for packaging in the UK. Experience suggests that the proposed conditioning is likely to be acceptable and the use of calcium hydroxide ($Ca(OH)_2$) to ameliorate possible delayed cement curing due to presence of boron and zinc in sludges is likely to be successful. Alternative approaches, such as the use of calcium sulpho-aluminate cement, could also be considered.	An analogy for these wastes is those wastes packaged in the UK by supercompaction (rather than lower-force compaction) and subsequent grouting. Experience suggests that a typical grout is likely to have acceptable strength and mass transport properties.
Chemical containment	Long-term degradation of organic polymers and effect on pH conditioning within backfilled vaults has some associated uncertainties. However, presence of grout within casks will reduce effects on backfill pH. Although the degradation products may include species that could complex radionuclides, current experience suggests	These wasteforms closely resemble those that have been considered previously. Experience suggests they are unlikely to affect chemical containment and will be acceptable	Although the composition of these wastes may be variable, they are likely to closely resemble those that have been considered previously. The possible presence of cellulosic material may lead to the formation of degradation products with a known ability to complex radionuclides. However, in the context of the overall amount of

Waste stream	Ion exchange resin	Filters, sludges	Operational waste
	these are not significant although the nature of any degradation products should be investigated. The wasteform is therefore likely to be acceptable.		cellulosic waste in the UK waste inventory it will represent a small addition and is expected to be acceptable.
Hazardous materials	No hazardous materials, with the exception of some chemo-toxic elements, are likely to be present in the waste. Where a comparison can be made, the amounts of chemotoxic materials are small compared to those already present in the UK inventory.	No hazardous materials, with the exception of some chemo-toxic elements, are likely to be present in the waste. Where a comparison can be made, the amounts of chemotoxic materials are small compared to those already present in the UK inventory.	No hazardous materials, with the exception of some chemo-toxic elements, are likely to be present in the waste. Where a comparison can be made, the amounts of chemotoxic materials are small compared to those already present in the UK inventory.
Gas generation	Radiolytically-generated gas may expand the wasteform or migrate, and pressurise the inner steel shielding container. The inclusion of a suitable gas leakage path for the inner container (e.g. by not welding the lid) could be considered. The dimensional stability of the epoxy resin would need to be addressed as part of a future LoC submission.	The wastes closely resemble those that have been considered previously. Experience suggests gas generation is unlikely to raise significant issues and that the permeabilities of standard waste encapsulation grouts are likely to be sufficient to prevent pressurisation of the wasteform.	The wastes are expected to be typical of those that have been considered previously. Experience suggests gas generation is unlikely to raise significant issues and that the permeabilities of standard waste encapsulation grouts are likely to be sufficient to prevent pressurisation of the wasteform.
Wasteform evolution	Although there are current uncertainties over the long-term degradation of organic polymers and ion-exchange resin and consequent effect on wasteform performance, similar wastes have been considered previously and it is expected that wasteform evolution is likely to be acceptable.	The filter wasteform closely resembles standard wastes that have been considered previously and experience suggests that wasteform evolution is likely to be acceptable. The long-term evolution of a well-formulated sludge wasteform is also likely to be acceptable.	Containment of the waste is provided partly by the physical barrier offered by the grout annulus. Evidence will be required for a future LoC assessment that the annulus will retain its effectiveness as a physical barrier.

It is concluded that, intimate immobilisation of radionuclides and particulates is likely to be achieved with the possible exceptions of filters and operational wastes where cement grout may not fully infiltrate and encapsulate the wastes. In these cases particulates within the body of the waste may not be immobilised, and performance will need to rely more on the robustness of the surrounding grout and waste container combination. The ability of cement grout to infiltrate cartridge filters and trap particulate activity within the filter elements will require further demonstration although, consistent with RWMD Guidance [35], this is expected to be able to provide satisfactory immobilisation with appropriate formulations.

Radiolytically-generated gas may expand the epoxy resin polymer wasteform or migrate and pressurise the inner steel shielding container. The inclusion of a suitable gas leakage path for the inner container (e.g. by not welding the lid) could be considered. The gas generation rate and the dimensional stability of the specific epoxy resin would need to be addressed as part of a future LoC submission.

Variant Case 1

For Variant Case 1, it is proposed that the filters are grouted in 200 litre Drums and these are then grouted into 500 litre Drums. The other wastes would be dried and packed in 200 litre drum and then macro-grouted into 500 litre Drum to form a grout annulus. The key points of the wasteform evolution [33] are summarised in Table 19 for Variant Case1.

Table 19 Wasteform characteristics: Operational ILW - Variant Case 1, 500 litre Drum

Waste stream	Ion exchange resin, operational waste, sludge, evaporator concentrates	Filters
Conditioning proposal	Waste dried and packed in 200 litre drum and then macro-grouted into 500 litre Drum. No capping grout is present.	Filter grouted in 200 litre drum and then macro-grouted into 500 litre Drum. No capping grout is present.
Physical immobilisation	Wastes are not intimately immobilised but surrounded by grout and outer container. Lack of immobilisation would have to be mitigated by physical barrier offered by the grout annulus. Contamination on the external surfaces of 200 litre drum is likely to be immobilised by external grout, but this would require confirmation during wasteform development study.	Wastes likely to be partially immobilised but surrounded by grout and outer container. Based on experience appropriate methods of immobilising filters could be developed (see RWMD guidance on filters [34]). Partial immobilisation would have to be mitigated by physical barrier offered by the grout annulus. Contamination on the external surfaces of 200 litre drum is likely to be immobilised by external grout, but this would require confirmation during wasteform development study.
Mechanical/ physical properties	Mechanical and physical properties of the wasteforms are dependent on the grout annulus. Experience suggests that satisfactory mechanical and physical performance can be achieved.	Intimate grouting of filters within an outer container closely resembles typical wasteforms that have been considered previously and experience suggests that satisfactory mechanical and physical properties can be achieved.
Chemical containment	Although the composition of these wastes may be variable, they are likely to closely resemble those that have been considered previously. The possible presence of cellulosic material may lead to the formation of degradation products with a known ability to complex radionuclides. However, in the context of the overall	The wastes closely resemble those that have been considered previously. Experience suggests they are unlikely to affect chemical containment and will be acceptable

Waste stream	Ion exchange resin, operational waste, sludge, evaporator concentrates	Filters
	amount of cellulosic waste in the UK waste inventory it will represent a small addition and is expected to be acceptable. Long-term degradation of other organic polymers and effect on pH buffering within backfilled vaults has some associated uncertainties. However, presence of grout within the 500 litre drum will mitigate effects on pH. Although the degradation products of the non-cellulosic organic materials may include species that could complex radionuclides, current experience suggests this suggests these are not significant although the nature of any degradation products should be investigated.	
Hazardous materials	No hazardous materials, with the exception of some chemo-toxic elements, are likely to be present in the waste. Where a comparison can be made the amounts of chemotoxic materials are small compared to those already present in the UK inventory.	No hazardous materials, with the exception of some chemo-toxic elements, are likely to be present in the waste. Where a comparison can be made the amounts of chemotoxic materials are small compared to those already present in the UK inventory.
Gas generation	No data provided. The wastes closely resemble those that have been considered previously. Gas generation rate will be influenced by availability of moisture within the waste and further information on proposals for drying waste will be required. Experience suggests gas generation is unlikely to raise significant issues and that the permeabilities of standard waste encapsulation grouts are likely to be sufficient to prevent pressurisation of the wasteform. It is assumed that the 200 litre drum is not a sealed container.	No data provided. The wastes closely resemble those that have been considered previously. Experience suggests gas generation is unlikely to raise significant issues and that the permeabilities of standard waste encapsulation grouts are likely to be sufficient to prevent pressurisation of the wasteform. It is assumed that the 200 litre drum is not a gas-tight container.
Wasteform evolution	In the absence of water, wastes are expected to behave benignly. Wastes may experience accelerated reactions on contact with water in the disposal environment. Although there are current uncertainties over the long-term degradation of ion-exchange resin and consequent effect on wasteform performance, similar wastes have been considered previously as intimately grouted wasteforms and physical separation of these and boron in the evaporator concentrates from the grout annulus by the walls of the 200 litre Drum suggests wasteform evolution will be acceptable. As the grout annulus provides physical containment the effects of evolution on cracking of the annulus will	The filter wasteform closely resembles standard wastes that have been considered previously and experience suggests that wasteform evolution is likely to be acceptable.

Waste stream	Ion exchange resin, operational waste, sludge, evaporator concentrates	Filters
	need to be understood and further information on proposals for drying waste will be required.	

The ability of cement grout to infiltrate cartridge filters and trap particulate activity within the filter elements will require further demonstration although, consistent with RWMD Guidance [35], this is expected to be able to provide satisfactory immobilisation with appropriate formulations. The additional grout annulus will provide considerable protection.

For the other operational wastes the particulates within the body of the waste would only be partially immobilised, and performance will need to rely heavily on the robustness of the surrounding grout annulus and waste container combination. This proposal does not represent common practice in the UK and the acceptability of this packaging concept depends on the ability of the grouted 500 litre Drum container to compensate for the lack of conditioned wasteform. The wasteform assessment has concluded that the necessary performance may be achievable but further evidence would be required. Guidance on non-encapsulated waste has been issued by RWMD (see RWMD Guidance [36] and other guidance listed in Appendix A). It is also noted that, if it were found necessary, full immobilisation might potentially be achievable through application of a conditioning process to the materials inside the 200 litre drums or directly within the 500 litre drums.

Although these proposals represent the smallest volume of packaged waste of the three options considered, nevertheless, scope remains to improve the efficiency of the packaging within the 500 litre drums.

Variant Case 2

For the packages proposed under Variant Case 2, the wasteform proposals are similar to Variant Case 1 insofar as the package comprises unconditioned wastes with protection – in this case by a single barrier, a cast-iron cask, rather than a grout annulus and a thin walled container. The key points of the wasteform evolution [33] are summarised in Table 20 for Variant Case 2.

Table 20 Wasteform characteristics: Operational ILW - Variant Case 2, a Cast-iron Cask

Waste stream	Ion exchange resin, operational waste, sludge, evaporator concentrates	filters
Conditioning proposal	Dried and placed in a cast-iron cask (drying within a type of cast-iron cask has also been carried out overseas)	Assumed by RWMD to be grouted within cast-iron cask.
Physical immobilisation	Wastes are unconditioned and hence are not immobilised. Lack of immobilisation is addressed by the use of the robust cast-iron cask providing physical containment.	Waste is infiltrated with cement grout to form solid product. Extent of infiltration into filter elements not known at this time but, based on experience, with similar wastes this is likely to be acceptable.
Mechanical/ physical properties	Wasteform will not contribute to mechanical or physical properties. Mechanical and physical properties provided by the robust cast-iron cask and expected to be adequate.	Intimate grouting of filters within an outer container closely resembles typical wasteforms that have been considered previously. Regardless of this, robust cast-iron cask expected to be adequate.
Chemical containment	Although the composition of these wastes may be variable, they are likely to closely resemble those that have been considered previously. The possible presence of cellulosic material may lead to the formation of degradation products with a known ability to complex radionuclides. However, in the context of the overall amount of cellulosic waste in the UK waste inventory it will represent a small addition and is expected to be acceptable. Long-term degradation of other organic polymers and effect on pH buffering within backfilled vaults has some associated uncertainties. Although the degradation products of the non-cellulosic organic materials may include species that could complex radionuclides, current experience suggests this suggests these are not significant although the nature of any degradation products should be investigated.	The wastes closely resemble those that have been considered previously. Experience suggests they are unlikely to affect chemical containment and will be acceptable
Hazardous materials	No hazardous materials, with the exception of some chemo-toxic elements, are likely to be present in the waste. Where a comparison can be made the amounts of chemotoxic materials are small compared to those already present in the UK inventory.	No hazardous materials, with the exception of some chemo-toxic elements, are likely to be present in the waste. Where a comparison can be made the amounts of chemotoxic materials are small compared to those already present in the UK inventory.
Gas generation	The wastes closely resemble those that have been considered	The wastes closely resemble those that have been considered previously. Gas

Waste stream	Ion exchange resin, operational waste, sludge, evaporator concentrates	filters
	previously. Gas generation can arise through corrosion, radiolysis and microbial action. The rate of gas generation will be influenced by availability of moisture within the waste and further information on proposals for drying waste will be required. Pressurisation of the sealed cast-iron cask would be a concern to be addressed.	generation can arise through corrosion, radiolysis and microbial action. The rate of gas generation will be influenced by availability of moisture within the wasteform. Pressurisation of the sealed cast-iron cask would be a concern to be addressed.
Wasteform evolution	No conditioned wasteform present. These wastes would not have the benefit of being pre-treated by grouting and therefore initial contact with groundwater could result in more rapid reactions than might be the case for a grouted waste. Any such reactions would be delayed until the cast-iron cask was penetrated.	The filter wasteform closely resembles standard wastes that have been considered previously and experience suggests that wasteform evolution is likely to be acceptable.

In a future LoC assessment, it would need to be established whether the cast-iron cask provides the necessary degree of protection. A further significant issue is the potential for the wasteform to evolve and generate gases which could lead to a pressure rise within the container cavity because a cast-iron cask is a sealed container. Further information on this will be required in future LoC interactions. Nevertheless, it is judged that viable drying processes are currently available and satisfactory packages could be manufactured. Such packages are currently approved for the packaging of certain ILW from light water reactors in Germany.

Decommissioning Wastes

The decommissioning ILW wasteforms (Table 21) exhibit characteristics very similar to other decommissioning waste streams which are already covered by Letters of Compliance. In principle, production of wasteforms with the necessary integrity should be readily achievable. Future LoC interaction with operators will need to confirm corrosion rates for the particular grades of steel, but current expectations by RWMD are that these will be low within a grouted wasteform [33].

Table 21 Wasteform characteristics: Decommissioning Wastes

Waste stream	Reactor vessel – ferritic steel plate	Reactor internals – stainless steel plate
Conditioning proposal	Grouted in 4 metre Box	Grouted in $3m^3$ Box
Physical immobilisation	The majority of the radionuclide inventory is present as activation products distributed through the bulk metal rather than surface contamination and is therefore immobilised within the bulk metal. Waste is also infiltrated with cement grout to form a solid product.	The majority of the radionuclide inventory is present as activation products distributed through the bulk metal rather than surface contamination and is therefore immobilised within the bulk metal. Waste is also infiltrated with cement grout to form a solid product.
Mechanical/ physical properties	Intimate grouting of steel wastes within an outer container closely resembles typical decommissioning wasteforms that have been considered previously and experience suggests that satisfactory mechanical and physical properties can be achieved.	Intimate grouting of steel wastes within an outer container closely resembles typical decommissioning wasteforms that have been considered previously and experience suggests that satisfactory mechanical and physical properties can be achieved.
Chemical containment	Steel is not expected to affect chemical containment.	Steel is not expected to affect chemical containment.
Hazardous materials	No hazardous materials identified and experience suggests steel decommissioning wastes are unlikely to contain such materials or items. All steels contain common elements (e.g. Cr) that contribute to the chemotoxic inventory of the waste inventory.	No hazardous materials identified and experience suggests steel decommissioning wastes are unlikely to contain or such materials or items. All steels contain common elements (e.g. Cr) that contribute to the chemotoxic inventory of the waste inventory.
Gas generation	No data provided. Low corrosion rates expected. Experience suggests that typical corrosion rates and the use of standard permeability grouts in a vented 4 metre Box will be acceptable.	No data provided. Low corrosion rates expected but rates of radiolytic gas generation will be higher than for the reactor vessel steel. Experience suggests that typical gas generation rates and the use of standard permeability grouts in a vented $3m^3$ Box will be acceptable.
Wasteform evolution	No issues are expected. Wasteform expected to evolve in a slow and predictable manner due to expected slow rate of steel corrosion in cement grout.	No issues are expected. Wasteform expected to evolve in a slow and predictable manner due to expected slow rate of steel corrosion in cement grout.

4.1.3 Waste Package Performance

The Waste Package Performance assessments considered the performance of the proposed waste packages under accident conditions [37]. The context of the assessment is specified in RWMD's waste package specification and guidance documentation (WPSGD), as described below.

> For Impact Performance, the waste package should be designed such that in the event of an impact accident:
>
> - releases of radionuclides and other hazardous materials are low and predictable, exhibit progressive release behaviour with increasing impact severity and do not exhibit significant cliff-edge performance characteristics within the anticipated range of impact conditions;
> - both of the barriers to radionuclide release from the waste package (i.e. the waste container and the wasteform) should play an effective role in minimising those releases.
>
> The waste package shall be capable of being dropped, in any attitude, from a height of 0.3 metres onto a flat unyielding surface, whilst retaining its radioactive contents, and remaining suitable for safe handling during all subsequent stages of long-term management. Additionally for the 4 metre Box there shall be no loss of shielding integrity that would result in more than a 20% increase in radiation level at any external surface of the package.
>
> The release of radioactive contents from the waste package, as a result of credible impact accidents during transport and the operational period of a GDF, shall not result in the relevant regulatory radiation protection criteria for workers or members of the public being exceeded.
>
> (This criterion is supported by comprehensive guidance based on the transport and GDF operational safety assessments and includes a table of guidance values for acceptable releases.)

To assess impact accident performance, release fractions have been estimated by combining modelling with existing data on wasteform break-up. The simplified three steps in the analysis were:

- estimating the energy absorbed by the container and hence the wasteform;
- deriving the particulate generated within the wasteform based on small-scale break-up test data from similar or analogue materials;
- estimating the particulate release fraction to the external environment. In the absence of a detailed design, it could be pessimistically assumed that all of the particulate would be released. A recent impact evaluation of a 500 litre Drum applied an overall factor of 0.3 for the lid edge orientation. Therefore based on good engineering and design an improved overall factor could be applied for the retention and it was proposed that for this work to apply a factor of 0.1.

RWMD's waste package specification and guidance documentation similarly sets expectations for performance of waste packages under fire conditions.

> For Fire Performance, the waste package should be designed such that in the event of a fire accident:
>
> - releases of radionuclides and other hazardous materials are low and predictable, exhibit progressive release behaviour with increasing fire severity

and do not exhibit significant cliff-edge performance characteristics within the anticipated range of fire conditions;

- both of the barriers to radionuclide release from the waste package (i.e. the waste container and the wasteform) should play an effective role in minimising those releases.

The release of radioactive contents from the waste package, as a result of credible fire accidents during transport and the operational period of a GDF, shall not result in the relevant regulatory radiation protection criteria for workers or members of the public being exceeded.

(This criterion is supported by comprehensive guidance based on the transport and GDF operational safety assessments and includes a table of guidance values for acceptable releases.)

For fire accident performance, release fractions were estimated using existing thermal modelling to estimate the temperature profiles in the waste package and hence to determine the fractions of various radionuclides that would be released at those temperatures.

