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ASSESSMENT REPORT

Generic Design Assessment: Disposability Assessment for Wastes and Spent Fuel arising from Operation of the Westinghouse Advanced Passive Pressurised Water Reactor (AP1000)

Part 1: Main Report

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DOCUMENT HISTORY

STATUS	REVISION	DATE	COMMENTS
Draft 1	1	18 September 2009	Draft 1 is the first full draft, and is issued for accuracy checking by Westinghouse.
Issue 1	2	11 November 2009	Issue 1 incorporates changes in response to comments from Westinghouse. Issue 1 has been issued as final version of this report.
Issue 2	3	15 January 2010	Issue 2 removes protective marking from the report. This was undertaken after confirmation from Westinghouse that the information contained in the report is not considered to be commercially confidential. No other changes to the report were made.

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 Westinghouse Electric Company LLC, which is seeking an initial endorsement of the Westinghouse Advanced Passive Pressurised Water Reactor (AP1000) 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 and decommissioning. 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 in providing such advice) on the disposability in a Geological Disposal Facility of any proposed arisings of higher activity wastes and spent fuel[3].

In accordance with regulatory guidance, Westinghouse 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 and decommissioning of an AP1000. The reported assessment of the disposability of the higher activity wastes and spent fuel from the AP1000 is based on information on wastes and spent fuel, and proposals for waste packaging supplied by Westinghouse, supplemented as necessary by relevant information available to RWMD.

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

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 AP1000. 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 endorsing 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

Westinghouse has provided information on the higher activity wastes and spent fuel expected to arise from an AP1000 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. Westinghouse has provided information for the following three types of operational ILW:

- Primary Circuit Filters, including filters used in the Chemical and Volume Control System (CVCS), Spent Fuel pond cooling System (SFS), the Liquid Radwaste System (WLS) and the Solid Radwaste System (WSS);
- Primary Resins: including CVCS Mixed Bed Resin, CVCS Cation Bed Resin, SFS Demineraliser and Inorganic Resin from WLS;
- Secondary Resins: including Condensate Polisher Resins and Steam Generator Blowdown Material.

Westinghouse 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 Westinghouse 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 this value will represent the maximum of a range of burn-up values for individual fuel assemblies.

Proposals for Waste Packaging

Westinghouse has put forward proposals for the packaging of operational ILW based on the current practice for similar wastes in the UK. The Primary Circuit Filters would be cement grouted into a RWMD standard $3m^3$ Box. To package the Primary and Secondary Resins, Westinghouse proposes to cement encapsulate them in UK standard $3m^3$ Drums. The $3m^3$ Boxes and Drums would need to be transported in a reusable shielded transport overpack to meet the requirements of the transport regulations.

The proposals for the packaging of decommissioning ILW are also based on the use of UK standard waste containers consistent with RWMD standards and specifications. Westinghouse proposes to grout these wastes into standard $3m^3$ Boxes. Again, the $3m^3$ Boxes would need to be transported in a reusable shielded transport overpack to meet the requirements of the transport regulations.

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 insert. For consistency with previous assessments of the disposal of spent fuel undertaken by RWMD, it has been assumed that each disposal canister would contain 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 Westinghouse on the radionuclide inventories of the identified wastes and spent fuel has been used to derive assessment inventories for the proposed waste packages and spent fuel disposal packages. In some cases, to ensure a full coverage of potentially significant radionuclides, it has been necessary to supplement the information supplied by Westinghouse using information available to RWMD. The assessment inventories are intended to characterise the range of waste 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 waste 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 waste 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 reactor 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 resin 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 fuel 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 Westinghouse, 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 AP1000 and of existing PWRs, and the expectation that similar operating regimes would be applied.

The adoption of higher burn-up for the AP1000, 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 AP1000 (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 to more closely match the levels in existing legacy wastes, if required.

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 proposal 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 overpack has been assessed to eliminate potential challenges to the dose-rate limits set out in the IAEA Transport Regulations.

The proposed use of cement grout for waste conditioning conforms to existing practices for similar wastes in the UK and is expected to produce packages that would be compliant with existing RWMD standards and specifications.

The assessment of long-term disposal system performance in the GDA Disposability Assessment has been 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[5], 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 AP1000 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 AP1000 in a site of the type described would be consistent with meeting the regulatory risk guidance level. The consideration of a fleet of reactors does not alter this conclusion.

Overall, the proposals for the packaging of operational and 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 AP1000 and, to illustrate the potential implications of constructing a fleet of such reactors, consideration also has been given to a fleet of nine AP1000 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.

The potential impact of the disposal of AP1000 operational and decommissioning ILW on the size of a Geological Disposal Facility has been assessed. It has been concluded that the

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

necessary increase in the ‘footprint area’ is small, corresponding to approximately 65m of vault length for each AP1000. This represents approximately 1% of the area required for the legacy ILW, per reactor, and less than 10% for the illustrative fleet of nine AP1000 reactors. This is in line with previous estimates for potential new build reactor designs[6].

Assessment of Spent Fuel Packages

Westinghouse has indicated that the GDA Disposability Assessment for the AP1000 should assume that the reactor would operate with uranium dioxide fuel 4.5% enriched in U-235 to achieve a maximum fuel assembly average burn-up of 65 GWd/tU. This burn-up is higher than that achieved for the existing PWR at Sizewell B.

In practice, the average burn-up for AP1000 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 AP1000.