As has been noted previously some of the proposed waste package types are new to the UK as disposal packages and where this is the case the assessment has progressed by the use of analogous data from similar containers and/or wasteforms. In most cases RWMD has been able to draw initial conclusions and estimate the fraction of the activity that may be released in an accident (the release fraction). This has been used within the subsequent safety assessment sections - Section 4.2.2 and 4.2.3.

Using these methods, impact and fire accident release fractions were estimated for all of the waste packages proposed in the Reference Case (C1 and C4 Casks), Variant Case 1 (500 litre Drums), and for the decommissioning ILW (4 metre Boxes and $3m^3$ Boxes) [37]. For Variant Case 2 (cast-iron cask), insufficient information was available for estimation of quantified release fractions.

In the following paragraphs each of the proposed waste package/ wasteform combinations are compared against the above criteria and results presented in Tables 22 to 29. It should be noted that the values for calculated dose given in Table 22 to 29 have been used to test the potential acceptability of the proposed packages in advance of the full transport and operational safety assessment calculations which are reported in Sections 4.2.2 and 4.2.3 respectively.

Operational ILW - Reference Case

C1 and C4 waste packages have yet to be investigated for use in the UK as disposal packages and data to demonstrate performance against the RWMD waste package specification criteria are not readily available.

In advance of design drawings, testing and modelling to provide estimates of performance under impact and fire accident conditions, the RWMD evaluation followed the approach as described above using the descriptions supplied by EdF/Areva and supplementing this where possible by the use of UK generic test and modelling results for similar type 2 Industrial Packages (the 4 metre Box) noting that there are significant differences in the approaches to the two designs. The estimated impact release fractions were based on modelling combined with break-up data for ion exchange resins in grout, sludges in grout and metallic wastes in grout. The estimated fire accident release fractions were based on analogous small-scale furnace release fraction data for ion exchange resins in grout, sludges in grout and metallic wastes in grout. The evaluation is summarised in Tables 22 and 23. Further work, based on specific waste package designs and proposals for wasteforms, could be required at subsequent LoC stages to inform transport and disposal facility safety cases.

Table 22 Waste package impact performance: Operational ILW - Reference Case, C1/C4 casks

Waste stream	Ion exchange resin conditioned in epoxy resin	Filters and Sludges conditioned with cement grout	Operational waste compacted and cement grouted
Release of radionuclides and hazardous materials are low and predictable, exhibit progressive release behaviour and no cliff edges	Yes, based on impact modelling of analogous 4 metre Box waste package which has similar shielding features. The C1 and C4 Casks in most cases have an inner liner which could be considered an “inside-out” 4 metre Box. [The WAGR Box was also considered and is supported by full-scale supporting drop test data. The steel collars top and bottom make this less relevant however.]	Yes, based on impact modelling of analogous 4 metre Box waste package which has similar shielding features. The C1 and C4 Casks in most cases have an inner liner which could be considered an inside-out 4 metre Box. [The WAGR Box was also considered and is supported by full-scale supporting drop test data. The steel collars top and bottom make this less relevant however.]	Yes, based on impact modelling of analogous 4 metre Box waste package which has similar shielding features. The C1 and C4 Casks in most cases have an inner liner which could be considered an inside-out 4 metre Box. [The WAGR Box was also considered and is supported by full-scale supporting drop test data. The steel collars top and bottom make this less relevant however.]
Both barriers play effective role in minimising releases	Yes, based on modelling of analogous 4 metre Box waste package combined with break-up data for Ion exchange resin in polymer. Early drop tests of the WAGR Box raised concerns at the quantity of spalled material from the surface hence it was decided to add steel collars top and bottom. Similar to the inside-out analogy above, the 4 metre Box was considered the more appropriate analogy, although further work would be required to confirm that there is adequate shielding following an impact accident.	Yes, based on modelling of analogous 4 metre Box waste package combined with break-up data for sludge immobilised in grout.	Yes, based on modelling of analogous 4 metre Box waste package combined with break-up data for solid waste immobilised in grout.
Capable of being dropped from 0.3m with no release	Yes, licensed as IP2 transport package in France.	Yes, licensed as IP2 transport package in France.	Yes, licensed as IP2 transport package in France.

Waste stream	Ion exchange resin conditioned in epoxy resin	Filters and Sludges conditioned with cement grout	Operational waste compacted and cement grouted
Capable of being dropped from 0.3m with less than 20% increase in external dose. [Industrial Packages only]	Yes, licensed as IP2 transport package in France.	Yes, licensed as IP2 transport package in France.	Yes, licensed as IP2 transport package in France.
Activity release consistent with regulatory dose criteria for workers and members of the public	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 in WPS/730. Predicted maximum dose of 0.01 mSv.	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 in WPS/730. Predicted maximum dose of 0.01 mSv.	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 in WPS/730. Predicted maximum dose of 0.01 mSv

Table 23 Waste package fire performance: Operational ILW - Reference Case, C1/C4 casks

Waste stream	Ion exchange resin conditioned in epoxy resin	Filters and Sludges conditioned with cement grout	Operational waste compacted and cement grouted
Release of radionuclides and hazardous materials are low and predictable, exhibit progressive performance and no cliff-edges	Yes, based on thermal modelling of 4 metre Box, it is predicted that releases will be low and predictable. The outer steel container is a good conductor, but the 200 mm concrete shielding dominates the thermal analysis as it provides a significant barrier to heat transfer. Similarly the C1 and C4 Cask thermal performance is expected to be dominated by the 150 mm concrete.	Yes, based on thermal modelling of 4 metre Box, it is predicted that releases will be low and predictable. The outer steel container is a good conductor, but the 200 mm concrete shielding dominates the thermal analysis as it provides a significant barrier to heat transfer. Similarly the C1 and C4 Cask thermal performance is expected to be dominated by the 150 mm concrete.	Yes, based on thermal modelling of 4 metre Box, it is predicted that releases will be low and predictable. The outer steel container is a good conductor, but the 200 mm concrete shielding dominates the thermal analysis as it provides a significant barrier to heat transfer. Similarly the C1 and C4 Cask thermal performance is expected to be dominated by the 150 mm concrete.
Both barriers play effective role in minimising releases	Yes, based on modelling of analogous 4 metre Box waste package (to confirm the low expected temperatures in the wasteform) combined with small-scale furnace test data	Yes, based on modelling of analogous 4 metre Box waste package (to confirm the low expected temperatures in the wasteform) combined with small-scale furnace test data	Yes, based on modelling of analogous 4 metre Box waste package (to confirm the low expected temperatures in the wasteform) combined with small-scale furnace test data on releases from metal

	on releases from Ion exchange resin conditioned in polymer.	on releases from sludges in grout.	in grout.
Activity release consistent with regulatory dose criteria for workers and members of the public	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 in WPS/730. Predicted maximum dose of 0.02 mSv.	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 in WPS/730. Predicted maximum dose of 0.02 mSv.	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 in WPS/730. Predicted maximum dose of 0.02 mSv.

Operational ILW – Variant Case 1

It is proposed that most of the operational ILW would not be intimately immobilised in the 500 litre Drums, instead the wastes would be packed unimmobilised into smaller containers (200 litre drums) which would be grout enclosed within the 500 litre Drums. As it has been discussed in Section 4.1.2 (Wasteform), the lack of a conditioned wasteform would need to be compensated for by the provision of a robust waste container. In the case of Variant Case 1 when considering impact and fire accident performance, this is achieved by the provision of a protective grout annulus around the 200 litre drum and unconditioned waste.

In advance of design drawings, testing and modelling to provide estimates of performance under impact and fire accident conditions, the RWMD evaluation followed the same approach as described for the Reference Case using the descriptions supplied by EdF/Areva and supplemented this where possible by the use of UK generic test and modelling results for similar "annular grouted" waste packages but modified as appropriate to reflect the unimmobilised nature of the wasteform. For future submissions further work would be required to refine the estimated impact release fractions. The estimated fire accident release fractions were derived from analogies with UK wasteforms based on supercompacted sludges and heterogeneous metallic wastes in annular grouted waste packages. The evaluation is summarised in Tables 24 and 25.

Table 24 Waste package impact performance: Operational ILW – Variant Case 1, 500 litre Drums

Waste stream	Ion exchange resin, Sludges, Operational waste and Evaporator concentrates	Filters entombed with cement grout
Release of radionuclides and hazardous materials are low and predictable, exhibit progressive release behaviour and no cliff edges	Yes, based on impact modelling of annular grouted 500 litre Drum waste package, although there could be a cliff-edge effect if the impact was severe enough to cause a significant breach of the outer containment. To account for the uncertainties in the performance of the wasteform under these conditions the predicted release fraction was increased by 2 orders of magnitude. This should be investigated in further	Yes. This initial work is based on impact modelling of annular grouted 500 litre Drum waste package. It is recognised that there are multiple barriers to the unimmobilised waste: the release of radionuclide from outer steel container and the grout annulus. It is expected that the outer grout and filter housing will provide a further barrier to the release of activity. Without a design of filter housing and grout annulus it was not possible to consider these

Waste stream	Ion exchange resin, Sludges, Operational waste and Evaporator concentrates	Filters entombed with cement grout
	modelling as the unconditioned wasteform does not provide the same physical internal support to the waste container as demonstrated from previous models and drop tests of annular grouted 500 litre Drums with cement immobilised wasteform.	features specifically in this evaluation. To account for the uncertainties in the performance of the wasteform under these conditions the predicted release fraction was increased by 2 orders of magnitude. This should be investigated in further modelling. Overall it is unlikely that a breach through all these barriers would permit a pathway for unimmobilised particulate to the external environment.
Both barriers play effective role in minimising releases	Only the 500 litre Drum container and annulus contribute to containment in this case. Further work would be required to determine the effectiveness of the container and if there is any contribution provided by the unimmobilised waste itself.	The annular grouted 500 litre Drum will absorb most of the impact energy and provide most of the containment. Specific modelling to include the internal furniture, surrounding grout and filter housing could provide better estimate of performance.
Capable of being dropped from 0.3m with no release	Yes, based on impact modelling of annular grouted 500 litre Drum waste package.	Yes, based on impact modelling of annular grouted 500 litre Drum waste package.
Capable of being dropped from 0.3m with less than 20% increase in external dose. [Industrial Packages only]	N/A	N/A
Activity release consistent with regulatory dose criteria to workers and members of the public	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 in WPS/700. Predicted maximum dose of 17.4 mSv.	Yes. Scoping calculations indicate higher than guidance values of Table 2 in WPS/700. (See the note following this table.) Predicted maximum dose of 113 mSv (EPR12).

Note: For EPR12, the wasteform items are spent cartridge filters surrounded by grout and therefore most of the activity would be expected to be entombed within the waste package. All the three high release radionuclides that contribute to the release for EPR12 are likely to be associated with spent fuel: Pu-238, Pu-241 and Cm-244. The above predicted doses (mSv) can be significantly reduced if the waste can be quantified in terms of the geometry of the wasteform barriers (spent cartridge filter housing and surrounding grout) and if the material trapped in the spent cartridge filters is better characterised.

Table 25 Waste package fire performance: Operational ILW – Variant Case 1, 500 litre Drums

Waste stream	Ion exchange resin, Sludges, Operational waste and Evaporator concentrates	Filters entombed with cement grout
Release of radionuclides and hazardous materials are low and predictable, exhibit progressive performance and no cliff-edges	Yes, based on thermal modelling of annular grouted 500 litre Drum waste package it is predicted that releases will be low and predictable with the annulus providing a significant barrier to heat transfer.	Yes, based on thermal modelling of annular grouted 500 litre Drum waste package it is predicted that releases will be low and predictable with the annulus providing a significant barrier to heat transfer.
Both barriers play effective role in minimising releases	Yes, based on thermal modelling of annular grouted 500 litre Drum waste package combined with small-scale furnace test data on releases from ungrouted metal or dried compacted sludge.	Yes, based on thermal modelling of annular grouted 500 litre Drum waste package combined with small-scale furnace test data on releases from dried compacted sludge.
Activity release consistent with regulatory dose criteria for workers and members of the public	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 in WPS/700. Predicted maximum dose of 0.01 mSv.	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 in WPS/700. Predicted maximum dose of 0.02 mSv.

Operational ILW – Variant 2

Cast-iron casks have only recently been proposed for use in the UK. Accident consequences have yet to be assessed for the UK faults and hazards for these packages and hence issues have only been identified in outline regarding the impact and fire accident performance. Tables 26 and 27 explain waste package performance under impact and fire accidents respectively for Variant Case 2. Further work is required to define the waste package design and performance before release factions can be proposed with any confidence.

Table 26 Waste package impact performance: Operational ILW – Variant Case 2, Cast-iron Cask

Waste stream	Ion exchange resin, Sludges, Operational waste and Evaporator concentrates dried in Cask	Filter grouted in Cask
Release of radionuclides and hazardous materials are low and predictable, exhibit progressive release behaviour and no cliff-edge effects	Due to the robust construction of the cast-iron cask releases are expected to be zero or close to zero for impacts up to 9m. This is accepted by German regulator for the use of such containers for selected wastes in Germany.	Due to the robust construction of the cast-iron cask, releases are expected to be zero or close to zero for impacts up to 9m. This is accepted by German regulator for the use of such containers for selected wastes in Germany. Beyond that height the safety

Waste stream	Ion exchange resin, Sludges, Operational waste and Evaporator concentrates dried in Cask	Filter grouted in Cask
		will be based on the performance of the grout encapsulating the filter. Previous experience with encapsulation of filters in 500 litre Drums has optimised on the use of polymer to ensure void filling and a good overall wasteform.
Both barriers play effective role in minimising releases	Only the container contributes to containment in this case.	The container contributes the main containment in this case. It is expected that the filter housing will provide a further barrier to release of the waste.
Capable of being dropped from 0.3m with no release	Yes. Cask is licensed as Type 2 Industrial Package in Germany.	Yes. Cask is expected to comply with the requirements of Type 2 Industrial Package.
Capable of being dropped from 0.3m with less than 20% increase in external dose [Industrial Packages only]	Yes. Cask is licensed as IP-2 in Germany (as well as Type B).	Yes. Cask is licensed as IP-2 in Germany (as well as Type B).
Activity release consistent with regulatory dose criteria for workers and members of the public	Insufficient data supplied to determine release fractions and hence to determine doses.	Insufficient data supplied to determine release fractions and hence to determine doses.

Table 27 Waste package fire performance: Operational ILW – Variant Case 2, Cast-iron Cask

Waste stream	Ion exchange resin, Sludges, Operational waste and Evaporator concentrates dried in Cask	Filter grouted in Cask
Release of radionuclides and hazardous materials are low and predictable, exhibit progressive performance and no cliff-edges	Due to the robust construction of the cast-iron cask releases are expected to be low and predictable and are expected to increase in a progressive fashion. Dried wasteforms will behave in a benign manner. As the temperature of the wasteform increases there will be little driving force (e.g. steam) to promote mobility of radioactivity.	Due to the robust construction of the cast-iron cask releases are expected to be low and predictable and are expected to increase in a progressive fashion. For longer duration fires, pressurisation (from steam generation) and seal performance are potential issues.
Both barriers play effective role in minimising releases	The seal may be compromised at elevated temperatures, but it is likely that pressure build-up within	The seal may be compromised at elevated temperatures and there may be internal pressure

Waste stream	Ion exchange resin, Sludges, Operational waste and Evaporator concentrates dried in Cask	Filter grouted in Cask
	the cavity will be retained as there will be little addition from water vapour.	arising from water vapour during drying of the wasteform.
Activity release consistent with regulatory dose criteria for workers and members of the public	Insufficient data supplied to determine release fractions and hence to determine doses.	Insufficient data supplied to determine release fractions and hence to determine doses.

Decommissioning ILW

These metallic wastes are clearly defined and will be directly immobilised in standard waste packages. The waste containers are well-known and there has been extensive testing of the mechanical and thermal performance of metal in grout wasteforms.

In advance of design drawings, testing and modelling to provide estimates of performance under impact and fire accident conditions, the RWMD evaluation followed the approach described previously using the descriptions supplied by EdF/Areva and supplemented by the use of UK generic test data and modelling results for similar generic decommissioning waste packages. For impact performance measured release fraction data for break-up of metal in grout wasteforms was applied. For fire accident performance the estimated release fractions were based on the measurements of releases from active small-scale metal in grout samples when heated at a range of temperatures in a furnace. The evaluation is summarised in Tables 28 and 29. Specific modelling of the waste items within the container would be required to improve on these assumptions in support of future LoC submissions.

Table 28 Waste package impact performance: Decommissioning ILW

Waste stream	Reactor vessel (ferritic steel plate) grouted in 4 metre Box	Reactor internals (stainless steel plate) grouted in $3m^3$ Box
Release of radionuclides and hazardous materials are low and predictable, exhibit progressive release behaviour and no cliff-edge effects	Yes, based on impact modelling of 4 metre Box waste package.	Yes, based on impact modelling of $3m^3$ Box waste package.
Both barriers play effective role in minimising releases	Yes, based on modelling of 4 metre Box waste package combined with break-up data for metal in grout.	Yes, based on modelling of $3m^3$ Box waste package combined with break-up data for metal in grout.
Capable of being dropped from 0.3m with no release	Yes, is expected to comply with the requirements of Type 2 Industrial Package.	Yes, based on impact modelling of $3m^3$ Box waste package.
Capable of being dropped from 0.3m with less than 20% increase in external dose [Industrial Packages only]	Yes, is expected to comply with the requirements of Type 2 Industrial Package.	N/A

Waste stream	Reactor vessel (ferritic steel plate) grouted in 4 metre Box	Reactor internals (stainless steel plate) grouted in $3m^3$ Box
Activity release consistent with regulatory dose criteria for workers and members of the public	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 WPS/730. Predicted maximum dose of 0.1 mSv.	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 WPS/710. Predicted maximum dose of 0.63 mSv.

Table 29 Waste package fire performance: Decommissioning ILW

Waste stream	Reactor vessel (ferritic steel plate) grouted in 4 metre Box	Reactor internals (stainless steel plate) grouted in $3m^3$ Box
Release of radionuclides and hazardous materials are low and predictable, exhibit progressive performance and no cliff-edges	Yes, based on thermal modelling of a generic 4 metre Box, it is predicted that releases will be low and predictable with the concrete wall providing a significant barrier to heat transfer.	Based on thermal modelling of a generic $3m^3$ Box, it is predicted that releases will be low and predictable. Characterisation of the waste and the development of internal furniture to locate the waste would provide further barriers to the release of activity from the waste package. (See note following this table.)
Both barriers play effective role in minimising releases	Yes, based on modelling of a generic 4 metre Box waste package combined with small-scale furnace test data on releases from metal in grout.	Yes, based on modelling of a generic $3m^3$ Box waste package combined with small-scale furnace test data on releases from metal in grout.
Activity release consistent with regulatory dose criteria for workers and members of the public	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 in WPS/730. Predicted maximum dose of 2.2 mSv.	Scoping calculations indicate higher than guidance values of Table 2 in WPS/710. (See the note following this table.) Predicted maximum dose of 10.8 mSv (EPR08).