The GDA Disposability Assessment for the AP1000 has assumed that spent fuel would be overpacked 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. Overpacking 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 AP1000 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 AP1000 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 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. The sensitivity of the cooling period to fuel burn-up has been investigated by consideration of

[6] 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.

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 temperature criterion.

RWMD planning for the transport of packaged spent fuel to a Geological Disposal Facility and the subsequent emplacement of the containers is at an early stage of development. Consequently, although the AP1000 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.

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 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 AP1000 fuel would increase the IRF as compared to that for lower burn-up fuel. Generally available information[7] 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 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, 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 nine AP1000 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.

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

The potential impact of the disposal of AP1000 spent fuel on the size of the Geological Disposal Facility has been assessed. The assumed operating scenario for an AP1000 (60 years operation) gives rise to an estimated 640 disposal canisters, requiring an area of approximately 0.11 km^2 for the associated disposal tunnels. A fleet of nine such reactors would require an additional area of approximately 1 km^2, excluding associated service facilities. This represents approximately 6% of the area required for legacy HLW and spent fuel per AP1000 reactor, and approximately 55% for the illustrative fleet of nine AP1000 reactors. This is in line with previous estimates for potential new build reactor designs[6].

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[8]. 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 AP1000 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 AP1000. 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 Westinghouse. 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 Westinghouse 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 AP1000 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 AP1000, 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

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

expected to arise from the existing PWR at Sizewell B. Given a disposal site with suitable characteristics, the wastes and spent fuel from the AP1000 are expected to be disposable.

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 Westinghouse Electric Company LLC, which is seeking an initial endorsement of the Westinghouse Advanced Passive Pressurised Water Reactor (AP1000) 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[9] 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, Westinghouse has requested that the Radioactive Waste Management Directorate (RWMD) of NDA provides advice on the disposability of the higher activity wastes and spent fuel expected to arise from the operation of an AP1000. The reported assessment of the disposability of the higher activity wastes and spent fuel from the AP1000 is based on information on wastes and proposals for waste packaging supplied by Westinghouse, supplemented as necessary by relevant information available to RWMD.

Comprehensive details of the information supplied to RWMD by Westinghouse, 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. The 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

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[9] 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.

reactors, RWMD would expect to assess more specific and detailed proposals through the existing Letter of Compliance assessment process [5].

The assessment has been undertaken in response to the purchase order from Westinghouse dated 3 February 2009 (Purchase Order 4500280026, RWMD Document Reference #10200692) 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 Westinghouse and NDA, dated 8 September 2008.

1.2 Objectives

The purpose of the 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 AP1000. The assessment has been commissioned by Westinghouse 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 and requesting parties, including Westinghouse, 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 AP1000, 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 AP1000;

- comment on initial proposals by Westinghouse 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 AP1000 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 the 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 spent fuel, 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 nine AP1000 reactors noting that the actual impact on the UK's waste inventory as a result of new nuclear power stations will depend on the mix of reactor types and size of construction programme.

This document describes the GDA Disposability Assessment for the AP1000 and presents the results of the assessment. In particular, this report describes the higher activity wastes and spent fuel expected to be generated through operation and decommissioning of the AP1000, describes options for conditioning and packaging these materials, and identifies issues and further information requirements from the perspective of transport and disposal, which will 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 AP1000, 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 nine AP1000s, which would provide generating capacity of approximately 10 GW(e) (nine AP1000s would produce approximately 9.9 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 AP1000 reactor programme.

1.4 Document Structure

This GDA Assessment Report for the AP1000 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 Westinghouse proposals have been assessed and the assessment methodology applied;
- Section 3 provides an overview of the AP1000, the assumptions regarding operation of an AP1000 used in the GDA Disposability Assessment and

summarises the inventory, packaging proposals, disposal package numbers and disposal package characteristics for AP1000 ILW and spent fuel;

- Section 4 describes the assessment of AP1000 operational and decommissioning ILW;
- Section 5 describes the assessment of AP1000 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 AP1000 has considered the conditioning and packaging proposals put forward by Westinghouse. 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.

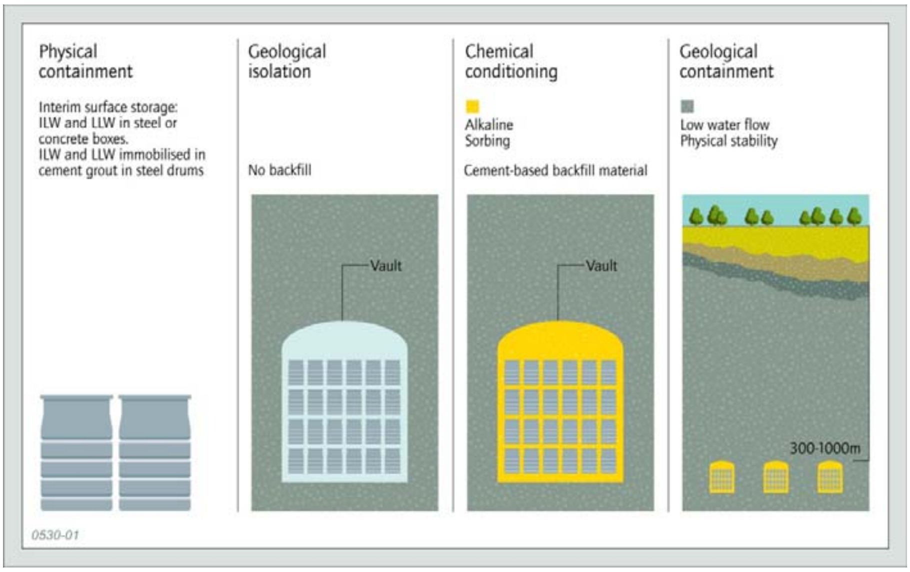

Figure 1 Concept for the disposal of ILW

A reference disposal concept for spent fuel is also used in the provision of disposability advice [11]. Under this concept, spent fuel would be overpacked 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 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. 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].