Note: In common with many other UK decommissioning wastes, these items are activated steels and therefore most of the activity would be expected to be embedded in the waste rather than readily accessible as surface contamination and surface corrosion. All the three high release radionuclides that contribute to the release for EPR08 have chemical forms that are potentially very volatile (i.e. can form gaseous compounds): Cl-36, Se-79 and C-14. The above predicted doses (mSv) can be significantly reduced if the waste can be quantified in terms of the large fraction of activity that is locked within the steel matrix and if the actual chemical forms of the high volatile radionuclides are identified and applied to the release calculations with less volatile characteristics.

4.2 Disposal System Issues

4.2.1 Impact on disposal facility design

Context

The GDA Disposability Assessment for the EPR has considered implications for GDF design of disposing of ILW from an EPR, and the scale of the impact of the additional ILW from operation and decommissioning of an EPR on the projection of the GDF area on the land surface (the "footprint"). This analysis is presented for the three waste packaging options, and is based on the ILW GDF design presented by RWMD in [38]. It should be noted that this generic design is subject to update to be consistent with the revised "baseline inventory" identified in the Managing Radioactive Waste Safely (MRWS) White Paper on implementation of geological disposal [39]. As the MRWS process progresses RWMD will develop designs based on information relevant to specific sites and settings

Results and Implications

The evaluation of design impact [40] assumed that operational ILW would be emplaced in unshielded ILW (UILW) vaults, decommissioning ILW packaged in 4 metre Boxes would be emplaced in shielded ILW (SILW) vaults, and decommissioning ILW packaged in $3m^3$ Boxes emplaced in UILW vaults. It is recognised that there is an option whereby the concrete C1 and C4 Casks and cast-iron casks are routed for emplacement in the SILW vaults (these are self-shielded packages and the external dose rate from the packages is likely to be sufficiently low for disposal to the SILW vaults). However, this option has not been explored and the Design Impact assessment has progressed on the assumptions above. In the event that a GDF were to be presented with C1 and C4 Casks and/or cast-iron casks by future operators, then further consideration of optimisation of emplacement options would be considered. For present purposes routing assumptions will have little impact on the footprint, as the UILW and SILW vaults have similar cross-sections.

As has been noted previously C1 and C4 concrete casks and cast-iron casks have not been considered previously in GDF design studies. In order to receive and dispose of C1 and C4 Casks or cast-iron casks, surface receipt and underground transfer and buffer storage facilities would need to be fitted with additional waste package handling equipment, including cranes and transport wagons to allow for the different dimensions and design of lifting features incorporated into the containers [40]. Such modifications to the waste package handling equipment would be readily achievable. Disposal of these packages may require modifications to the spacing of waste package stacks currently envisaged.

The fractional change in the footprint area of the GDF, as compared to the area required for the disposal of legacy ILW has been determined for the three waste packaging options. In all cases the volumes of ILW generated by the operation of an EPR are small compared to the volume of legacy ILW. Depending on the assumed packaging approach, operation of a single EPR would require [40]:

- for the Reference Case, an additional length of UILW vault of approximately 60 m, which is equivalent to approximately 20% of one vault, and an additional SILW vault length of 3m;
- for Variant Case 1, an additional length of UILW vault of approximately 30m, which is equivalent to approximately 10% of one vault, and an additional SILW vault length of 3m;

- for Variant Case 2, an additional length of UILW vault of approximately 50 m, which is equivalent to approximately 17% of one vault, and an additional SILW vault length of 3m.

Therefore, in all cases the necessary increase in the area is small, corresponding to a maximum of approximately 60m of vault length for each EPR. This represents approximately 1% of the area required for the legacy ILW, per reactor, and less than 10% for the illustrative fleet of six reactors.

4.2.2 Transport safety

Context

RWMD is planning the transport infrastructure necessary to allow ILW to be delivered from sites of arising to a GDF. This includes development of transport container concepts which will enable packaged wastes to be transported to a GDF in full compliance with IAEA regulations for the Safe Transport of Radioactive Material [41], as incorporated into UK transport legislation. In support of this work RWMD has produced a Generic Transport Safety Assessment (GTSA) and this is routinely used within the Letter of Compliance process to check that proposed waste packages are compliant with transport plans and do not compromise the generic safety case.

The generic transport infrastructure and associated safety case recognises two general classes of transport:

- 500 litre Drums, $3m^3$ Boxes and $3m^3$ Drums transported within a reusable and shielded transport container referred to as the Standard Waste Transport Container (SWTC). The SWTC provides shielding and containment for compliance with transport legislation as a Type B package;
- 4 metre and 2 metre Boxes transported as transport packages in their own right. These packages are designed to meet the requirements of a Type 2 Industrial Package (IP-2).

Proposals for transport of operational and decommissioning ILW for the EPR have been tested following the above approach. The transport safety assessment has addressed [42]:

- transport of Reference Case C1 and C4 concrete casks as IP-2 Packages;
- transport of Variant Case 1 500 litre Drums in SWTCs as Type B Packages;
- transport of Variant Case 2 cast-iron casks as Type B Packages;
- transport of decommissioning ILW in 4 metre Boxes as IP-2 Packages;
- transport of decommissioning ILW in $3m^3$ Boxes in SWTCs as Type B Packages.

For the Transport Safety assessment, it was not necessary to consider all waste streams and all packaging options. Instead, a screening process was devised to identify bounding and representative waste packages for more detailed consideration [42]. Waste packages were screened using estimated release fractions and A_2 content to identify bounding cases. A bounding case was selected for a representative of each type of container proposed (except cast-iron casks, which, as explained previously, were not subject to detailed evaluation). Selected waste packages were [42]:

- C1 Cask: EPR01 Ion exchange resins conditioned in epoxy resin;
- C4 Cask: EPR02 Spent cartridge filters conditioned in cement grout;

- 500 litre Drum: EPR11 Ion exchange resins placed unconditioned in a 200 litre Drum which would subsequently be placed in a UK standard stainless steel 500 litre Drum with an annular grout lining assumed to be 100 mm thick;
- 500 litre Drum: EPR12 Spent cartridge filters grouted into a 200 litre Drum which would subsequently be placed in a UK standard stainless steel 500 litre Drum with an annular grout lining assumed to be 100 mm thick;
- 4 metre Boxes: EPR06 decommissioning reactor vessel conditioned in cement grout;
- 3m^3 Boxes: EPR08 decommissioning lower reactor internals conditioned in cement grout.

In addition, for all waste packages assumed to be transported as IP-2 transport packages, an assessment has been undertaken to compare wasteform characteristics with the definition of low-specific activity (LSA) material provided in the IAEA Regulations for the Safe Transport of Radioactive Material [41].

Results and Implications

A range of issues have been identified through the transport assessment [42] and are discussed below. These are principally related to the assumptions regarding the maximum package inventories and management of these inventories during packaging, and RWMD expect that these issues would be considered in a future Letter of Compliance interaction with the operators.

Reference Case

The concrete casks are licensed for the transport of selected wastes from existing PWRs in France. This provides confidence that equivalent wastes from an EPR potentially could be accommodated using the same containers. Furthermore, RWMD has judged that it should be feasible to develop design concepts for the transport of packages based on such containers to the GDF.

Although it may be anticipated that many wastes from an EPR would be equivalent to those from existing PWRs, confirmation and/or identification of potential issues has been sought by comparing the EPR assessment inventory with the requirements for transport of waste packages [42]. This comparison suggests that, in some cases, the estimated external dose rates for packages containing the current maximum inventory would exceed regulatory limits by a factor of up to two. Nevertheless, it is also recognised that a number of approaches would be available to ensure compliance in practice. These include refinement of the assessment inventory, management of waste loading, introduction of additional shielding, decay storage and, ultimately, management of reactor operations to reduce the activity of the waste (for example through more frequent removal of filters from the reactor circuit).

RWMD has judged that, in light of the ready availability of potential mitigations and the moderate factor by which the limits may be exceeded in a few cases, it is appropriate to conclude that the reference case packages could be transported. Further development of the necessary arguments would be expected as part of a future submission under the LoC process.

Variant Case 1

The proposal under Variant Case 1 to use standard RWMD waste containers, in conjunction with shielded transport over-pack, provides compliance with existing RWMD standards and specifications for waste packages which have been developed to be consistent with transport regulations. [41].

The use of 500 litre Drums transported in a SWTC as a Type B package in Variant Case 1 provides a robust transport solution with low external dose rates and added protection for impact and fire faults. For this scenario, information will be needed in a future LoC interaction demonstrating that the proposed waste packages meet the requirements of the transport package safety case; this will include data on the potential release rates of radioactive and other gases from the package.

Variant Case 2

The use of the cast-iron casks transported as Type B package in Variant 2 has not been fully evaluated due to the reduced level of information available to support this packaging option. However, it is known that these containers, with their associated impact limiters, are licensed for use in Germany as IAEA Type B containers for certain waste types and this gives confidence that there should not be any major transport safety issues that cannot be resolved as part of a future LoC interaction. Successful licensing of the cast-iron casks would require demonstration that gas pressurisation would not be a problem during transport by ensuring that the waste was dried before packaging and this practice is currently followed in Germany.

Decommissioning ILW

The proposed decommissioning ILW packages comprise metal items immobilised into standard containers using a cement grout. These proposals conform to existing practices for decommissioning wastes in the UK and are expected to produce packages that would be compliant with existing RWMD standards and specifications. The current maximum assessment inventory for the decommissioning ILW proposed to be packaged in 4 metre Boxes challenges some aspects of the transport regulations in relation to dose-rates but it is judged that this issue could be addressed by refining the assessment inventory, modifying the proposals to include additional shielding, management of waste loading or employing containers that necessitate the use of a shielded overpack for transport (i.e. the $3m^3$ Box proposed for the remainder of the decommissioning ILW) [42].

The 4 metre Box packaging option for decommissioning ILW (EPR06) is likely to raise few issues for transport safety. The 4 metre Box concept allows for the internal concrete shielding to be varied in thickness to suit the inventory of waste being carried. It is likely that concrete shielding beyond the 100 mm assumed might be required.

The $3m^3$ Box packaging option for decommissioning ILW (EPR07 and EPR08) again provides a robust packaging solution for the transport of such wastes and is likely to present a transport package that meets IAEA transport regulation requirements. The assessment has identified that the maximum package inventory could challenge this conclusion in respect of the release of gaseous radionuclides tritium and carbon-14. The pessimistic assumptions used to estimate the inventory of the radionuclides (discussed earlier in Section 3.3.3) leads to the transport package safety case contents activity limit [43] being exceeded for tritium (tritium activity is 110% of limit) and being a significant factor of the limit for carbon-14 (carbon-14 activity is 66% of limit). It is thought that this issue will be resolved by removal of pessimisms from the inventory determination and should be considered as part of future LoC interactions.

Criticality Safety

IAEA regulations for the safe transport of radioactive materials [41] specify that if the total mass of fissile materials is less than 15g, the waste package can be classified as 'fissile excepted' and not subject to further criticality safety requirements. The maximum quantity of fissile material in any of the ILW packages is 0.6g in EPR12. Given the nature of the wastes, such estimated low levels of fissile material seem appropriate. Should these quantities be confirmed, all types of transport package for operational ILW and decommissioning ILW would be fissile excepted, and would not require further criticality assessment.

The IAEA regulations also specify requirements on the masses of deuterium and beryllium in packages containing fissile excepted material. These requirements depend on the average hydrogen density of the wastes and the type of fissile material present. The limiting requirement is that the masses of deuterium and beryllium in the package are both less than 1.8g [41]. No information was available regarding the expected masses of deuterium and beryllium in the waste packages, but neither is expected to be present in significant quantities. In any future submission under the LoC process, the operator will need to confirm that deuterium and beryllium are not present in significant quantities in ILW from an EPR.

Risks

The impact these wastes would have in addition to the transport movements required for legacy wastes was considered by application of the Transport Safety Assessment Toolkit (TranSAT). In all cases only small increases to the routine risk to the public and to the worst case individual were noted.

Summary

In summary, the operational and decommissioning ILW from an EPR is considered to be compatible with the requirements for transport as expressed by the IAEA transport regulations. Some minor issues have been identified in the Transport Safety assessment, but these are considered to be matters for clarification, and can be managed through more realistic estimation of package inventories and would be taken forward by interaction with operators through the Letter of Compliance process.

4.2.3 Operational Safety

Context

The GDF work being undertaken by RWMD is supported by a Generic Operational Safety Assessment (GOSA). This is routinely used within the Letter of Compliance process to test proposed waste packages and to check compliance with assumed performance and accident consequence criteria. A similar approach has been adopted for the EPR waste disposability assessment.

When ILW packages arrive on the GDF site they are assumed to be subject to acceptance checks and dispatched underground using the onsite transportation system. Packages arriving in the SWTC will be routed to an inlet cell where operations to unload the SWTC are completed and the 3m^3 Box or stillage of four 500 litre Drums transferred to the emplacement location in the disposal vault. IP-2 packages such as the 4 metre Box are similarly routed underground but directed to a buffer store area awaiting a campaign of emplacement in a separate vault.

In the case of EPR wastes in non-standard packages it was necessary to consider where they should be routed. For the purposes of Design Impact described in Section 4.2.1, it was assumed that the non-standard containers would be routed to the unshielded ILW (UILW) vaults, but, for the purposes of exploring operational safety issues, it was concluded that it would be more instructive to assume that they would be routed for emplacement in shielded ILW (SILW) vaults where they may be directly accessible to GDF workers. In the event that such containers are preferred by potential operators, then RWMD would need to identify an optimum emplacement approach based on ALARP considerations. The operational safety assessment was based on the following operations [44]:

- Reference Case C1 and C4 concrete casks are assumed to be routed directly for handling and emplacement operations in SILW vaults;

- Variant Case 1 500 litre Drums are assumed to be routed via the Inlet Cell, unloaded from the SWTC and emplaced in UILW vaults within a four-drum stillage;

- Variant Case 2 cast-iron casks are assumed to be routed to the SILW vaults although it would be necessary to first remove any impact limiters which had been applied for the transport journey;

- Decommissioning ILW in $3m^3$ Boxes would be handled as described above for Variant Case 1, and decommissioning ILW in 4 metre Boxes would be routed directly for emplacement in SILW vaults.

The same approach to definition of representative and bounding waste packages as described previously for the Transport Safety assessment (Section 4.2.2) was applied in the Operational Safety assessment.

The GOSA is supported by a fault and hazard schedule which is routinely used within the LoC process to check the performance of packages if subjected to the postulated accidents. This is achieved by use of the Repository Operational Safety Assessment (ROSA) toolkit which is used to assess on-site and off-site doses for a range of design basis faults.

For EPR wastes, package performance data and consequential release fractions have been combined in the toolkit with waste stream inventories to estimate dose consequences for a range of fault sequences [44]. The estimated doses have then been compared to targets for design basis fault sequence mitigated doses currently being considered by RWMD. These targets are reproduced in Table 30.

Table 30 Targets for design basis fault sequence mitigated doses used in the EPR Operational Safety Assessment

Location	Basic Safety Level (BSL)	Basic Safety Objective (BSO)
On-Site	20 mSv for initiating fault frequencies > 10^{-3} per annum 200 mSv for initiating fault frequencies between 10^{-3} and 10^{-4} per annum 500 mSv for initiating fault frequencies < 10^{-4} per annum	0.1 mSv
Off-Site	1 mSv for initiating fault frequencies > 10^{-3} per annum 10 mSv for initiating fault frequencies between 10^{-3} and 10^{-4} per annum 100 mSv for initiating fault frequencies < 10^{-4} per annum	0.01 mSv

Results and Implications

Assessment of Design Basis Faults

The results of the Repository Operational Safety Assessment (ROSA) toolkit assessments are summarised here in terms of the waste type, based on the discussion in the EPR Operational Safety assessment [44]:

- operational ILW: Reference Case
  - (Off-site public) protected doses are all below the most stringent BSL (1 mSv), meaning that there is confidence that an operational safety case can be made for all assessed packaging options from the point of view of accidental doses to the public during the disposal facility's operational period.
  - (On-site worker) protected doses for one accident were estimated to be above the most stringent BSL of 20 mSv (estimated dose 22.2 mSv). This fault is an event involving crane collapse onto a single waste package (C1 Cask containing EPR02 spent cartridge filters). The dose calculated is considered to be justifiable and ALARP based a judgement of the expected frequency of crane collapse given the available information. All other protected doses were below the most stringent BSL.
- operational ILW: Variant Case 1
  - (Off-site public) protected doses are all below the most stringent BSL (1 mSv), meaning that there is confidence that an operational safety case can be made for all assessed packaging options from the point of view of accidental doses to the public during the disposal facility's operational period.
  - (On-site worker) protected doses for all accidents involving a single waste container, are below the most stringent BSL (20 mSv). A few severe impact faults involving multiple 500 litre Drum stillages (waste streams EPR11 and EPR12), each containing 4 drums loaded with the maximum radionuclide inventory, give doses above the most stringent BSL (maximum dose 85 mSv, associated with crane collapse fault). These events are judged to be severe events that would have frequencies much less than 10^{-3} per year in a modern standards facility, and can be accepted at this stage, given that further efforts to safeguard against such events will continue as the GDF design develops.
- operational ILW: Variant Case 2:
  - Quantitative assessment of Variant Case 2 was not carried out due to the lack of information regarding release fractions for the cast-iron casks [44]. In any future submission for packaging in cast-iron casks, the operator will be required to confirm that they have similar or better performance than the Reference Case packages under impact and thermal challenges.
- decommissioning ILW
  - (Off-site public) protected doses are generally below the most stringent BSL with the exception of those from faults involving thermal challenges to the EPR08 ($3m^3$ Box) packages which gave public protected doses above the 1 mSv BSL with a maximum predicted dose of 10.8 mSv. This is due to the contribution from the C-14, Cl-36 and Se-79 inventories which are conservatively assumed in the ROSA Toolkit to be in gaseous form. The exclusion of these radionuclides reduces the doses to below the most stringent BSL of 1 mSv. Given that the waste concerned is activated steel, and that these metallic radionuclides will be fixed within the crystalline structure prior to corrosion of the wasteform in the long term, it is considered reasonable to discount these radionuclides and to consider

the modified doses as being more representative of potential consequences, which means that an appropriate safety case could be made.

- o (On-site worker) protected doses are all below the most stringent BSL (20 mSv), meaning that there is confidence that an operational safety case can be made for all assessed packaging options from the point of view of accidental doses to workers during the repository's operational period.

The operational safety assessment for ILW from an EPR did not identify any issues that challenge the disposability of these wastes [44]. Both worker and public mitigated doses for the Reference Case and Variant Case 1 packages are below the required standards indicating acceptable performance. Although no supporting quantitative analysis has been undertaken, performance of Variant Case 2 cast-iron casks for the relevant range of faults is expected to be similar to that of the other packages and these are also judged to be acceptable at this stage of development. In some cases, doses estimated for decommissioning ILW are not compliant with existing standards, but RWMD has judged that this issue may be addressed through future refinement of the assessment methodology and continued efforts to safeguard against such events as the repository design develops. There may also be an reduction in the assessed doses from a more detailed understanding of the release of radionuclides in gaseous form during fire accidents.

Operational Safety under Normal Conditions

IAEA Regulations for the safe transport of radioactive materials [41] require that dose rates at 1m and in contact with a transport package are below 0.1 mSv/h and 2 mSv/h respectively. The expectation that packages would comply with these limits will bound the dose rates from all transport containers when handled in operations at the GDF, that is all SILW packages at all stages and UILW packages up to the point at which the waste package is removed from the transport container in the inlet cell. Since UILW packages would be handled remotely subsequent to removal from the transport container, dose rates during handling of transport containers also would be bounding on the dose rates from UILW packages.

The expectations outlined above are examined for each case below.

Reference Case

It is likely that the C1 and C4 concrete casks would be transported as IP-2 packages and emplaced in the SILW vaults. Currently it is expected that this would entail contact-handling of the packages at the GDF. Consequently, the requirement to meet dose-rate limits established in the Transport Regulations may be regarded as constraining the maximum dose-rates to be experienced during contact handling in operations.

Although the dose rates estimated for the C1 and C4 concrete Casks with maximum inventory waste may exceed the limits for transport as IP-2 packages (0.1 mSv/h at 1m and 2 mSv/h on contact), the assessment of Transport safety has identified a number of potential means for reducing the actual to acceptable levels,

Despite the expectation that the limits established by the Transport Regulations would be fulfilled for packages to be contact handled, this does not necessarily demonstrate that the doses accumulated during the handling of such packages during operations at the GDF would be ALARP. The current concepts for the operation of a GDF are preliminary and RWMD is continuing to develop the specifications for packages to include any necessary limits on dose-rates during contact handling operations. Consequently, further reductions in the expected dose-rates might be required to ensure the safety of workers at the GDF. Such reductions could be achieved through similar measures to those identified in the assessment of Transport Safety, for example, introducing additional shielding or a lower waste loading. Such measures potentially would be over-and-above any required to meet Transport Regulations, but at this time it is judged that they are feasible.

Alternatively, as discussed in the introduction to this section, should such measures prove impractical, the option to emplace these containers in the UILW vaults via a remote handling route might need to be explored.

Variant Case 1

Handling and emplacement of the Variant Case 1 packages (unshielded packages), provided they are transported in SWTCs with 285 mm of shielding, is unlikely to contribute significantly to operational doses [44], owing to the remote handling philosophy adopted, limited handling time and the relatively small number of transport containers generated.

Variant Case 2

As discussed for the reference case, while it may be anticipated that the dose-rates for the cast-iron casks would be compliant with Transport Regulations (where necessary through using appropriate additional shielding), this does not necessarily demonstrate that the doses accumulated during the contact handling of such packages during operations at the GDF would be ALARP. Consequently, further reductions in the expected dose-rates might be required to ensure the safety of workers at the GDF. Such reductions could be achieved through similar measures to those identified in the assessment of Transport Safety, for example, introducing additional shielding or a lower waste loading. Such measures potentially would be over-and-above any required to meet Transport Regulations, but at this time it is judged that they are feasible.

Alternatively, as discussed in the introduction to this section, should such measures prove impractical, the option to emplace these containers in the UILW vaults via a remote handling route might need to be explored.

Decommissioning ILW

In the case of the 4 metre Box containing decommissioning ILW (EPR06), even the dose-rates for the maximum inventory wastes are below the limits established in the Transport Regulations (0.1 mSv/h at 1m and 2 mSv/h on contact). Nevertheless, as argued above for Operational ILW packages, this does not necessarily demonstrate that the doses accumulated during the contact handling of such packages during operations at the GDF would be ALARP. Should additional reductions in dose-rates be necessary, it is judged that these could be achieved through measures such as introducing additional shielding or a lower waste loading. As indicated, RWMD continues to develop the necessary specifications to include any requirements derived from contact handling operations at the GDF.

Gas Generation and Radioactive Gas Release

In all cases RWMD has assessed the expected rates of bulk gas generation and the potential for radioactive gas generation during operations and has concluded that these are not likely to be significant issues [33]. This reflects the nature of the wastes, the small quantities of potentially gaseous radionuclides in the assessment inventories and, in some cases, the use of effectively sealed containers.

Criticality

All types of waste package for operational ILW and decommissioning ILW meet the Generic Criticality Safety Assessment (GCSA) waste package screening level of 50g Pu-239 fissile material equivalent [45]. However, application of the GCSA limit is only applicable in conditions where the waste can be confirmed to meet specific limitations on quantities of graphite, beryllium, deuterium, exotic fissile materials, moderating materials and favourable sites for sorption of fissile material [44]. In future LoC interactions the operator will need to confirm that these screening criteria will be met.

Summary

The operational and decommissioning ILW from an EPR is considered to be compatible with the targets for design basis fault mitigated doses currently being considered by RWMD. Some minor issues have been noted where packages are currently assessed to exceed existing limits in both protected accidental and routine operational doses. RWMD has judged that these issues may be addressed through future refinement of the assessment methodology, including a more detailed understanding of the release of radionuclides in gaseous form during fire accidents. This issue would be taken forward in future interactions with operators of the EPR through the LoC assessment process.

4.2.4 Environmental evaluation

Context

The Environmental Issues assessment has been included within the scope of the GDA Disposability Assessment to provide a mechanism for assessment of the main likely non-radiological environmental and socio-economic effects in relation to the disposal of radioactive waste from new build reactors within the GDF.

The assessment considers the non-radiological environmental effects of waste arising from a single reactor at the generic (non site-specific) level. This is an initial appraisal based on the information available at this time, which relates primarily to the type and quantity of ILW. Further assessment, including consideration of site-specific effects, would be required in the future to meet Environmental Impact Assessment requirements.

Results and Implications

As discussed in Section 3.2, the volume of ILW from an EPR is relatively small, and, therefore, disposal of the waste is unlikely to have a significant overall effect on GDF environmental impacts such as the extent of underground excavations, storage of spoil on site, transport of spoil, or the visual intrusion of surface facilities.

Therefore, RWMD has judged that there are no environmental considerations that challenge the disposability of EPR ILW.

It is noted that EdF/Areva proposes a forty-year period before removal and transport of ILW from the site. Such a strategy will permit waste segregation and application of the waste hierarchy, and may be beneficial in environmental terms, through minimising the volume of waste required to be accommodated at the GDF and, consequently, minimising the associated environmental effects.

4.2.5 Security and Safeguards assessments

Context

The Security assessment has included consideration of:

- Physical Protection, in particular determination of the likely security categorisation of the proposed waste packages and estimation of the quantity of Nuclear Material (NM);
- Safeguards, in particular commenting on requirements for accountancy and independent verification of the use of Nuclear Material.

The objective of the assessment is to determine the likely content of Nuclear Material in ILW from the EPR and to determine whether this would have any impact on assumptions regarding security arrangements for the existing GDF.

Results and Implications

The ILW likely to arise from operation of an EPR contains only small amounts of Nuclear Material and presents no identifiable challenges to expected security arrangements at this stage of assessment.

Bulk Nuclear Material is not expected to be present in any EPR ILW stream. Small quantities could, however, be present in the form of contamination of circuit components, filters and resins as a result of fuel failure which as explained previously is expected to be a rare occurrence.

The maximum quantity of Nuclear Material that could be present in any of the proposed waste packages is small (i.e. up to ~10g based on the assumptions explained in Section 3.3.3) comprising mainly uranium with trace quantities of plutonium. It is expected that the NM present will be in the form of fine particulate or chemically combined with the other wastes present. Based on the inventories and package characteristics discussed in Section 3.3, the ILW from an EPR would require physical protection to no higher than Category IV standards [46] for the movement of any of the projected waste packages.

The current RWMD Security Plan proposes that movements of ILW to, and within, the GDF be protected to Category III standards. Accordingly, the proposed waste packages raise no issues with respect to Physical Protection.

For ILW from an EPR there is not likely to be any safeguards issues, because of the small quantity of nuclear materials present and their wide dispersion across the packages.

4.3 Post-Closure Safety

Following emplacement of intermediate level wastes and the decision to seal and close the GDF, the void space around ILW packages will be backfilled with suitable material. The current disposal concept adopts a cementitious backfill material although other materials could be selected. The cementitious backfill is designed to provide a highly alkaline environment, which will act as a chemical barrier to the release of radioactivity and provide one of the multiple barriers of the disposal system.

Following backfilling and sealing of tunnels and access ways, the GDF will be expected to resaturate with groundwater and the disposal areas will ultimately turn anaerobic as oxygen is consumed by corrosion processes. In such alkaline and anaerobic conditions the corrosion processes affecting waste packages will be very slow and the vast majority of radioactivity within ILW is expected to remain and decay within the "near-field" of the disposal system.

The post-closure safety case is a component of the Environmental Safety Case (ESC) which is required to demonstrate to regulators the expected behaviour of the disposal system in the long term. At this early stage of GDF development, the post-closure safety component of the ESC exists for a generic GDF design and geological setting and is published as the Generic Post-Closure Performance assessment (the GPA) [73]. It is routinely used to determine and explore the impact of new wastes and new packaging proposals on the disposal system in the post-closure phase.

In the case of EPR operational and decommissioning ILW, the post-closure safety assessment has used quantitative comparison and expert judgement to consider the likely performance of the proposed waste packages relative to the performance of waste packages considered in the GPA. This comparison included consideration of the potential risk resulting from future human exposure to radionuclides from the groundwater and gas pathways, human intrusion and criticality. It has also considered impacts due to chemotoxic species contained in the ILW from a single EPR. These issues have been considered by comparison of the wastes for each EPR waste stream with an existing legacy waste stream from Sizewell B [47]. In addition, a comparison has been made between the waste arising from a programme of six EPR reactors and the waste from the legacy programme.

4.3.1 Results and Implications

Groundwater and Gas Pathways

The assessment of long-term system performance in the GDA Disposability Assessment has been based on the assumed characteristics for a generic site for the GDF [47]. Since the properties of any selected site necessarily would need to be consistent with meeting regulatory risk targets, this assessment assumed a groundwater flow rate and return time that would meet regulatory requirements when considering the inventory of legacy ILW. The additional radionuclide inventory associated with the ILW from an EPR represents only a small fraction of that of the legacy wastes, particularly for the majority of the radionuclides that determine risk in the post-closure phase.

Operational ILW

The conditioned waste volume of EPR operational ILW (Reference Case) is 1,262 m^3 [47], which is small compared to the total conditioned waste volume of 168,000 m^3 for legacy ILW assessed in the GPA. Similarly, the number of packages, 1,830 [47], is also small compared to the 285,000 legacy ILW packages assessed in the GPA. For a fleet of six EPRs the conditioned waste volumes and package number are less than 5% of the quantities assessed in the GPA. Furthermore, for a fleet of six EPRs, the contribution to the total inventory of each radionuclide assessed in the GPA is less than 0.1% for all radionuclides except Ag-108m for which the contribution is 1.6%. Even at this latter proportion, the safety significance of this radionuclide is considered trivial because Ag-108m is not a major contributor to risk in the GPA.

Recognising the requirements to refine inventory data and confirm the viability of packaging proposals identified previously, the additional risk for the disposal of ILW from a single EPR in a site of the type described would be consistent with meeting regulatory targets. The consideration of a fleet of six reactors would not alter this conclusion.

These results indicate that operational ILW from an EPR would present few challenges to disposability. However, assessment of the characteristics of the operational ILW waste packages [47] indicated two issues that the operator will need to address in any future submission under the LoC process:

- The use of epoxy resin as an encapsulant, as proposed for the Reference Case (EPR01), may necessitate a revision to the quantity of local backfill assumed for the GDF. Long-term degradation products from the resin may be acidic and affect the local pH of the groundwater, and, therefore, also affect radionuclide solubility and sorption.
- The cast-iron casks proposed as Variant 2 are expected to corrode in the long-term and produce bulk hydrogen gas. Since the GDF is expected to contain significant quantities of waste steel, the additional burden from cast-iron casks is not expected to be significant. RWMD would expect corrosion to occur at a slow rate under GDF conditions, but this would need to be confirmed in future LoC interactions. No issues with radioactive gases were identified in the assessment.

Decommissioning ILW

As with operational ILW, the conditioned volume of decommissioning ILW, at 332 m^3 [47], is low compared to the total conditioned waste volume of 168,000 m^3 for legacy ILW assessed in the GPA. The radionuclide activities of streams EPR07 and EPR08 were compared with an equivalent ILW stream from Sizewell B in Section 3.3.4, and shown to be similar [47]. Given that the decommissioning ILW will have relatively low volumes and contains comparable radionuclides to legacy wastes, it has therefore been judged that the waste is acceptable from a post-closure perspective at this stage of assessment.

However, EPR07 and EPR08, which are both stainless steel wastes associated with the reactor pressure vessel, have high specific activity for a range of radionuclides, in particular C-14. These waste streams contain 102TBq and 818TBq of C-14 respectively. The acceptable release rate, via the gas pathway, calculated by RWMD for C-14 for GDF conditions is 0.02 TBq/y. Further consideration/examination should be given to the assumed inventory of C-14 and its release rate from the steel matrix. In particular, it will be necessary to determine the fraction of the C-14 that would be released as carbon dioxide (and would react with the cementitious backfill) and the fraction that would be released as methane (and which could migrate to the biosphere). The form in which C-14 is released from ILW is a matter of ongoing generic research within RWMD. A reduced rate of release may be justified on account of the C-14 being ‘locked up’ in thick steel plates.

The heat output from EPR08 packages is 10 W/m^3. This is less than the values permitted in the waste package specification for a 3m^3 Box waste package but is nevertheless high relative to the heat output from legacy wastes, and would require appropriate placement of the packages during disposal to the GDF to avoid the potential for convective groundwater occurring around the packages following emplacement. This would, therefore, be an issue for further consideration in future LoC interactions. There are only 46 packages for EPR08, and these are late arising, so this issue might be suitably managed through design of the GDF and its operations.

Human Intrusion Pathway

The siting process adopted by Government [48] has identified geological environments that should be avoided due to the presence of natural resources and which are, therefore, areas where human intrusion may occur. Addressing the Environment Agencies’ Guidance on Requirements for Authorisation requirements [49] for human intrusion requires that any practical measures to reduce the risk from human intrusion are implemented in the GDF and that potential risks from human intrusion are optimised. These requirements do not relate, therefore, to the fundamental disposability of ILW.

Criticality

The potential for post-closure criticality of ILW from an EPR was assessed through examination of the quantity of fissile material in the waste. The minimum critical mass of a homogeneously water-moderated and fully water-reflected sphere of Pu-239 is about 510g [50]. No operational ILW stream is estimated to contain more than 5g of fissile material and no decommissioning ILW stream more than 28g of fissile material. Furthermore, based on the conservative assumptions described in Section 3.3.3, there is an assumed total of about 40g fissile material in all operational and decommissioning ILW. Therefore, the fissile material content of each waste stream is less than a minimum critical mass under the most pessimistic conditions, and the total fissile material content of all operational and decommissioning ILW is also less than a minimum critical mass.

Summary

The operational and decommissioning ILW from an EPR is considered to be compatible with current concept and assumptions for the geological disposal facility from a post-closure safety perspective. The conditioned wasteforms are small in volume and the number of packages and the waste streams are similar to those already considered acceptable. Some issues have been noted which would be taken forward in future interactions with operators through the Letter of Compliance process, including the choice of encapsulants, choice of container material for the operational ILW packages, the C-14 content and its impact on risk from the gas pathway and heat output from decommissioning ILW.

4.4 Summary of the Disposability of EPR ILW

4.4.1 General

Taking into consideration the analysis of the wastes covered in Section 3.3, the waste package properties discussed in Section 4.1, the performance of the waste packages during transport to and emplacement in the GDF discussed in Section 4.2 and the performance of the packages following sealing and closure of the GDF discussed in Section 4.3, all three cases for the packaging of operational ILW and the proposals for the packaging of decommissioning ILW have been judged to be potentially disposable.

While further development needs have been identified, including ultimately the need to demonstrate the expected performance of the packages, these would represent requirements for future assessment under the Letter of Compliance process. These issues have been listed in Appendix B. The key conclusions regarding the disposability and major issues for further consideration are highlighted in this section.

4.4.2 Inventory

The GDA Disposability Assessment has developed a good understanding of the nature and quantities of higher activity wastes that would arise from operation of an EPR. The principal radionuclides present in the ILW are the same as those present in existing UK legacy wastes, and, in particular, with the anticipated arisings from the existing PWR at Sizewell B (Section 3.3.4). This conclusion reflects both the similarity of the designs of the EPR and of existing PWRs, and the expectation that similar operating regimes would be applied.

For operational ILW, the conditioned waste volume (1,262 m^3) and number of packages (1830) is small compared to legacy wastes (168,000 m^3 and 285,000). In addition, the total radionuclide inventory in the lifetime arisings from a single reactor is small compared to legacy wastes, and is less than 0.01% of the legacy activities for each radionuclide apart from Ag-108m, for which the activity in EPR wastes is 0.28% of the activity from legacy wastes.

For decommissioning ILW, the total activities of the five radionuclides with the highest activities in EPR stainless steel decommissioning ILW streams are similar (within a factor of three) to the equivalent waste streams from Sizewell B (Section 3.3.4, Table 10). The inventory associated with the operational ILW would depend on operating decisions, for example the permitted radioactive loadings of Ion exchange resins and Filters, and therefore could be managed to more closely match the levels in existing legacy wastes.

The assumed carbon-14 content of the decommissioning ILW is high, and as discussed in Section 3.3.3, this is primarily due to the assumed pre-cursor concentration. For carbon-14, the precursor is nitrogen, which is assumed to be present in reactor internal steel at a concentration of 800ppm. The concentration of nitrogen in reactor internal steels is likely to be lower than this in practice and can be managed by specification of steel grades during construction of the reactor.

4.4.3 Waste Packages

The proposals for the packaging of ILW discussed in Section 4.1 include outline descriptions of the means proposed for immobilising the activity associated with waste. Detailed descriptions and supporting evidence as to the performance of the proposed packages are not provided at this stage. This is consistent with expectations for the GDA Disposability Assessment. In future, RWMD would expect to work with potential reactor operators and provide assessment of fully-developed proposals through the Letter of Compliance process.

The Reference Case proposals, based on non-standard concrete casks, are not compliant with some aspects of existing RWMD standards for waste packages. Nevertheless, RWMD has judged that it should be feasible to develop design concepts for the transport of packages to the Geological Disposal Facility, and for their subsequent handling and emplacement in disposal vaults. Further development of the proposed conditioning methods, using either a polymer or cement grout, will be required, but RWMD considers that, based on experience of similar wastes, such methods can be developed.

The proposal under Variant Case 1 to use standard RWMD waste containers provides compliance with many aspects of the existing standards and specifications.