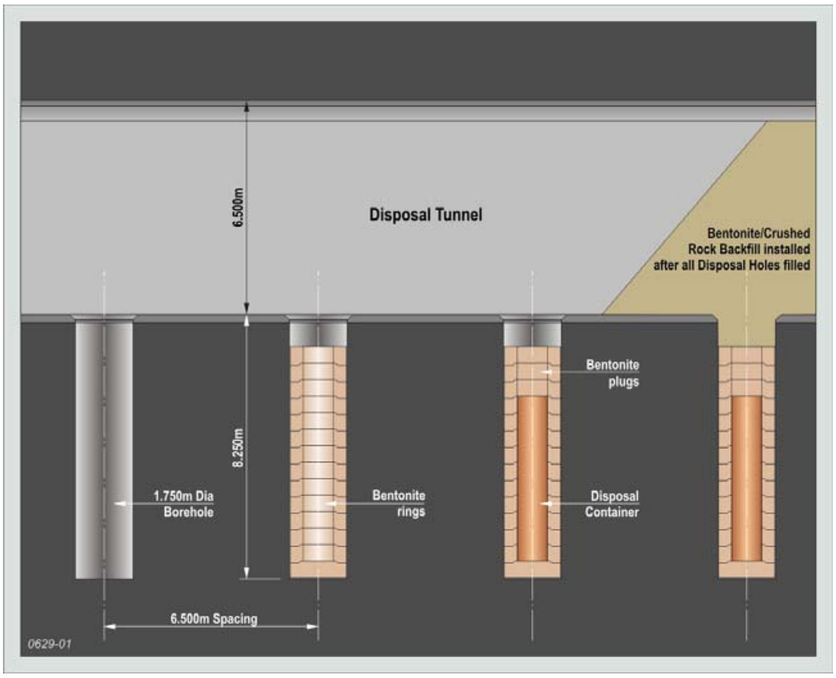

Figure 2 Concept for the disposal of spent fuel illustrating the disposal holes and emplacement of disposal canisters

2.2 Assessment Approach and Constraints

2.2.1 Approach followed for GDA Disposability Assessment

Overview

The GDA Disposability Assessment of the AP1000 was based on a protocol [6] agreed with Westinghouse, 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 type and package numbers.

Conclusions have been drawn regarding the suitability of Westinghouse proposals through comparison of information on AP1000 ILW and spent fuel with legacy wastes and spent fuel, 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 Westinghouse 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 AP1000 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 AP1000 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[10]". For spent fuel, separate

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

consideration was given to the characteristics of the spent fuel assemblies and the disposal package characteristics. Work under this stage used information supplied by Westinghouse, supplemented by additional information generated by RWMD. In particular, knowledge of the characteristics of radioactive waste arising at the Sizewell B pressurised water reactor (PWR) was used to add value to the GDA Disposability Assessment.

The Nature and Quantity evaluation was used to collate data on the operational and decommissioning ILW, and the spent fuel from the AP1000, 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 AP1000, ILW package and spent fuel disposal canister release fractions have been developed for the ILW streams and spent fuel, and packaging scenarios proposed by Westinghouse.

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 waste 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 AP1000 with existing assessments of RWMD reference disposal concepts. These assessments provide the basis for judging the potential disposability of AP1000 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 human 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 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 AP1000 OPERATION, WASTES AND SPENT FUEL, PACKAGING PROPOSALS AND PACKAGE CHARACTERISTICS

This section provides a summary of the information used in the GDA Disposability Assessment for the AP1000. RWMD used the information supplied by Westinghouse, 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 AP1000 (Section 3.1);
- assumptions regarding the operation of an AP1000 (Section 3.2);
- description of the higher activity radioactive waste streams and spent fuel that will be generated through operation and decommissioning of an AP1000 (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 AP1000 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 AP1000 Design and Operation

The AP1000 is an evolutionary PWR design with a rated thermal power of 3400 MW and an electrical power output of approximately 1117-1154 MW(e), depending on site-specific factors.

The AP1000 evolutionary design is based on experience from operation of Light Water Reactors (LWR) worldwide, primarily those incorporating the most recent technologies. The primary system adopts the most reliable design features of both civil and naval PWRs and introduces many passive safety features that simplify the management of faults including loss of cooling events. The AP1000 design received US NRC Certification in January 2006 and four reactors are under construction in China.

In PWRs such as the AP1000, 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 AP1000 design is furnished with a two-loop, primary circuit system composed of a reactor vessel that contains the fuel assemblies, a pressuriser including control systems to maintain system pressure, two reactor coolant pumps per loop, one steam generator per loop, one hot leg and two cold legs per loop (Figure 3).

In the reactor coolant system, the primary cooling water is pumped through the reactor core and the tubes inside the steam generators, in two 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 feedwater to the steam generators. A schematic of the whole heat transfer and electricity production system in a PWR is provided in Figure 4.