EdF/Areva has indicated that most of the operational ILW would not be directly conditioned into the 500 litre Drums under Variant Case 1. Instead, the wastes would be packed into smaller containers (200 litre drums) that would be grout enclosed within the 500 litre Drums. This does not represent common practice in the UK and further is not an efficient use of packaging volume. Nevertheless, the GDA Disposability Assessment has concluded that the necessary performance potentially would be available from such packages due to their robust nature. It is also noted that full immobilisation could be achieved through application of a conditioning process to the materials inside the 200 litre Drum or to the materials loaded directly into the 500 litre Drum.

For the packages proposed under Variant Case 2, based on fully-sealed, cast-iron containers, the proposal is similar in that they would contain unconditioned wastes. In this case, it is anticipated that the robust nature of the containers alone potentially would provide containment of unconditioned wastes. Further development ultimately would be required of the means of treating the wastes prior to packaging. In particular drying to remove water to control the evolution of the wastes and prevent gas pressurisation. Nevertheless, it is judged that viable treatment processes are currently available. Furthermore, such packages are currently approved for the packaging of ILW from light water reactors in Germany.

The proposed decommissioning ILW packages comprise metal items conditioned in standard containers using a cement grout. These proposals conform to existing practices for decommissioning wastes in the UK and are expected to produce packages that would be compliant with existing RWMD standards and specifications.

4.4.4 Impact on Design

The potential impact of the disposal of EPR operational and decommissioning ILW on the size of the GDF has been assessed (Section 4.2.1). This impact has been characterised as a fractional change in the 'footprint area' of the GDF, as compared to the area required for the disposal of the legacy ILW. Although the impact has some dependence on the particular case considered for operational ILW, in all cases the necessary increase in the area is small, corresponding to approximately 60m of vault length for each EPR. This represents approximately 1% of the area required for the legacy ILW, per reactor, and less than 10% for the illustrative fleet of six reactors.

4.4.5 Transport Safety

The transport safety assessment has identified that carbon-14 has the potential to challenge safety limits. In the main this is because of the highly conservative approaches used in the RWMD assessment toolkits which assume release of carbon-14 from thick steel plates even though these activation products are likely to be locked up within the metal structures. In support of future assessments, RWMD recognises that improved methods will need to develop for the evaluation of such materials.

Although the Reference Case concrete casks are licensed for the transport of wastes from existing PWRs in France, application of the EPR assessment inventory suggests that some packages containing operational ILW at the maximum bounding inventory could exceed dose-rate limits permitted under current Transport Regulations. RWMD has judged that this issue may be addressed through future refinement of the assessment inventories, including provision of better data to remove pessimisms, consideration of an appropriate time for radioactive decay and/or development of the detailed packaging methods.

For Variant Case 1 waste packages, the requirement for such packages to be transported in a shielded transport over-pack eliminates potential challenges to the dose-rate limits set out in the Transport Regulations.

The use of the cast-iron casks transported as a Type B package in Variant 2 has not been fully evaluated due to the reduced level of information available to support this packaging option. However, it is known that these containers are licensed for use in Germany as IAEA Type B containers for certain waste types and this gives confidence that there should not be any major transport safety issues that cannot be resolved as part of a future LoC interaction. Successful licensing of the cast-iron casks would require demonstration that gas pressurisation would not be a problem during transport by ensuring that the waste was dried before packaging and this practice is currently followed in Germany.

The current maximum assessment inventory for the decommissioning ILW proposed to be packaged in 4 metre Boxes challenges some aspects of the Transport Regulations in relation to dose-rates but it is judged that this issue could be addressed by refining the assessment inventory, modifying the proposals to include additional shielding, management of waste loading or employing containers that necessitate the use of a shielded overpack for transport (i.e. the 3m^3 Box proposed for the remainder of the decommissioning ILW).

4.4.6 Operational Safety

The operational safety assessment for ILW from an EPR did not identify any issues that challenge the disposability of these wastes. Both worker and public mitigated doses for the Reference Case and Variant Case 1 packages are below the required standards indicating acceptable performance. Performance of Variant Case 2 cast-iron casks for the relevant range of faults is expected to be similar to the other packages and these packages are also judged to be acceptable at this stage of assessment.

The doses arising from packages containing decommissioning ILW are generally acceptable, although off-site discharges of potentially gaseous species released in fire accidents are relatively large. RWMD has judged that the expected doses may be reduced in future through refinement of the assessment methodology, including a more detailed understanding of the release of radionuclides in gaseous form during fire accidents.

4.4.7 Environmental Considerations

No environmental issue that challenge the viability of the disposal of ILW from an EPR has been recognised (Section 4.2.4).

4.4.8 Security and Safeguards

The ILW to be disposed of from operation of an EPR present no security or safeguards issues of significance (Section 4.2.5).

4.4.9 Post-closure Safety

The assessment of long-term system performance in the GDA Disposability Assessment has been based on the assumed characteristics for a generic UK site for the Geological Disposal Facility. Since the properties of any selected site necessarily would need to be consistent with meeting the regulatory risk guidance level [49], based on the approach adopted for Letter of Compliance assessment, this assessment assumed a groundwater flow rate and return time to the accessible environment that would meet regulatory requirements when considering the inventory of legacy ILW. The additional radionuclide inventory associated with the ILW from an EPR represents only a small fraction of that of the legacy wastes, particularly for the majority of the radionuclides that determine risk in the long-term.

Even considering the conservative approach to inventory assessment and recognising the potential for future optimisation of packaging proposals, the additional risk from the disposal of ILW from a single EPR in a site of the type described would be consistent with meeting the regulatory guidance level. The consideration of a fleet of six reactors would not alter this conclusion.

5 ASSESSMENT OF SPENT FUEL

In this section, we discuss the assessment of EdF/Areva packaging proposals for spent fuel (described earlier in Section 3.4) against RWMD's preliminary waste package specification [8] and disposal system specification [10]. The assessment approach follows that described in Section 2.2.

The assessment is reported in five sections:

- Section 5.1 describes the assessment of the interim storage period required for the spent fuel prior to emplacement for disposal;
- Section 5.2 describes the assessment of wasteform properties and performance of the overall waste package including the predicted behaviour in accident conditions;
- Section 5.3 describes the impact of spent fuel disposal packages on the disposal system, including engineering design impact, transport safety, safety during receipt, handling and emplacement in the GDF, environmental issues, and security and safeguards implications.
- Section 5.4 describes the assessment of the impact of spent fuel disposal packages on long-term safety following closure of the GDF;
- Section 5.5 provides a statement regarding the overall disposability of spent fuel from an EPR.

For each component of the assessment, the report addresses the context (i.e. the required performance), and the results and the implications of the assessment. Issues identified through GDA Disposability Assessment for each component of the evaluation are listed in Appendix B.

5.1 Interim Storage Period for Spent Fuel

Context

Spent fuel contains both short-lived and long-lived radionuclides, which will decay through various decay chains emitting ionising radiations and generating heat. Following discharge from the reactor, spent fuel will be maintained in interim storage on the power plant site for a period of initial cooling. This cooling allows the activity of short-lived radionuclides to decay significantly, and, therefore, makes transport and disposal of the spent fuel less challenging. EdF/Areva has proposed that, for the first 10 years, cooling will take place in the fuel building storage pool. Fuel will later be transferred from this pool to interim storage (the storage conditions remain to be decided at this time, and may be either dry or wet storage).

A key requirement for estimating spent fuel disposal package properties which are of relevance to transport and disposal is the development of an appropriate estimate for the period of interim storage that is required.

5.1.1 Results and Implications

As described in Section 2.1, current disposal concept work envisages that a bentonite buffer is emplaced around the disposal package. It is widely recognised that the heat generated by spent fuel can potentially affect the performance of the engineered barriers, especially the bentonite buffer, for example through alterations to the mineralogy of the bentonite.

Therefore, the preliminary waste package specification for spent fuel [8] currently specifies an upper limit on disposal package thermal output determined by a temperature constraint on the “near-field” of the GDF[17]. The current thermal constraint of 100ºC that RWMD applies to the near-field of the GDF is based on international precedent, for example [51]. Therefore, heat transfer calculations conducted by RWMD in support of the GDA Disposability Assessment have applied 100ºC as a limit to the inner boundary of the bentonite. It should be noted that there is uncertainty over the impact of thermal processes on the near-field, for example the temperature at which potentially detrimental mineral transformations occur is subject to uncertainty and there is evidence that the transition may occur at temperatures higher than 100ºC [52]. The applicability of the 100ºC limit will be maintained under review by RWMD.

A heat transfer model has been used to calculate the temperature profile across the cast-iron inner vessel, the disposal canister, the buffer and the host rock and has been used to explore how this profile varies with time. Based on the time-dependent heat output from the spent fuel, it has been possible to estimate the interim storage period needed to comply with the disposal temperature constraint.

Inventory and Burn-up Assumptions

The heat output from spent fuel is dependent upon the activity of key heat emitting radionuclides. At cooling times of 30 to 100 years (which RWMD consider to be typical times anticipated for interim storage of spent fuel based on knowledge of national waste management programmes) the key heat emitting radionuclides include Sr-90, Cs-137, Pu-238 and Am-241. The activity of these key radionuclides increases with fuel burn-up. To provide a base case estimate for the interim storage time, the inventory calculations adopted a pessimistic approach as follows:

- Maximum fuel assembly burn-up of 65 GWd/tU for all assemblies – noting that the average burn-up is likely to be closer to 50 GWd/tU (as discussed in Section 3.4.3).

- Irradiation history[18] that maximised the total activity in the fuel at one year cooling.

To investigate the sensitivity of interim storage time to fuel burn-up, a variant fuel inventory calculation was performed based on a fuel assembly burn-up of 50 GWd/tU.

Thermal Modelling

Four different calculations have been performed; two for the 65 and two for the 50 GWd/tU fuel burn-up cases. For both the 65 GWd/tU base case and the 50 GWd/tU variant case, two calculations have been performed, one assuming four fuel assemblies per disposal canister and one assuming three fuel assemblies per canister. This provides an understanding of how the interim storage period is influenced by the number of assemblies in the disposal package.

The thermal model assumes the disposal canister to be made of copper and the geometry of the canister and buffer to be as shown in Figure 2. After emplacement in the deposition hole and placement of the bentonite buffer, the temperature of the disposal canister and buffer

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[17] The near-field comprises the engineered barriers and the host rock immediately surrounding the engineered barriers, and which is affected by construction and operation of the GDF.

[18] In an EPR, fuel burn-up is accumulated in an irradiation “history” typically consisting of 3 to 5 irradiation cycles, each of 12 to 24 months at power with 10 to 20 days with the reactor shutdown for maintenance purposes.

climb and reach a maximum after about 20 years. After this time, temperatures gradually decrease as a result of the falling heat output of the fuel. Figure 10 shows the temperature transient for a disposal canister containing four 100-year cooled maximum burn-up assemblies (in Figure 10, the buffer inner surface refers to the part of bentonite adjacent to the disposal canister, and buffer outer surface refers to the part of bentonite adjacent to the near-field rock). For this case it can be seen the peak temperature reached by the buffer (yellow line) is 100 ºC, the limit adopted for buffer temperature in the current assessment.

Figure 11 presents the dependence of the peak buffer temperature on fuel cooling time prior to emplacement (interim storage time) for a disposal canister containing three or four 65 GWd/tU fuel assemblies. This figure shows that the cooling time prior to emplacement required for the three and four reference fuel assembly cases are approximately 74 years and 98 years respectively. The 98 year cooling time is presented as a rounded 100 years in the remainder of this report.

The nature of the temperature transients calculated for the 50 GWd/tU variant fuel inventory cases are very similar to those shown in Figure 10 for the 65 GWd/tU base case but with the cooling time axis shifted approximately 20 years earlier. Hence it is estimated that the cooling times prior to emplacement, for a disposal canister containing three or four 50 GWd/tU fuel assemblies, are 56 and 77 years respectively.

In addition to the option of reducing the number of spent fuel assemblies in each waste package, other options can be identified for modifying the disposal concept to allow for greater flexibility in disposal of heat generating waste. These include consideration of a double-layered buffer [53], use of prefabricated engineered modules to ensure that the bentonite remains dry and mineralogically stable during the post-closure thermal phase [54], and use of different emplacement geometries to those assumed in concepts developed by RWMD to date [55].

The use of a steel shell for the disposal canister rather than a copper shell would not have a significant impact on the interim storage period because the temperature in the engineered barrier system is controlled by regions of low thermal conductivity such as the ceramic fuel, air gaps, the bentonite buffer and the host rock. Both copper and steel have a high thermal conductivity relative to these components of the near-field; hence the outer part of the disposal canister makes a negligible contribution to the overall temperature profile in the vault.

RWMD are continuing to investigate thermal constraints on the disposal facility near-field but significant progress is unlikely to be made in this stage of GDA. For the purposes of the GDA Disposability Assessment, RWMD has carried forward the 100-year estimate for interim storage as its base case assumption for this stage of assessment.

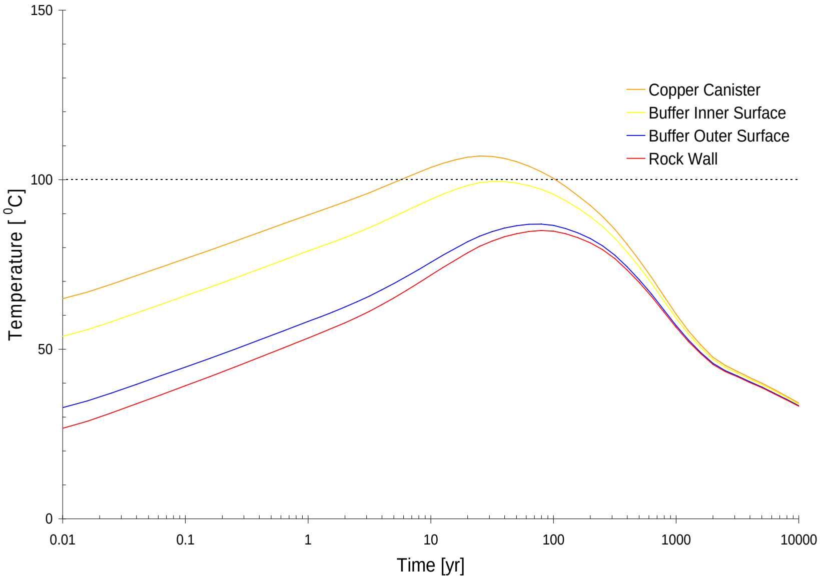

Figure 10 **Near-field temperature history for 65 GWd/tU base case following emplacement of spent fuel waste packages; buffer inner surface refers to the part of bentonite adjacent to the disposal canister, and buffer outer surface refers to the part of bentonite adjacent to the near-field rock**

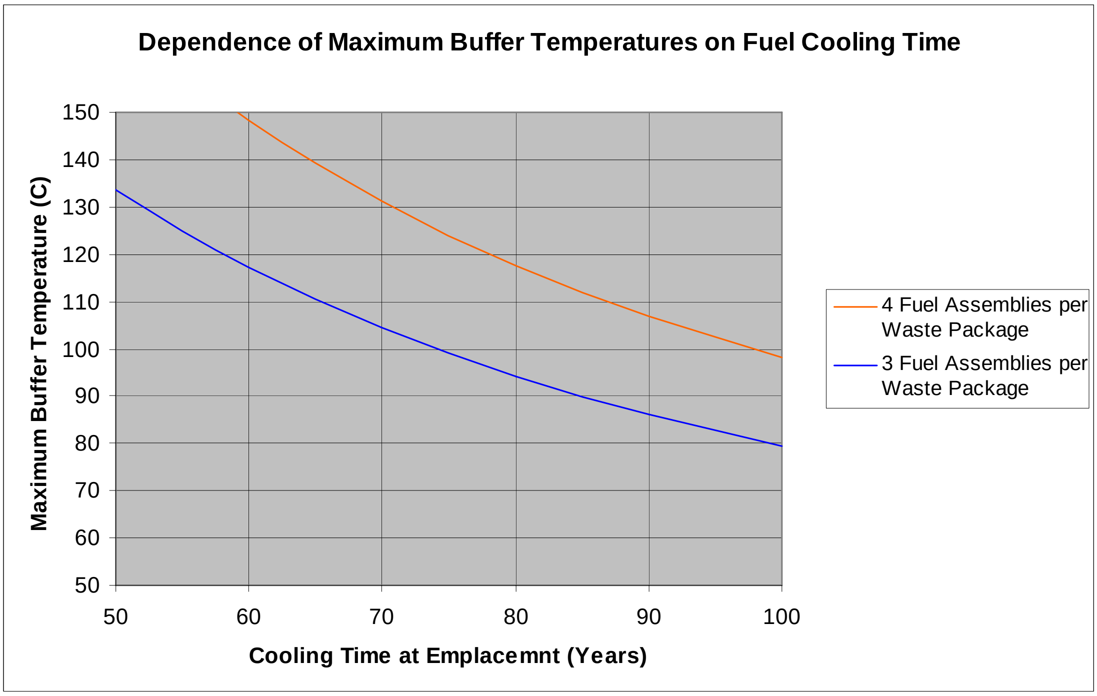

Figure 11 **Interim storage cooling times for 65 GWd/tU base case required to attain required temperatures in the inner surface of bentonite buffer**

5.2 Spent Fuel Disposal Package Properties

5.2.1 Wasteform

Context

The provision of a hermetically sealed, durable copper or steel disposal canister will provide primary containment of radioactivity in the spent fuel in the short and medium term, following emplacement in the GDF. However, in the long term, and in the event that the waste container is breached through corrosion, then the wasteform will contribute to controlling the rate of release of radionuclides. The Wasteform evaluation has therefore sought to provide an understanding of the properties of the spent fuel assembly to provide information to input to subsequent stages of the assessment.

A particular issue for the Wasteform evaluation has been to develop an understanding of the impact of irradiation on the properties of the fuel. This is particularly relevant for spent fuel from the EPR because of the high burn-up assumed.

Physical properties identified as relevant to disposability safety cases are the distribution of radionuclides within, and the physical integrity of, the spent fuel. The fraction of activity that is readily released upon contact with groundwater is referred to as the instant release fraction (IRF). The IRF represents the radionuclide-specific fraction of the inventory that is estimated to be present in readily soluble form in the gap between fuel pellets and the cladding, in grain boundaries and fractures in the fuel pellets, and in the rim region of fuel pellets.

Results and Implications

Although the use of high-integrity Zircaloy M5 cladding is expected to provide protection to spent fuel pellets following discharge from the reactor, at present there is little or no evidence available to RWMD that credit can be taken for cladding integrity in the long term. The U.S. Nuclear Regulatory Commission has noted that there is some (limited) evidence that burn-ups up to and beyond 60 GWd/tU can threaten cladding integrity through oxidation [56]. In addition, irradiation to high burn-ups may cause thermal and stress cracking damage to the fuel matrix, and production of particulates contained within intact cladding tubes is possible [57]. Until and unless further research is undertaken to demonstrate the long-term continued integrity of the fuel cladding, the RWMD safety case will proceed on the basis of an instantaneous fraction of radionuclides being released from spent fuel immediately following container failure followed by longer-term leaching. This is consistent with approaches in other national disposal programmes in which no credit is taken for the cladding in post-closure safety assessments.