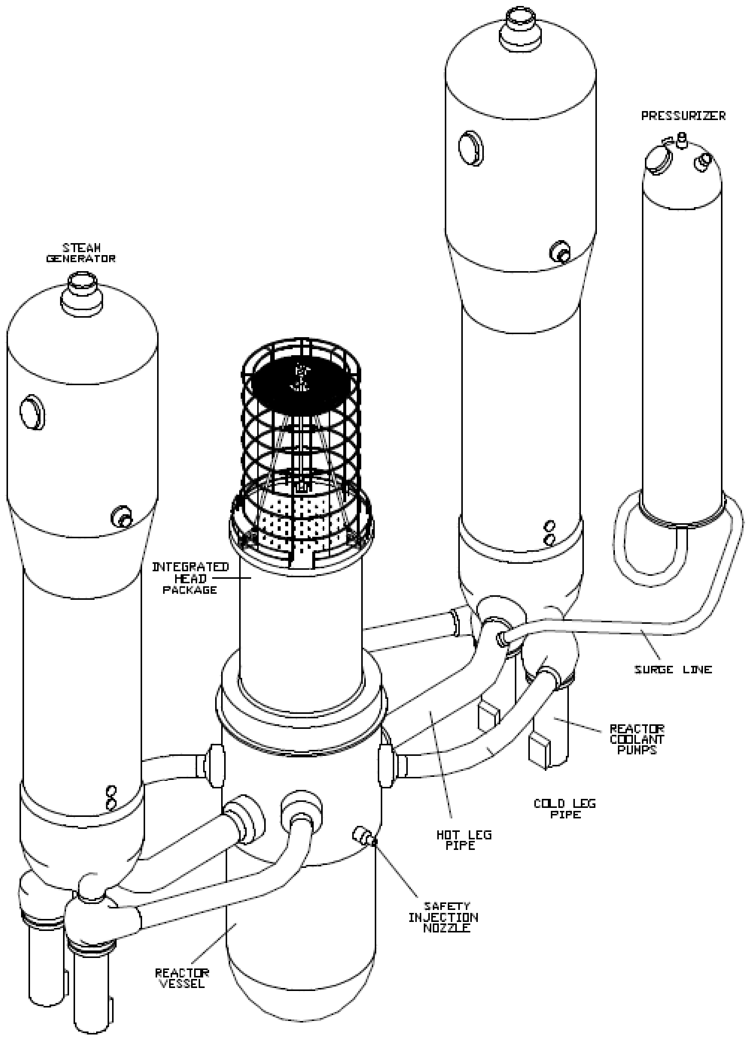

Figure 3 Principal primary circuit components of an AP1000. Figure reproduced from [15].

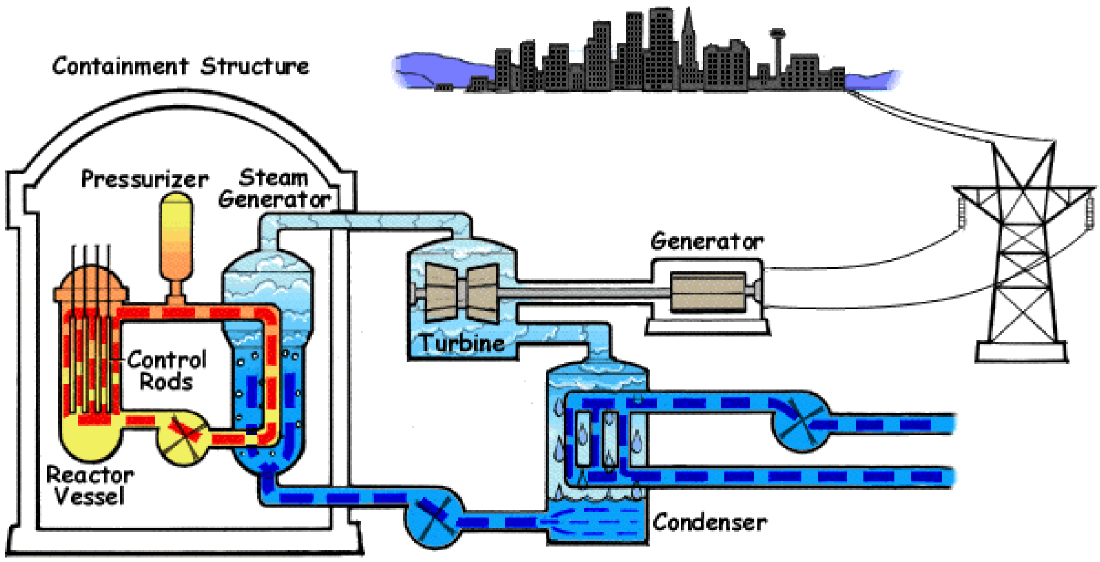

Figure 4 Principal systems of a PWR. Figure reproduced from [16].

3.2 Assumptions

The GDA Disposability Assessment for the AP1000 was based on the following assumptions:

- The AP1000 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. Sixty-four 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 AP1000 would commence in the UK is uncertain. In the GDA Disposability Assessment for the AP1000, estimates of time-dependent properties, e.g. those related to radioactive decay, are assessed from 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)[11] of 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.

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[11] 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 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.

- The fuel used in the AP1000 would be manufactured from freshly mined uranium (i.e. not reprocessed uranium) enriched to an initial U-235 content of 4.5% and would contain only a nominal quantity of U-236 prior to irradiation.
- It is assumed that ILW and spent fuel from the AP1000 will arrive at the GDF in a packaged state, ready for disposal.