Estimates for the IRF for EPR spent fuel have been collated from published information on PWRs at a range of burn-ups up to 70 GWd/tU [58]. These data have been used to estimate radionuclide-specific IRFs at 65 GWd/tU. IRFs are higher for high burn-up fuel, for example, based on a linear interpolation of data presented in [58] RWMD estimates the IRF for the key post-closure radionuclide iodine-129 following burn-up of 65 GWd/tU is 13%, whereas [58] estimates the IRF for I-129 for lower burn-up fuel (e.g. 37-41 GWd/tU) is 3%. Note that RWMD has used the best estimates given in Reference [58] in the post-closure safety calculations presented in Section 5.4 because they are based on actual fission gas release correlations.

5.2.2 Spent Fuel Disposal Package Performance

Context

Preliminary expectations for the required performance of spent fuel disposal packages have been defined by RWMD [6]. The specification for the packages is based on preliminary safety assessments for the performance of spent fuel disposal packages. It is recognised that the specification will need to be revisited as the safety case is developed.

For impact performance, the following requirements have been specified for the disposal package:

- the package should be designed such that, in the event of an impact accident, the release of radioactive material is low and predictable, exhibits progressive behaviour with increasing impact severity and does not exhibit significant deterioration in package performance for a small adverse change in conditions;

- the package shall be capable of withstanding normal handling, including minor impacts etc, and remain suitable for safe handling during all subsequent phases of disposal;

- the package shall be capable of being dropped, in any attitude, from a height of 8 metres onto an unyielding surface, whilst retaining its radioactive contents;

For assessment of disposal package performance in fire accidents, the following requirements have been specified [8]:

- the package should be designed such that, in the event of a fire accident, the release of radioactive material is low and predictable, exhibits progressive behaviour with increasing event severity and does not exhibit significant deterioration in package performance for a small adverse change in conditions within the anticipated range of fire conditions;

- the package should be capable of withstanding a fully engulfing, 1000°C hydrocarbon pool fire of 1 hour duration, with a release of contents that should not result in an on-site dose consistent with requirements in HSE's safety assessment principles (SAPs) [59].

Results and Implications

The performance of spent fuel disposal packages under impact accident and fire accident conditions is determined by the combined performance of the outer shell of the disposal canister, the cast-iron inner vessel and, during transport, the shielded transport container. Evaluation of disposal package performance under such accident conditions has been undertaken by RWMD [37] based on modelling studies for similar disposal packages previously undertaken by Posiva [60] and RWMD [61].

Initial analysis of potential accidents has indicated that breaches caused by impact and fire accidents are either implausible or can be designed out of the disposal system, and, therefore, the release fraction for packaged spent fuel under accident conditions has been assumed to be zero in subsequent safety analyses [37]. These findings would need to be confirmed in future Letter of Compliance process assessments. In particular, it will be necessary to confirm that the spent fuel canister is not subject to inappropriate gas pressurisation under both normal and fire accident conditions. Discussion of transport and operational safety is presented in Sections 5.3.2 and 5.3.3 respectively.

5.3 Disposal System Issues

5.3.1 Impact on disposal facility design

Context

The Design Impact evaluation has sought to establish an understanding of the impact of EPR spent fuel on the design of the disposal facility [40].

A key issue impacting design, safety and potentially siting of a GDF is the increased volume of host rock required in the event that spent fuel from a new build EPR is disposed alongside legacy wastes and/or legacy materials. The implication of this can be estimated in the form of a "footprint" area increase, where the footprint is the projection of the disposal facility area on the land surface.

The Design Impact evaluation has considered the impact on the GDF of a single EPR based on the assumption that the spent fuel is packaged prior to consignment. The impact of a fleet of six EPRs has also been considered [40].

The footprint estimates developed in the evaluation are idealised, and are based on a regular array of horizontal deposition tunnels and regular spacing of deposition holes within the tunnels. In practice, at a specific site, the spacing of deposition tunnels and deposition holes would be based on site-specific geological, hydrogeological and geotechnical data available at the time of construction. Variation from this idealised layout would be expected, for example the footprint could be larger than considered in the idealised design in order to avoid unsuitable features of the host rock, or could be smaller by constructing the disposal tunnels on more than one levels.

The disposal concept considered in the Design Impact assessment is a generic design that was developed as a basis for preliminary planning for geological disposal of spent fuel [11]. RWMD expects to revisit this design to tie in with the revised "baseline inventory" identified in the White Paper. As the MRWS process progresses RWMD will review the design based on information relevant to a specific site and a specific setting.

Results and Implications

For a packaging assumption of four fuel assemblies per canister, the 3,600 fuel assemblies would require 900 disposal canisters. These would be placed in individual disposal holes within the deposition tunnels. This arrangement is the same as that adopted for legacy spent fuel in previous disposal assessments, although the length of the canisters would be extended from the current longest length of 4.5 m to 5.2 m to accommodate the longer fuel assemblies of the EPR [40]. Other design impacts associated with this include [40]:

- increased canister weight to 21t. This might require an increase in the safe working load of specific handling equipment (e.g. cranes, transport wagons and transport containers);
- increased deposition tunnel height. In order to accommodate longer disposal packages the disposal tunnel height would be increased to approximately 6.5m. This would increase the excavated volume of rock, and increase the quantity of material used in the disposal holes and the volume of the backfill for the deposition tunnels;
- possible modification of lift heights at transfer points;
- modified specification for the deposition machine.

Based on ~ 45 disposal holes per disposal tunnel, 20 disposal tunnels would be required for disposal of the 900 spent fuel disposal canisters from an EPR. The area required for 20 disposal tunnels is approximately 0.15 km^2 [40]. The disposal tunnels required for emplacement of spent fuel from operation of a fleet of six EPRs would require 0.9 km^2 (the overall GDF footprint would in fact be slightly greater, due to a need for extra underground supporting infrastructure). This represents approximately 8% of the area required for the legacy HLW and spent fuel per reactor, and approximately 50% for the illustrative fleet of six reactors.

The EdF/Areva proposals did not include any information regarding disposal package identification markings [40]. In any future LoC interaction the operator will need to describe how spent fuel package identifiers will be included in line with existing requirements (i.e. Appendix B of [8]).

For an EPR that commenced operation around 2020, disposal of spent fuel from the reactor would commence in approximately 2120.

5.3.2 Transport Safety

Context

Based on the assumption that spent fuel will be packaged for disposal before dispatch to a GDF (Section 3.2), it follows that arrangements will be required to transport spent fuel packaged in disposal canisters safely through the public domain. As described earlier (Section 4.2.2) RWMD is planning the transport system that will be required to ship all higher activity wastes from sites of arisings to a GDF. This will be achieved in the case of spent fuel by provision of a shielded transport container that meets the requirements of the IAEA Transport Regulations [41] as implemented by UK transport legislation.

The Disposal Canister Transport Container (DCTC), which was described earlier (Section 3.4.2) is the transport container concept developed by RWMD for transport of spent fuel through the public domain [62]. Further work is required to develop the DCTC into a detailed design, but it provides a baseline for assessment of transport issues.

The DCTC as currently envisaged would provide shielding to reduce external gamma and neutron radiation. Steel shielding of 140mm and neutron shielding material of 50mm have been calculated to be sufficient to reduce external dose rates for legacy spent fuel to levels compliant with IAEA requirements.

The transport assessment has checked EPR spent fuel for compatibility with the existing DCTC concept and against the generic transport risk assessment.

Results and Implications

Arrangements for the transport of packaged spent fuel to a Geological Disposal Facility are at an early stage of development. Consequently, although the EPR spent fuel may significantly influence the necessary arrangements, for example additional shielding requirements, it is currently judged that sufficient flexibility exists to allow suitable arrangements to be developed. Comments on specific issues considered in the transport safety assessment are provided below.

Activity Content

The current design of the DCTC is subject to a contents limit of 10^5 A_2 because it has not been designed to withstand an ‘enhanced water immersion test’ (Paragraph 730 of the IAEA

Transport Regulations [41]). Based on the data for EPR spent fuel (Section 3.4, Table 15), this limit would be challenged by the A_2 activity content of EPR fuel. Two options are available: remove pessimisms from the fuel inventory data or design the DCTC to withstand the water immersion test. RWMD is confident that the DCTC design could be modified to meet this requirement.

External Dose Rates

The external dose rate from a loaded transport container has been calculated and compared to the limit of 0.1 mSv/hr at 1 m from the transport container specified in IAEA Transport Regulations for non-exclusive use [25]. For gamma radiation the dose rate is 0.1 mSv/hr at 1 m from the transport container, while for neutron radiation the dose rate is 0.02 mSv/hr at 1 m. Although the total estimated dose rate is slightly above the IAEA limit, this may be addressed through optimising the shielding provided. For example, although the current conceptual design includes 140 mm of steel, several designs of existing fuel flask provide greater shielding than this. Furthermore, these initial shielding calculations assume a conservative burn-up and a cooling period of 90 years, whereas the actual design of the transport container would be influenced by cooling times (and burn-ups) that may be further varied in future. On this basis, it is concluded that the design of the DCTC could be expected to provide acceptable dose rates.

Gas Generation under Normal Conditions

The waste package is expected to be seal welded closed once the spent fuel has been loaded. Gas generation leading to pressurisation of the DCTC cavity is therefore not expected to be an issue.

Containment under Normal Conditions

Radioactive and bulk gas releases into the cavity of the DCTC are expected to be zero under normal conditions.

Containment under Accident Conditions

Estimation of the release fractions in the disposal package performance evaluation concluded that zero release fractions should be used in the GDA Disposability Assessment for the EPR. Therefore, the design of the DCTC is expected to be sufficient to meet the requirements for containment under accident conditions. In future submission under the LoC process, the operator will need to confirm zero or low release fractions from the disposal package in accident conditions through testing and modelling of the disposal packages.

Heat Output

The GDA Disposability Assessment estimated that the heat output from the disposal canister will be approximately 1.43 kW, based on the conservative assessment inventory. The actual heat output from spent fuel would be affected by the assumptions regarding burn-up and the period of interim storage. It is also recognised that there would be several options for modifying the DCTC to accommodate the heat output of its intended contents, for example the addition of fins to increase the surface area of the container and facilitate heat transfer. On these grounds, it is concluded that design measures would be sufficient to ensure that the DCTC would meet IAEA transport regulation limits on heat output, surface temperature and surface heat flux.

Weight Limits

For rail transport, a maximum gross weight of 65 t is applicable for a four-axle rail wagon, which is NDA's current design basis [63]. The mass of the DCTC loaded with a disposal canister containing four EPR spent fuel assemblies is estimated to be approximately 45t, which is therefore compatible with existing design assumptions for transport by rail.

Transport Operational Risks

The additional transport movements associated with transport of EPR spent fuel to a GDF have been compared with the generic transport risk assessment [42], which was conducted for ILW. It has been found that the number of transport movements leads to an increase in the routine risk to the public, routine dose to the worst case individual and maximum effective dose to train crews. However, the doses calculated are below the design limits set in the Radiological Protection Policy Manual [64]. No increase has been observed for accident risk since radioactive release in accident conditions is expected to be zero.

Criticality

Nuclear fuel is most reactive prior to irradiation and fresh fuel is readily transported to reactor sites prior to use. Subsequent to irradiation, the increased irradiation anticipated for EPR would reduce the reactivity compared to spent fuel from current PWRs. Furthermore, it has been reported that fresh fuel from the Swedish programme contained in a sealed (water-tight) disposal canister would be sub-critical [65].

The most significant challenge to the maintenance of spent fuel in a criticality-safe condition during transport would an accident that resulted in the introduction of a potential moderator into the disposal canister, in particular water ingress. However, analyses of impact accidents involving the DCTC carrying a spent fuel package indicate that the container would remain watertight under impact conditions. Criticality scenarios involving water leakage into the DCTC or disposal canister therefore can be excluded.

On the basis of these arguments, it has been concluded it should be possible to construct a criticality safety case for the transport of EPR spent fuel in the DCTC sufficient to fully meet IAEA requirements for criticality safety. The development of such a case would be considered further in a future assessment under the LoC process.

5.3.3 Operational Safety

Context

The operational safety of spent fuel disposal has previously been considered in a generic operational safety assessment undertaken during development of the reference disposal concept [66] for provision of disposability advice. This assessment used a fault schedule that was based on the fault schedule applied in the GOSA. More recently, RWMD has updated the safety assessment using revised fault schedules [67]. This work was undertaken in connection with packaged HLW, but is equally applicable to packaged spent fuel.

The operational safety assessment undertaken for the GDA Disposability Assessment for the EPR [44] considered the following situations:

- design basis accident conditions;
- doses to workers under normal conditions;

- criticality safety.

The analysis of design basis accident conditions used the faults developed in [67] judged to be relevant to disposal of spent fuel packages. Of these faults, five external radiation faults were considered to require further consideration:

- Entry to Underground Transfer Facility with waste packages present;
- Underground Transfer Facility shield doors opened with waste packages present;
- Accidental export of unshielded waste packages from Underground Transfer Facility;
- Entry to Deposition Tunnel during emplacement;
- Delivery transport container or deposition machine opened for maintenance contains overlooked waste packages.

For these faults, protected (mitigated) doses were estimated through use of dose rates at 3 m from the disposal package calculated in the N&Q assessment [25]. In the analysis, dose rates at a distance of less than 3 m were estimated using an inverse square law and dose rates at a distance of greater than 3 m were estimated using an inverse linear relationship. Assumptions regarding the distance at which exposure occurs and the period of exposure were based on expert judgement of operational practices. The estimated doses were compared to the targets for design basis fault sequence mitigated doses presented in Table 30. Protected doses take account of the correct functioning of any safety systems included in the design.

In addition to the external radiation events, the assessment considered a single contamination event fault - excessive surface contamination on delivery transport container, and compared the estimated doses to the targets for design basis fault sequence mitigated doses presented in Table 30.

The assessment did not undertake any quantitative assessment of impact and fire events because the Waste Package Performance evaluation had concluded that the release fractions from spent fuel disposal packages should be assumed to be zero at this stage of assessment.

Although the Operational Safety assessment calculated doses, given the current status of the design of the facility and the assessment of spent fuel emplacement operations, the purpose of the calculations is to provide insight into the key issues affecting operational safety rather than make any claim for the acceptability of the doses. Therefore, the Operational Safety assessment was judged qualitatively by RWMD, by using the information from the calculations to identify potential issues for further analysis.

Results and Implications

The safety of spent fuel emplacement operations is dependent on the properties of the disposal canister and the protection against exposure to radiation provided by the safety systems included in the design of the disposal facility. The disposal canister is a robust package that is expected to withstand plausible accidents within the disposal facility. The safety systems that will be included within the disposal facility will include gamma monitoring systems and interlocks to prevent worker exposure to the disposal canisters in regions of the disposal facility where the disposal canister is transferred from the transport container to an emplacement machine.

Arrangements for the emplacement of packaged spent fuel in a Geological Disposal Facility are at an early stage of development. Consequently, although the EPR spent fuel may significantly influence the necessary arrangements, for example additional shielding requirements; it is currently judged that sufficient flexibility exists to allow suitable arrangements to be developed. Comments on specific issues considered in the operational safety assessment are provided below.

Design Basis Accident Conditions

All results are below the most stringent BSL for workers (20 mSv) indicating that the robust construction of the disposal packages and installation of protection measures in the GDF should readily permit the making of a safety case.

Doses to Workers under Normal Conditions

At all times when operators may be present, under normal conditions of operation, the spent fuel is kept behind shielding in either the DCTC, the Underground Transfer Facility where the spent fuel will be transferred from the DCTC to the deposition machine used to emplace the waste in the deposition holes, or in the deposition machine itself.

The integrated dose incurred by workers will be proportional to the time for which they are exposed. For receipt of transport containers, time will be spent on monitoring and transferring the containers between conveyances. Some exposure will also occur during their transport underground via the drift and transferring them into the Underground Transfer Facility. Underground, the normal operations dose accrued will be determined by the thickness of shielding afforded on both the Underground Transfer Facility cell-line and the deposition machine.

For all of stages the dose will be controlled by provision of shielding sufficient for the protection of workers to the requisite standard.

Criticality

The disposal packages containing EPR spent fuel would be handled and placed individually, and it is anticipated that the necessary spacing of disposal holes would ensure minimal neutronic interaction between packages. Consequently, at this stage it is concluded that the arguments pertaining to criticality safety during transport may be extrapolated to operations at the GDF.

As is the case for transport, the most significant challenge to the maintenance of spent fuel in a criticality safe condition during operations would an accident that resulted in the introduction of a potential moderator into the disposal canister, in particular water ingress. In addition to the judgement that the container would remain watertight under impact conditions, it is noted that significant volumes of water are not expected to be present during GDF operations. Criticality scenarios involving water leakage into the DCTC or disposal canister therefore can be excluded.

Based on the above, it may be concluded that a criticality safety case for the handling of disposal packages containing EPR spent fuel during operations at the GDF could be produced. Although any such case would need to consider the detailed plans for handling packages, it is anticipated that the development of such plans could readily incorporate any requirements arising from a criticality safety case. Furthermore, the development of such a case would be considered in a future assessment under the LoC process.

5.3.4 Environmental Issues

Context

The context for the Environmental Issues assessment is as described in Section 4.2.4.

Results and Implications

As discussed in Section 5.2, the disposal of spent fuel arising from a single EPR in a geological disposal facility will have an associated impact on the GDF footprint. It is estimated that an extra 500,000 m^3 of rock would be excavated if EPR spent fuel were to be disposed of in an existing facility for legacy wastes. This will have an effect on the extent of excavations, the amount of spoil generated and on strategies for storage of spoil whether on site or off site.

5.3.5 Security and Safeguards Evaluation

Context

The Security and Safeguards evaluation included consideration of:

- Physical Protection, in particular, identification of Nuclear Material (NM) and determination of the likely security categorisation of the proposed waste packages;
- Safeguards, in particular, commenting on requirements for accountancy and independent verification of the Nuclear Material.

The objective of the assessment was to determine the likely content of Nuclear Material in spent fuel from the EPR and to determine whether this would have any impact on assumptions regarding security arrangements for a GDF.

Results and Implications

The total maximum quantity of Nuclear Material that could be present in the proposed disposal packages would be ~2t, comprising mainly uranium, but also containing 23.9kg plutonium (Table 15). Trace quantities of U-233 and thorium would also be present. EPR spent fuel could be classified as Category I Material by the Office of Civil Nuclear Security (OCNS) on account of this quantity of Nuclear Material. The current RWMD Security Plan would need to be updated to include for the provision of spent fuel transport. Accordingly, it is planned to seek OCNS advice with regard to the physical protection requirements for the transport of spent fuel to a GDF.