3.3 ILW Streams, Packaging Assumptions, Package Numbers and Characteristics

3.3.1 Operational ILW Streams and Packaging Assumptions

Westinghouse has indicated that three operational ILW streams would arise from normal operation of an AP1000:

- AP01 (Primary Circuit Filters), including filters used in the Chemical and Volume Control System (CVCS), Spent Fuel pond cooling System (SFS), the Liquid Radwaste System (WLS) inlet and outlet, and the Solid Radwaste System (WSS) resin fines filters;

- AP02 (Primary Resins): including CVCS Mixed Bed Resin, CVCS Cation Bed Resin, SFS Demineralizer, and Resins (organic and inorganic) from the WLS;

- AP03 (Secondary Resins): including Condensate Polisher Resins and Steam Generator Blowdown Material.

To package the Primary Circuit Filters, it is assumed that about 80 filters, with a raw waste volume of $0.9m^3$ would be cement grouted into a RWMD standard $3m^3$ Box. To accommodate all the filters arising from a 60 year operational lifetime for a single AP1000, 24 such waste packages would be produced. For transport, the $3m^3$ Boxes would be carried inside a Standard Waste Transport Container (SWTC) which is being developed by RWMD to transport such waste packages. The SWTC is proposed to be manufactured in steel with two shielding thicknesses, 70mm and 285mm. It has been calculated that the $3m^3$ Boxes would need to be transported in a SWTC-285 to meet the IAEA Transport Regulation dose rate requirements.

To package the Primary and Secondary Resins, Westinghouse propose to cement encapsulate them in $3m^3$ Drums. Westinghouse indicates that on the basis of grouting trials, an adequate wasteform may be generated if the resins occupy 25% of the wasteform by volume. To accommodate all the Primary and Secondary Resin arising from a 60 year operational lifetime for a single AP1000, 1020 (for Primary Resins) and 190 (for Secondary Resins) waste packages would be produced. The $3m^3$ Drums would need to be transported in a SWTC-285 to meet the IAEA Transport Regulation dose rate requirements. Both the $3m^3$ Box and $3m^3$ Drum are standard RWMD waste packages and are illustrated in Figure 5.

3.3.2 Decommissioning ILW Streams and Packaging Assumptions

The reference decommissioning assumption 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, Westinghouse 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 assumed that all concrete would be LLW after 40 years storage, this remains to be proven. Nevertheless, given the compact nature of an AP1000, RWMD estimates that the volume of any such ILW is unlikely to exceed 100m^3, and would be unlikely, therefore, to cause significant concern 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, and prior to Care and Maintenance (Stage 2).

Decommissioning ILW has been defined in two broad waste streams as follows:

- AP04 (ILW Steel), which consists of the stainless steel associated with pressure vessel internals including: Radial shield Baffle, Barrel, Neutron Pads and Formers; Upper and Lower Axial Shield, Loop pipes, Radial shield insulation and Liner. These steels are expected to have plate-like structures with a thickness of the order of 0.01m. The raw waste volume for this waste stream is 20 m^3. Westinghouse proposes to grout this waste into 3m^3 Boxes (about 20 packages being required to accommodate the whole waste stream). The 3m^3 Boxes would need to be transported in a SWTC-285 to meet the IAEA Transport Regulation dose rate requirements.
- AP05 (Pressure Vessel), which consists of ferritic steel, associated with the mid-height section of the pressure vessel and from the internal 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. The raw waste volume for this waste stream is 39 m^3. Westinghouse proposes to grout this waste into 3m^3 Boxes (about 40 packages being required to accommodate the whole waste stream). The 3m^3 Boxes should be transportable in an SWTC-70 and still satisfy the IAEA Transport Regulation dose rate requirements.

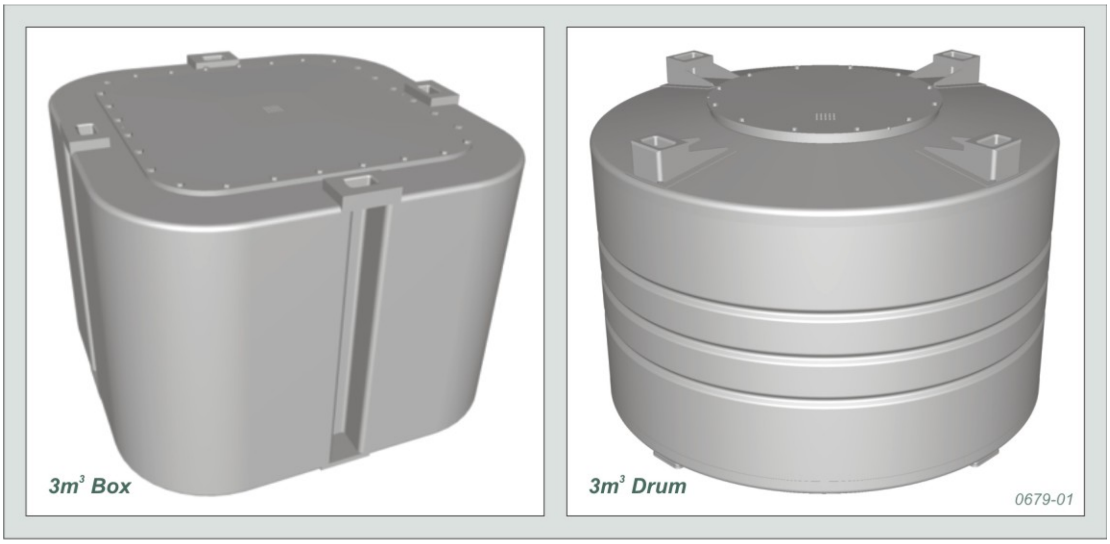

Figure 5 Illustration of the 3m^3 Box and 3m^3 Drum as proposed for packaging of operational and decommissioning ILW from the AP1000