Under the present safeguards arrangements, it can be assumed with a high degree of confidence that the spent fuel will be subject to safeguards on receipt in the GDF. Furthermore, it can be assumed that the presence of spent fuel in the GDF will result in a range of safeguards-related measures being applied to the GDF itself and its environs (surface and sub-surface).

It is not possible at this time to precisely define the safeguards impact on the design or operation of the GDF resulting from the disposal of spent fuel from EPR or any other reactor type. The IAEA is developing a generic approach which is likely to be made available for widespread Member State review and comment within the next two years. This will provide a clearer indication of the extent of the measures that could be applied to the UK's GDF.

There are no safeguards-relevant characteristics present in the EPR spent fuel that are likely to make it significantly different to spent fuel from any other civil reactor type.

5.4 Post-closure Safety

Context

As described earlier, the post-closure safety assessment is one component of the Environmental Safety Case (ESC) which is required to demonstrate safety of the disposal system in the long-term following backfilling, sealing and closing of the GDF. A successful post-closure safety case is based on an understanding of how the facility will evolve in the long term, and the ability to describe and quantify how this evolution may impact human health and the environment.

The long-term safety of geological disposal is achieved by a combination of engineered barriers and the natural geological barrier to isolate and contain the radioactivity in the wastes. The safety case typically includes an assessment of the radiological impacts of possible releases of radionuclides from this multi-barrier containment system as a result of natural processes.

In the case of spent fuel, this multi-barrier system includes the wasteform, the disposal canister, the buffer and the geological environment. Understanding of how these barriers contribute to safety is therefore an important aspect of the safety case. The requirements that need to be met in the safety case are specified in the Environment Agencies Guidance on Requirements for Authorisation (GRA) [49], and include a series of principles and requirements.

Requirement R6 of the GRA, which relates to radiological risk from a disposal facility after the period of authorisation, specifies a risk guidance level of 10^{-6} per year to a representative person, and the environment agencies expect that consistency with the risk guidance level is demonstrated through a risk assessment (commonly referred to as a post-closure performance assessment).

Previous work by RWMD on the disposal of spent fuel in the UK has included the development of a preliminary post-closure safety assessment [68]. The post-closure safety assessment of EPR spent fuel was undertaken by considering whether the disposal of EPR spent fuel would challenge any of the conclusions from this previous assessment. The assessment considered potential radiological impacts due to the groundwater and gas pathways, human intrusion and criticality. An assessment of the environmental impacts due to chemotoxic species contained in the spent fuel from the lifetime arisings of a single EPR was not carried out as information on such species were not available at the time of this assessment, but the quantity of toxic materials is expected to be insignificant. The assessment also included comparison of the characteristics of EPR spent fuel with spent fuel arising from operation of the PWR at Sizewell B. Quantitative assessment of risks to humans from the groundwater pathway was conducted using the GoldSim [69] code.

As noted above, the post-closure assessment is a component of the ESC, development of which is at an early generic stage. The assessment is based on a “generic” GDF design and host environment. This also includes assumptions regarding the nature of the geology and hydrogeology pertaining to the near- and far-field environments and regarding the biosphere. The ESC under development by RWMD is considered to be bounding, i.e. the assumptions are thought to be representative of the wide range of geological environments and disposal scenarios likely to be encountered in the UK.

Groundwater Pathway

Method for Groundwater Pathway

The disposability assessment has considered how spent fuel packages would evolve in the very long term post-disposal, recognising that radionuclides would be released only subsequent to a breach in a disposal canister. As noted in Section 5.1 decisions on overpack canister material have not yet been made. In line with previous work both copper and mild steel have been considered and detailed risk calculations performed for the bounding case of a canister manufactured from mild steel.

Subsequent to any canister failure, the radionuclides associated with the spent fuel would be able to leach into groundwater. The rate at which radionuclides are leached, in combination with the assumed properties of the geosphere, the behaviour of individual radionuclides and the mechanisms through which the radionuclides behave in the biosphere, may then be used to assess the subsequent time-dependency of risk to humans.

The assessment of long-term system performance in the GDA Disposability Assessment has been based on the assumed characteristics for a generic site for the GDF. Since the properties of any selected site necessarily would need to be consistent with meeting regulatory guidance values for risk, this assessment assumed the same groundwater flow rate and return time that would meet regulatory requirements when considering the inventory of legacy ILW.

In the GDA Disposability Assessment for the EPR, the quantitative assessment considered a single waste package containing four spent fuel assemblies all irradiated to 65 GWd/tU. A bounding case assessment was undertaken, based on the mild steel overpack.

For this bounding case, corrosion of a steel canister is initially assumed to result in a small penetration at the site of a defect, the resulting small hole offering some resistance to groundwater flow and radionuclide transport. It is assumed that this small hole eventually develops over time into a significant failure that is sufficiently large to offer no resistance to groundwater flow and radionuclide transport.

The time required for an initial penetration to arise at a defect depends on the thickness of the steel at that point and the assumed corrosion rate. For the purposes of assessment, the relevant thickness is assumed to lie between that of the canister walls (50mm) and possible thinning where the lid is welded (represented as a minimum thickness of 15 mm). The assumed corrosion behaviour is based on that developed in [70], which indicates an initial period of rapid aerobic corrosion, resulting in 11mm of penetration, followed slower, uniform anaerobic corrosion at a rate of 1µm y^{-1}. The combination of the corrosion behaviour and the range of possible thicknesses results in time periods for initial penetration of between 4,000 years and 39,000 years after closure of the GDF. The significant failure is assumed to occur after 39,000 years. Once water has penetrated the canister, a fuel dissolution rate of 1.5×10^{-5} kg/m^2yr was used, based on information from the Swedish waste management programme [71].

The canister corrosion performance for the steel canister may be compared with the estimates of the lifetime of a copper canister of the same thickness, which is reported to be in excess of 1,000,000 years [72]. Copper canisters with this performance have been adopted in the Canadian, Finnish and Swedish disposal programmes.

Results for Groundwater Pathway

Figure 12 illustrates the near-field flux for key radionuclides for a single steel disposal canister containing four spent fuel assemblies. This is the result of a 'Monte Carlo' simulation in which parameter uncertainty (e.g. canister failure time, sorption coefficients, groundwater travel time) have been sampled to calculate an 'expectation' value of radionuclide flux.

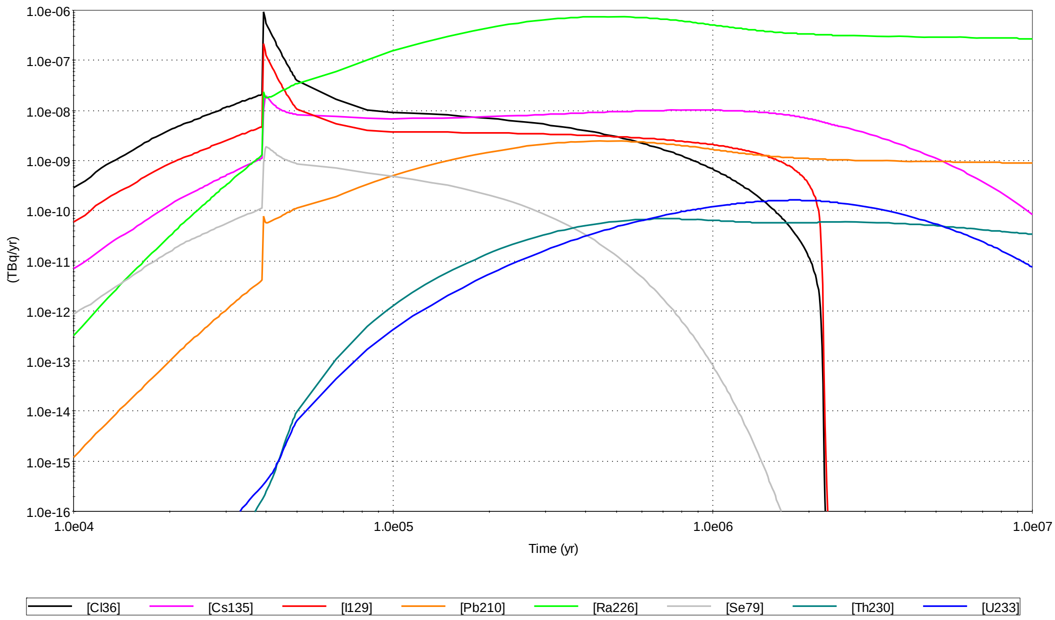

Figure 12 Near-field fluxes from a single steel overpacked disposal package containing four EPR spent fuel assemblies

When the overpack initially fails via a small defect, water infiltrates the disposal canister and is assumed to immediately contact the oxide fuel (no credit is claimed for containment by the cladding). The radionuclides begin to dissolve in this water as fuel is dissolved. In addition, a fraction (the Instant Release Fraction (IRF), see earlier discussion Section 5.2.1) of some radionuclides (e.g. 19% of Cl-36 and 13% of I-129) are dissolved immediately. The concentration of dissolved radionuclides builds up inside the overpack since they are only able to diffuse slowly out through the defect. When the overpack fails completely (i.e. a large hole), the accumulated dissolved contaminants are able to migrate more rapidly as shown by the spike occurring at about 40,000 years in Figure 12. This spike in flux represents the maximum flux for low sorption species such as Cl-36 and I-129. The remaining inventory of radionuclides is then released as the fuel is dissolved, over a period of about 2 million years. If a higher value were chosen for the instant release fraction then the spike in release from Cl-36 and I-129 would be expected to increase in proportion to the increase in IRF but the longer term release of these radionuclides would be reduced since there would be less inventory left in the fuel.

The result of the Monte Carlo risk calculations for the assessment of a single steel canister is illustrated in Figure 13. The peak risk for steel canister is calculated to be 9.9 x 10^{-11} per year occurring at 83,000 years. Compared to the shape of the near field flux curve, the 'spike' in release which is just discernable at 83,000 years, has been spread out due to dispersion and sorption processes as contaminants are transported through the geosphere. The spike is also delayed due to the time it takes to travel through the geosphere - for the generic geosphere used in this assessment the central value of the water travel time from the near field to the surface ranged from 30,000 years to 300,000 years.

Risks from disposal of spent fuel from a fleet of reactors would also be distributed through time due to differences that would be expected in the failure times for canisters and other parameters. The peak risk would scale in proportion to the number of canisters. On this basis, a risk of 5.3 x 10^{-7} per year for the lifetime arisings of a fleet of six EPR reactors each generating a lifetime total of 900 canisters is calculated. This is below the risk guidance level [49]. Therefore, the post-closure assessment has not identified any post-closure safety issues for the groundwater pathway.

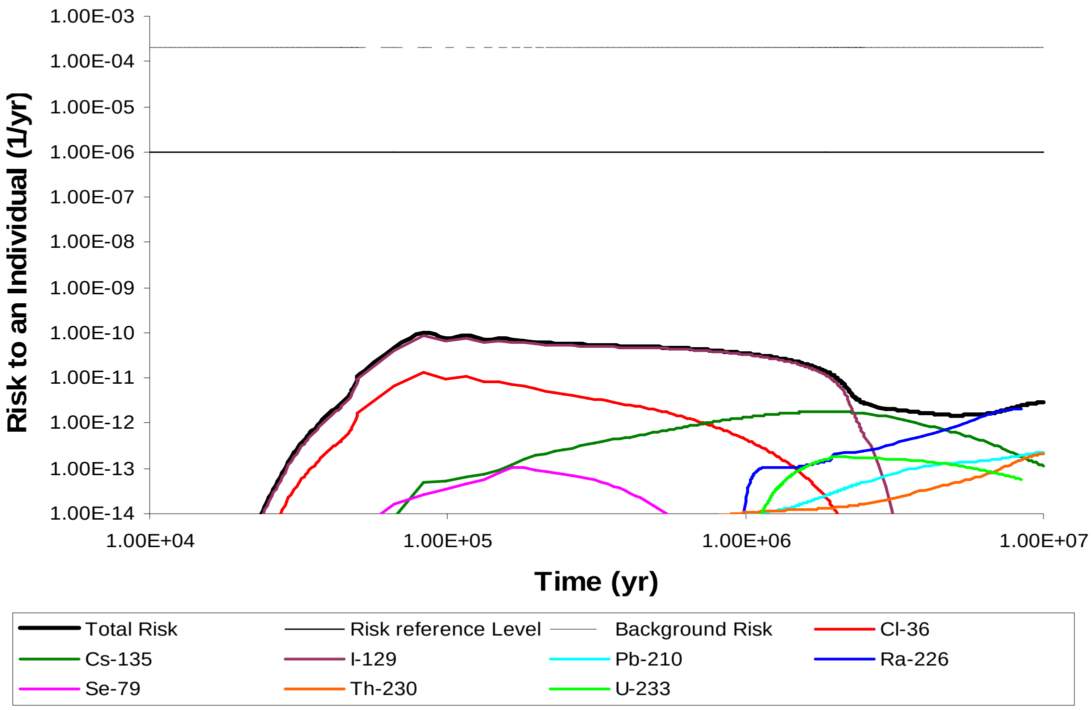

Figure 13 Total risk from a single EPR disposal package containing 4 spent fuel assemblies, assuming a steel canister

Results for Gas Pathway

It is assumed that both copper and steel spent fuel canisters would contain iron which could corrode to produce bulk gases. For the steel canister, which is the bounding case for the two canister options considered, disposal of 5,400 canisters (i.e. equivalent to the spent fuel from 6 EPRs) is estimated to lead to the production of 378 m^3y^{-1} of hydrogen, which is below the gas production threshold of 877 m^3y^{-1} identified as the limit for a surface flammability hazard in the generic post-closure performance assessment [73]. The assessment concluded that any radioactive gases associated with the spent fuel that would not represent a significant risk through the gas pathway, primarily due to the relatively short half-lives of such gases compared to the times required for any possible failure of the disposal packages.

Results for Human Intrusion

Regarding human intrusion, the siting process adopted by Government [74] has identified geological environments that should be avoided due to the presence of natural resources and which are, therefore, areas where human intrusion may occur. Addressing the GRA requirements for human intrusion requires that any practical measures to reduce the risk from human intrusion are implemented in the GDF and that potential risks from human intrusion are optimised. These requirements do not relate, therefore, to the fundamental feasibility of spent fuel disposal.

Criticality

EPR spent fuel would contain about 27kg of fissile material per disposal package. The inventory of fissile material per disposal canister for Sizewell B PWR spent fuel is approximately 17.6kg and that for UK AGR spent fuel is 24.1kg [68]. Therefore, the quantity of fissile material in EPR spent fuel is similar to that of UK AGR and Sizewell B PWR spent fuel. Reference [68] notes that there is no risk of criticality whilst fissile material remains in the canister. Furthermore, with low canister failure rates, there is a low probability of immediately adjacent canisters failing and, therefore, a low probability of the fissile material from more than one canister accumulating together.

The potential for fissile material accumulation out of the canisters and post-closure criticality also has been considered in the SR-Can safety assessment undertaken by the Swedish programme [12]. This assessment presented analyses of plutonium and uranium dissolution and migration rates through engineered barrier materials, and calculations of minimum fissile masses required for criticality in a canister, in the bentonite buffer and in a tunnel [75]. This study showed that insufficient Pu-239 could be accumulated in any location for criticality to occur prior to its decay to U-235. It also showed that uranium from many canisters would need to accumulate in one location for criticality to occur, and determined that uranium migration rates through barrier materials would be too slow for sufficient uranium to accumulate and form a critical mass on a timescale of a million years.

Based on these arguments, it has been concluded that a criticality safety case for the disposal of EPR spent fuel could be constructed once sufficient details of the design of the GDF are available. This would be considered further in future LOC assessments for EPR spent fuel, and in the general development of the GDF safety case.

Implications

On the basis of the information provided and what are expected to be conservative calculations of canister performance, it is estimated that the spent fuel from a fleet of six EPR reactors, packaged in mild steel canisters, would give rise to a risk below the risk guidance level based on these geological conditions.

RWMD is currently developing a Generic Disposal System Safety Case covering the Baseline Inventory of waste and wastes that may potentially arise in the future as set out in the Managing Radioactive Waste Safely White Paper [76]. RWMD is also considering an upper bound inventory reflecting the uncertainty around the Baseline Inventory, including the potential for wastes and spent fuel to arise from a new nuclear build power programme. This will provide information on the disposability of the various categories of waste and nuclear materials in a single facility. It is planned that the Generic Disposal System Safety Case will be published in September 2010 to support the geological disposal facility site selection and assessment process, as well as the ongoing provision of advice on the disposability of wastes, including those that may potentially arise in the future from a new nuclear build power programme.

The risks calculated for the disposal of spent fuel reflect the assumed performance of the proposed packaging methods. The analysis presented assumes packaging in a mild steel container and shows that even with this bounding case for canister material, risks remain below the risk guidance level. The assumed characteristics of the canisters and the disposal site mean that the calculated risk is not strongly influenced by the IRF.

RWMD recognises that the performance of disposal canisters would be an important element of a safety case for the disposal of spent fuel. Consequently, it is anticipated that RWMD would continue to develop the canister designs, with the intention of substantiating current assumptions and optimising the designs.

5.5 Summary of the Disposability of EPR Spent Fuel

5.5.1 General

Taking into consideration the analysis of the spent fuel covered in Section 3.4, the disposal package properties discussed in Section 5.2, the performance of the disposal packages during transport to and emplacement in the GDF discussed in Section 5.3 and the performance of the packages following sealing and closure of the GDF discussed in Section 5.4, packages containing spent fuel from an EPR have been judged to be potentially disposable.

While further development needs have been identified, these would represent requirements for future assessment under the Letter of Compliance process. These issues have been listed in Appendix B. The key conclusions regarding the disposability of spent fuel based on the information supplied by EdF/Areva for the GDA Disposability Assessment are highlighted in this section.

5.5.2 Inventory

The GDA Disposability Assessment for the EPR has shown that the principal radionuclides present in EPR spent fuel are the same as those present in existing UK legacy wastes and spent fuel, and, in particular, are consistent with the anticipated arisings from the existing PWR at Sizewell B. This conclusion reflects both the similarity of the designs of the EPR and existing PWRs, and the expectation that similar operating regimes would be applied.

EdF/Areva has indicated that the GDA Disposability Assessment for the EPR should assume that the reactor would operate to achieve a fuel assembly maximum burn-up of 65 GWd/tU. This burn-up is higher than that for the existing PWR at Sizewell B.

In practice, the average burn-up for EPR spent fuel assemblies would be less than 65 GWd/tU and this maximum would represent the extreme of a distribution of burn-up values for individual fuel assemblies. However, in the absence of detailed information on the distribution of burn-up between fuel assemblies, for the purposes of the GDA Disposability Assessment it has been conservatively assumed that the value of 65 GWd/tU applies uniformly to all of them. The adoption of a higher burn-up for the EPR, as compared to Sizewell B, would be expected to result in increased concentrations of radionuclides in the spent fuel.