3.3.3 ILW Package Numbers and Characteristics

The information supplied by Westinghouse on the radionuclide inventories of the identified wastes has been used to derive assessment inventories for the various proposed waste packages. To ensure a full coverage of potentially significant radionuclides it has been necessary to supplement the information supplied by Westinghouse with information available to RWMD [17]. The assessment inventories are intended to characterise the range of waste package inventories, taking account of variability between packages and uncertainties.

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.

Westinghouse supplied data on the raw waste volumes and package numbers for each waste stream that would be expected to occur in one year. Two values were quoted for each stream:

- waste arisings for a year in which the waste arisings are at a maximum, which is assumed to be the result of fuel defects leading to an increase in resin production; the maximum value is assumed to be realised, on average, every five years, and the distribution of fuel defects over the 60-year life of the reactor is expected to be random;

- waste arisings for a normal year, which is assumed to be the result of operation of the reactor without the result of fuel defects, and which is referred to as “average” packages in our evaluation.

The package numbers assumed by Westinghouse were checked by RWMD against the proposed raw waste volumes [17].

The material composition of the waste was also provided by Westinghouse.

Operational ILW (AP01, AP02, AP03)

Data on activities per package for various radionuclides were provided by Westinghouse. The estimates for filters (AP01) were relevant to average packages and, following checking against the published numbers in the UK Radioactive Waste Inventory [18] were found to be consistent with concentrations found in relevant waste streams from Sizewell B [17].

However, for the resins (AP02 and AP03), the values supplied for Cs-134 and Cs-137 were significantly higher than the values found in other sources. For example, the activity of Cs-137 in the primary resins was 8.06 TBq/m^3, compared to 0.03 TBq/m^3 in the Sizewell B ILW resin stream (3S12). Since Cs-137 is used as the reference radionuclide from which other, difficult to measure, fission products are determined, the pessimistic approach will extend to many other radionuclides.

The high numbers in the inventory occur because Westinghouse's estimate of activities in the resins from the AP1000 was based on a conservative estimate of 0.25% fuel cladding defects, and, therefore, the values were considered to be relevant to the maximum packages. Such fuel failure rates are believed to be conservative, based on operational experience, but lower values have not been provided at this time. Overall it seems likely that the radionuclide content declared for the AP1000 resin wastes contains significant conservatisms.

Although Westinghouse was able to supply separate volume data for the Primary and Secondary resins, and radionuclide data for the Primary resins, reliable radionuclide data for the Secondary resins were not available. For assessment purposes, it was conservatively assumed that the radionuclide concentrations in the Secondary resins were the same as in the Primary resins.

In order to derive average package activities for the resins, Westinghouse developed the scaling factors shown in Table 1.

Table 1 Factors used to calculate average package activities from maximum package activities for AP1000 Resins, relevant radionuclides are those recognised by RWMD as relevant to long-term management [19]

Radionuclide	Average / Maximum Activity	Relevant Radionuclide?
Mn-54	8.81E-01	yes
Fe-55	9.12E-01	yes
Co-60	3.91E-01	yes
Sr-90	1.04E-01	yes
Cs-134	3.20E-02	yes
Cs-137	5.08E-02	yes

Co-58	6.77E-01	no
Cr-51	8.13E-01	no
Fe-59	8.56E-01	no
Br-84	5.79E-01	no
Rb-88	3.86E-02	no
Sr-89	5.86E-02	no
Sr-91	1.48E-01	no
Y-91	6.82E-06	no
I-131	2.61E-02	no
I-132	5.28E-02	no
I-133	3.19E-02	no
I-134	9.43E-01	no
I-135	9.16E-02	no
Cs-136	1.84E-03	no
Ba-137m	5.13E-02	no
Ba-140	5.29E+00[1]	no

[1] The factor for Ba-140 is greater than 1 because it is based on an estimate of the activity of Ba-140 for an average package that is known to be based on conservatisms in excess of the conservatism used to generate the maximum package activity.

Most of the data on activities provided by Westinghouse were for short-lived radionuclides, and did not include information on actinides:

- for the filters, activities were given for 35 radionuclides, but just seven (Mn-54, Fe-55, Co-60, Sr-90, I-129, Cs-134 and Cs-137) are in the NDA set of 'relevant radionuclides' [19];
- the resins stream lists 58 radionuclides, of which 11 are relevant (the seven listed for filters, plus Zn-65, Ag-110m, Cs-135 and Ce-144).

Experience at Sizewell B and at other light-water reactors around the world suggests that many more radionuclides will arise in practice than those for which Westinghouse supplied data. Therefore, an 'enhancement' exercise was undertaken by RWMD to estimate the likely inventories of other radionuclides, based on the declared inventories.

The enhancement exercise estimated package activities based on three approaches [17]:

- use of recommended scaling factors for various types of LWR waste, versus Co-60 and/or Cs-137 [20], applying the recommended scaling factor for an appropriate waste type were such data where available;
- use of scaling factors derived by RWMD for this study [20], from information in the 2007 UK Radioactive Waste Inventory [18] for equivalent waste streams from Sizewell B:
    - o stream 3S03 spent cartridge filters (ILW) was used for the filters;
    - o stream 3S12 CVCS resins and spent resins (ILW) was used for the resins.

the specific activities in the Sizewell B streams were used to calculate scaling factors, versus either Co-60 (for actinides and activation products) or Cs-137 (for fission products); Nb-93m, Nb-94 and Mo-93 were scaled relative to both Co-60 and Cs-137, and the maximum taken;

- uranium data for the filters were estimated by undertaking a set of calculations using ORIGEN for PWR uranium fuel irradiated to 55 GWd/tU to determine scaling factors for the uranium isotopes against Pu-239.

Once these estimates had been calculated, average and maximum activities for each radionuclide in each waste stream were selected:

- for filters, the average package activity was the maximum of the value originally stated in the AP1000 datasheet and those derived from the three methods above (Figure 6);
- for filters, the maximum package activity was derived by multiplying the average package activity by 10;
- for resins, the average package activity was the maximum of the value originally stated in the AP1000 datasheet (adjusted by the average-to-maximum factors shown in Table 1), and the two scaling factor approaches described above;
- for resins, the maximum package activity was the maximum of the value originally stated in the AP1000 datasheet and the two scaling factor approaches described above.

These approaches are illustrated in Figures 6 for filters and Figures 7 and 8 for resins (average and maximum packages).

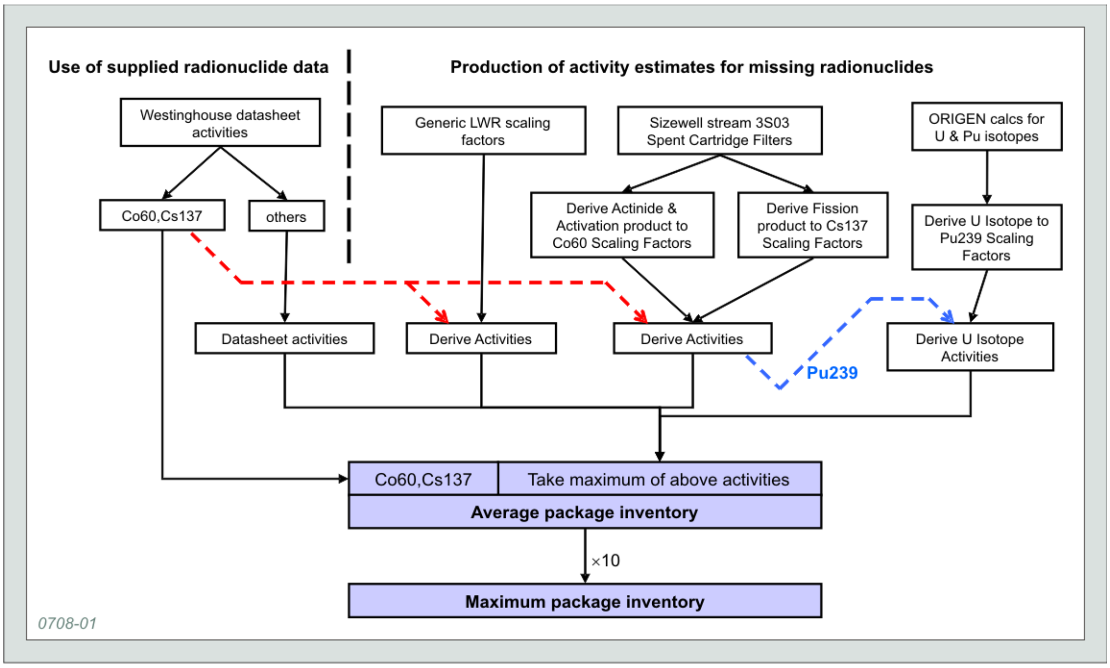

Figure 6 Methods used for estimating radionuclide inventories in average and maximum packages of AP1000 Filters