An increased burn-up implies that the fuel is used more efficiently and that the volume of spent fuel to be disposed of would be smaller per unit of electricity produced. For example, an EPR operating for 60 years at 1.6 GW(e) would produce 3,600 spent fuel assemblies, which is equivalent to 37.5 spent fuel assemblies for every GW(e) year. In comparison, assuming the PWR at Sizewell B operates for 40 years at 1.188 GW(e) and produces 2,228 spent fuel assemblies, 46.9 spent fuel assemblies for every GW(e) year.

However, individual fuel assemblies would contain an increased concentration of fission products and higher actinides, leading to higher thermal output and dose-rates. This difference is recognised as an important consideration in the assessment of spent fuel from EPR, particularly in comparison with the spent fuel expected from Sizewell B.

For EPR spent fuel, radionuclide activity per disposal canister is about twice that of the Sizewell B fuel, which is to be expected because the burn-up of an EPR is assumed to be approximately twice that of Sizewell B (Section 3.4.3, Table 16). However, the detailed methodology has led to some significant differences in the radionuclide content of spent fuel from an EPR compared to that from Sizewell B, in particular:

- the use of pessimistic chlorine concentrations in precursor materials;
- inclusion of Ni-59 activities in Inconel 718 grid springs;
- the use of revised nuclear data libraries for Se-79;
- the impact of assumptions regarding the irradiation history on the estimates developed for Pu-238, Pu-242 and Am-243 activities.

5.5.3 Waste Packages

The GDA Disposability Assessment for the EPR has assumed that spent fuel would be over-packed for disposal. Under this concept, spent fuel would be over-packed into durable disposal canisters manufactured from suitable materials, which would provide containment for the radionuclide inventory over both the short-term (as required for transport and operational safety) and over the long-term (as required for post-closure safety). Although the canister material remains to be confirmed, the assessment has considered the potential performance of both copper and steel canisters. In both cases, the canister has provided sufficient containment.

The reference disposal concept for spent fuel used for providing disposability advice provides an initial criterion for the acceptable heat output from a disposal canister (Section 5.1). This is based on a conservative temperature limit intended to ensure that the performance of the bentonite buffer material to be placed around the canister is not damaged by excessive temperatures (the inner surface of the bentonite is restricted to a temperature of 100ºC). Based on a canister containing four EPR fuel assemblies, each with the maximum burn-up of 65 GWd/tU and adopting the canister spacing used in existing concept designs, it would require of order of 100 years for the activity, and hence heat output, of the EPR fuel to decay sufficiently to meet the existing temperature criterion.

It is acknowledged that the cooling period specified above is greater than would be required for existing PWR fuel to meet the same criterion. Nevertheless, it is noted that the period may be able to be reduced through refinement of the assessment inventory (for example by considering a more realistic distribution of burn-up), by reducing the fuel loading in a canister, or by consideration of alternative disposal concepts. For example, the estimated length of the interim storage period is 56 years for a disposal canister containing three 50 GWd/tU fuel assemblies.

A further issue associated with the higher burn-ups experienced by spent fuel compared to existing spent fuel is the impact that this may have on the properties of the fuel and cladding. The leaching of radionuclides from spent fuel is characterised by an initial elemental ‘instant release fraction’ (IRF), and by a more general dissolution rate. The IRF is the fraction of each radionuclide that is assumed to be readily released upon contact with groundwater and is influenced by the properties of the spent fuel. In the case of higher burn-up fuel, the increased irradiation of the EPR fuel would increase the IRFs as compared to that for lower burn-up fuel. Generally available information on the potential performance of higher burn-up fuel has been used to provide suitable IRFs for assessment. The IRFs estimated for EPR spent fuel lead to acceptable post-closure performance given the assumptions regarding the disposal concept and geological environment used in the GDA Disposability Assessment.

5.5.4 Impact on Design

The potential impact of the disposal of EPR spent fuel on the size of the GDF has been assessed. The assumed operating scenario for an EPR (60 years operation) gives rise to an estimated 900 disposal canisters, requiring an area of approximately 0.15 km^2 for the associated disposal tunnels. A fleet of six such reactors would require an area of approximately 0.9 km^2, excluding associated service facilities. This represents approximately 8% of the area required for the legacy wastes HLW and spent fuel, per reactor, and approximately 50% for the illustrative fleet of six reactors.

As discussed in Section 5.3.1, there are a range of disposal concepts that can be implemented for disposal of spent fuel, and these include concepts in which the footprint requirements are reduced for the equivalent quantities of waste (e.g. construction of disposal tunnels on two levels).

5.5.5 Transport Safety

RWMD is planning for the transport of packaged spent fuel to a Geological Disposal Facility and the subsequent emplacement of containers is at an early stage of development. Consequently, although the EPR spent fuel may significantly influence the necessary arrangements, for example through the need for additional shielding, it is judged that sufficient flexibility exists in the current concept to allow suitable arrangements to be developed.

5.5.6 Operational Safety

The operational safety assessment has considered the design basis faults that have been identified in operational safety assessments conducted to date. The disposal canister is a robust package that is expected to withstand plausible accidents within the disposal facility. The safety systems that will be included within the disposal facility will include gamma monitoring systems and interlocks to prevent worker exposure to the disposal canisters in regions of the disposal facility where the disposal canister is transferred from the transport container to an emplacement machine.

Arrangements for the emplacement of packaged spent fuel in a Geological Disposal Facility are at an early stage of development. Consequently, although the EPR spent fuel may significantly influence the necessary arrangements, for example through additional shielding requirements; it is currently judged that sufficient flexibility exists to allow suitable arrangements to be developed.

5.5.7 Environmental Considerations

No environmental issue that challenge the viability of the disposal of spent fuel from an EPR has been recognised.

5.5.8 Security and Safeguards

No security or safeguards issues were identified for EPR spent fuel that have not already been recognised for legacy spent fuel.

5.5.9 Post-closure Safety

The GDA Disposability Assessment has considered how spent fuel packages would evolve in the very long term post-disposal, recognising that radionuclides would be released only subsequent to a breach in a disposal canister. A limited sensitivity analysis has been performed, examining two different canister materials (copper and steel) and testing the influence of the assumed corrosion properties.

Subsequent to any canister failure, the radionuclides associated with the spent fuel would be able to leach into groundwater. The rate at which radionuclides are leached, in combination with the assumed properties of the host rock and the behaviour of individual radionuclides are then used to assess the potential risk to humans.

The assessment of long-term disposal system performance in the GDA Disposability Assessment has been based on the assumed characteristics for a generic UK site. Since the properties of any selected site necessarily would need to be consistent with meeting the regulatory risk guidance level, this assessment assumed the same site characteristics as assumed for the ILW assessment. On the basis of the information provided and what are expected to be conservative calculations of canister performance, it is estimated that the spent fuel from a fleet of six EPR reactors would give rise to an estimated risk below the risk guidance level based on these geological conditions.

The risks calculated for the disposal of spent fuel reflect the assumed performance of the proposed packaging methods. The sensitivity analysis demonstrated that while the calculated risk would be influenced by assumptions about the canister materials, for the assumed characteristics of the canisters and the disposal site, risks always remained below the regulatory guidance level, regardless of any impact that the high burn-up experienced by the fuel assemblies would have on the IRF.

RWMD recognises that the performance of disposal canisters will be an important element of a post-closure safety case for the disposal of spent fuel. Consequently, it is anticipated that RWMD will continue to develop canister designs, with the intention of substantiating current assumptions and optimising the designs.

6 CONCLUSIONS

RWMD has undertaken a GDA Disposability Assessment for the higher activity wastes and spent fuel expected to arise from the operation of an EPR. This assessment has been based on information on the nature of operational and decommissioning ILW, and spent fuel, and proposals for the packaging of these wastes, supplied to RWMD by EdF/Areva. This information has been used to assess the implications of the disposal of the proposed ILW packages and spent fuel disposal packages against the waste package standards and specifications developed by RWMD and the supporting safety assessments for a Geological Disposal Facility. The safety of transport operations, handling and emplacement at a Geological Disposal Facility, and the longer-term performance of the system have been considered, together with the implications for the size and design of a Geological Disposal Facility.

RWMD has concluded that sufficient information has been provided by EdF/Areva to produce valid and justifiable conclusions under the GDA Disposability Assessment. RWMD has concluded that ILW and spent fuel from operation and decommissioning of an EPR should be compatible with plans for transport and geological disposal of higher activity waste. It is expected that these conclusions eventually would be supported and substantiated by future refinements of the assumed radionuclide inventories of the higher activity wastes and spent fuel, complemented by the development of more detailed proposals for the packaging of the wastes and spent fuel and better understanding of the expected performance of the waste packages. At such later stages, RWMD would expect to assess, and potentially endorse, more specific and detailed proposals through the established Letter of Compliance process for assessment of waste packaging proposals.

On the basis of the GDA Disposability Assessment for the EPR, RWMD has concluded that, compared with legacy wastes and spent fuel, no new issues arise that challenge the fundamental disposability of the wastes and spent fuel expected to arise from operation of such a reactor. This conclusion is supported by the similarity of the wastes to those expected to arise from the existing PWR at Sizewell B. Given a disposal site with suitable characteristics, the wastes and spent fuel from the EPR are expected to be disposable.

Appendix A: The Letter of Compliance Process

Introduction

The Letter of Compliance assessment process has been developed by RWMD to provide advice to waste packagers on the disposability of proposed conditioned waste packages. The process is compatible with regulatory guidance on the management of higher activity wastes on nuclear licensed sites[19]. The LoC assessment provided by RWMD is expected to contribute to the reasoned arguments incorporated into the licensee's Radioactive Waste Management Case. The LoC process is described fully in RWMD guidance materials[20].

In the case of higher activity waste coming forward from the EPR it is expected that the GDA Disposability Assessment commissioned by EdF/Areva will be used by potential operators to guide their selection of waste conditioning and packaging technologies. Issues identified in the GDA Disposability Assessment where further information is required are expected to be addressed in the future by potential operators through LoC interactions.

LoC Stages

LoC interactions typically occur at three stages prior to the operation of a waste packaging plant; at Conceptual stage, Interim stage prior to placement of major design and build contracts and at a Final stage before active operations.

At the Conceptual stage it is to be expected that the Disposability Assessment will be in outline form only, but sufficiently developed to judge the overall feasibility of the packaging concept. The Conceptual stage Disposability Assessment is envisaged to be a development of the Disposability Assessment developed for GDA but specific to an operator's packaging proposals.

As the packaging concept and plant is developed through Interim and Final stages it is to be expected that the Disposability Assessment will become progressively developed such that at the Final stage it is robustly supported by all necessary design and research and can be presented to the site operator (site licensee) as a Disposability Case. In line with regulatory guidance it is envisaged that the Disposability Case presented in the Final stage Assessment Report will be adopted by the site licensee and incorporated into the Radioactive Waste Management Case for wastes under consideration.

At the Conceptual and Interim stages the RWMD Assessment will in addition to the Disposability Assessment, include RWMD's technical evaluation of the proposed waste package. This will highlight areas where further development or information is required and any actions necessary to take the disposability assessment to the next stage. Any issues flagged as requiring resolution or where further information, research or development is needed, are denoted as Action Points. All Action Points are given a unique identifier for tracking purposes and state at which stage the issue should be closed out.

LoC Bibliography

The Letter of Compliance process is well established and is supported by a suite of published guidance that operators will find helpful in undertaking LoC interactions with RWMD. The following documentation, published within the suite of Waste Package

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[19] HSE/EA/SEPA, *The Management of Higher Activity Radioactive Waste on Nuclear Licensed Sites, Part I The Regulatory Process, Guidance from the HSE, EA and SEPA to Nuclear Licensees*, 2007

[20] NDA RWMD, *Guide to the Letter of Compliance Process*, WPS/650, March 2008

Specification and Guidance Documentation (WPSGD), in particular is recommended as relevant based on the issues raised within the GDA Disposability Assessment.

- *Introduction to the Waste Package Specification and Guidance Documentation,* WPS/100
- *Waste Package Quality Management Specification*, WPS/200
- *Specification for 500 litre Drum Waste Package*, WPS/300
- *Waste Package Data and Information Recording Specification*, WPS/400
- *Waste Package Data and Information Recording Specification: Explanatory Material and Guidance,* WPS/850
- *Guidance on the Structure and Format of Waste Product Specifications*, WPS/620
- *Guidance on Environmental Conditions during Storage of Waste Packages*, WPS/630
- *Guidance on the Monitoring of Waste Packages during Storage*, WPS/640
- *Guide to the Letter of Compliance Process*, WPS/650
- *Guidance on the Preparation of Letter of Compliance Submissions*, WPS/908
- *Guidance Note on the Use of Organic Polymers for the Encapsulation of Intermediate Level Waste*, WPS/901
- *Guidance Note on the Packaging of Filters*, WPS/905

Copies of WPSGD are available on request from NDA RWMD.

Appendix B: Issues to be Addressed during Future LoC Interactions

During the assessment work described in Sections 3, 4 and 5, a number of requirements and/or opportunities for further development were identified, typically highlighted as issues that would need to be addressed in the future through the established Letter of Compliance (LoC) process. The identification of these areas for future development is entirely consistent with expectations at this stage of development of the proposals for the packaging of waste and spent fuel considered in the GDA Disposability Assessment and the relatively high-level assessments performed. The information submitted by EdF/Areva has been judged to be sufficient for the purposes of assessment at this time.

This Appendix summarises the main areas where potential development needs have been identified during the GDA Disposability Assessment.

As discussed in Section 2.2, it is expected that the GDA Disposability Assessment would be followed, at an appropriate time, by further interactions with potential EPR operators on more detailed and developed proposals for the packaging of waste and spent fuel. It is likely that such interactions would be governed by the LoC process, as summarised in Appendix A. A range of information and guidance has been developed by RWMD, describing the requirements of the LoC process. This information and guidance is also summarised in Appendix A.

The potential development needs identified in this Appendix would be expected to contribute to fulfilling the requirements of the LoC process for the relevant wastes or materials. However, this Appendix should not be assumed to represent a comprehensive basis for fulfilling the requirements of the LoC process.

Section B.1 details issues relating to the packaging of ILW, whereas Section B.2 details those relating to the packaging of spent fuel. In some cases, identified development needs are specific to a particular packaging option for operational ILW and can be ignored if the relevant option is not adopted.

B.1 ILW

B.1.1 Proposed Approach to ILW Management

An operator would be expected to provide further information on the waste management approaches adopted for particular plant. Issues that have been identified through the GDA Disposability Assessment for more detailed consideration in the future include a need for the operator to:

- provide further information on proposals for the management of RCCAs;
- confirm the absence of, or provide proposals for, any ILW residues from the incineration of evaporator concentrates;
- confirm whether wastes are intended to be transported in IP-2 or Type B packages.

B.1.2 Information on ILW Characteristics

An operator would be expected to provide further information on the waste characteristics. Issues that have been identified through the GDA Disposability Assessment for more detailed consideration in the future include a need for the operator to:

- provide information on the grade and composition of materials used in an EPR, for example stainless steel, taking account of the nitrogen impurities in the steel and provide information on the form of tritium, C-14, Ar-39, Cl-36 and Se-79 in activated metals;
- provide detailed information on the chemical composition of the wastes, including toxic element content;
- confirm that the contents of waste packages meet the “contents specifications” for transport, for example that masses of both deuterium and beryllium in the waste packages are less than 1.8g;
- provide information on the form of tritium and carbon-14 in the waste packages to support realistic modelling of their release during transport and operation;
- provide information on the products that would be generated from waste degradation, for example the rates of volatile amines produced by radiolysis and thermal degradation of anion-exchange resins.

B.1.3 Information on ILW Wasteform and Conditioning Process

An operator would be expected to provide information on the wasteform and on the methods used to condition waste prior to its consignment to a GDF. Issues that have been identified through the GDA Disposability Assessment for more detailed consideration in the future include a need for the operator to:

- consider the use of alternative conditioning matrices, for example organic resins (vinyl ester styrene systems) as an alternative to the use of epoxy resins to immobilise ion-exchange resins, as envisaged for EPR01, and to define the polymer processing envelope in terms of both satisfactory wasteform performance and plant operation;
- demonstrate that any grout used for conditioning of waste suitably infiltrates the waste and immobilises particulates successfully;
- consider the use of alternative approaches to grouting waste, such as the use of a calcium sulpho-aluminate cement to ensure that grout will set satisfactorily to counter the negative impact that the presence of boron and zinc in sludges (EPR05) may have on cement curing;
- provide data on the mass transport, thermal conductivity, and gas generation and pressurisation properties of the wasteforms;
- define the boundaries of the formulation envelope for grout and polymer encapsulants and demonstrate that the plant operational envelope falls with this;
- provide information on the use of capping grouts, for example confirm either that an inactive capping grout is applied to the top surface of all decommissioning ILW wasteforms, prior to lidding of the waste container, or that loose particulate material would not be present and that a capping grout is unnecessary.

B.1.4 Information on ILW Packaging and Container Design

An operator would be expected to provide information on the container and waste package design. Issues that have been identified through the GDA Disposability Assessment for more detailed consideration in the future include a need for the operator to:

- include information of the material composition of waste containers, including additives and reinforcement, for Reference Case C1 and C4 Casks;
- provide details of waste package handling features;

- confirm that package identifiers would be applied that would be compatible with current requirements;
- demonstrate that wastes to be transported as IP packages meet the LSA requirements of IAEA transport regulations.

B.1.5 Information on ILW Package Performance

An operator would be expected to provide further information on expected waste package performance under accident conditions. Issues that have been identified through the GDA Disposability Assessment for more detailed consideration in the future include a need for the operator to:

- mitigate the risk of mechanical damage to containers during packaging and handling of wastes, for example the potential for damage to concrete casks during waste compaction after placement of ILW in the concrete casks;
- demonstrate whether the wasteform or the waste container would be load bearing in the case of waste packages being stacked one on another;
- provide results from modelling or test work to better define the damage and the release from waste packages under impact accidents, and the heat loading and the release from the waste packages from fire accidents;
- consider the deterioration in the mechanical strength of waste packages owing to storage, and the impact of such deterioration on the accident performance;
- provide information on the performance of non-standard packages under impact events and fire.

B.2 Spent Fuel Issues

At the current stage of development of plans for spent fuel waste management, RWMD is taking the lead in developing designs of disposal canisters. These designs are an integral part of the disposal concept which would be determined by the geological host environment. RWMD would continue to work with potential operators to ensure that they are aware of the latest thinking in respect of disposal canisters.

Spent fuel issues identified during the GDA Disposability Assessment and which would need to be addressed through LoC interactions are primarily associated with understanding of the waste characteristics. In any future submission under the LoC process, the operator would be expected to:

- build confidence in the expected levels of cladding failure as a result of adoption of Zircaloy M5;
- provide information on the distribution of burn-up around the average and maximum and on irradiation history, to support modelling of radionuclide inventories;
- provide information on the properties of spent fuel following irradiation at high burn-up to support assumptions regarding long-term integrity of spent fuel, including estimation of the IRFs;
- provide information that could be used to evaluate the potential for the spent fuel canister to be subject to significant gas pressurisation under both normal and fire accident conditions.

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