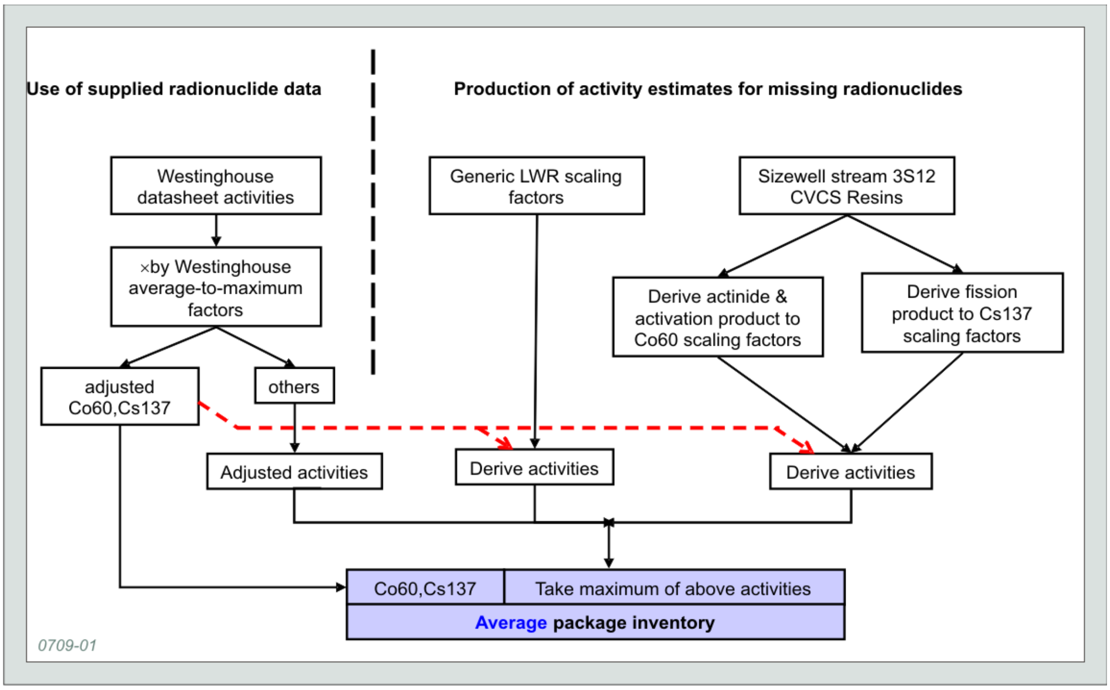

Figure 7 Methods used for estimating radionuclide inventories in average packages of AP1000 Resins

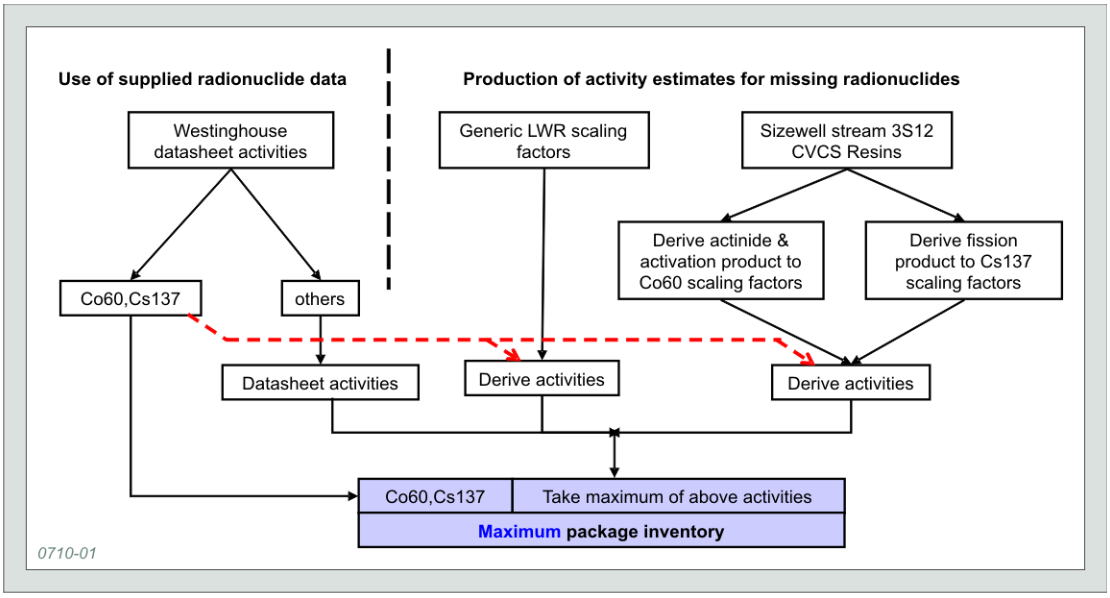

Figure 8 Methods used for estimating radionuclide inventories in maximum packages of AP1000 Resins

It is probable that the waste package inventories of I-129 in resins and filters derived by the above methodology are conservative. In the supporting references considered in [20], few positive determinations of I-129 were made in resins or filters, so that detection-limit values had to be used to derive the scaling factors. This produces conservative scaling factors, and hence conservative I-129 package inventories. This data limitation has probably also resulted in conservative Tc-99 inventories being declared for the filters.

Decommissioning ILW

For decommissioning ILW, Westinghouse supplied data on the raw waste masses and volumes, package numbers and material composition for each waste stream. Information provided by Westinghouse on the composition of the ILE Nuclear MaterialWSteel waste stream indicated that the steel was Type 304 stainless steel, and included mass fractions for nine elements (Table 2).

However, a comprehensive understanding of the radionuclide inventory resulting from activation of steel requires consideration of 82 elements. In previous studies, RWMD has developed upper-bound 82 element concentration data for a range of reactor materials including Type 304 stainless steel. For some elements, the RWMD upper-bound concentrations were derived from information reported for other steel, and, in a few cases, the concentrations were hypothetical, derived from the earth's crustal abundances. Despite these shortcomings, in the absence of specific information for the other 73 relevant radionuclides for the Westinghouse steels, RWMD estimates were considered to be the best available and were adopted for use in the AP1000 Disposability Assessment.

Table 2 Upper bound elemental mass fractions for Type 304 stainless steel assumed in the Disposability Assessment - values shown with a pink background were supplied by Westinghouse submission and those in yellow are based on RWMD data

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	1.00E-03	V	8.77E-04	Zr	1.58E-05	Ce	7.74E-04	Os	4.45E-08
N	1.00E-03	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	7.48E-01	Ru	1.00E-07	Sm	4.02E-07	Au	5.00E-07
Ne	1.27E-05	Co	5.00E-04	Rh	2.56E-05	Eu	7.66E-08	Hg	5.00E-05
Na	2.99E-05	Ni	1.10E-01	Pd	1.03E-06	Gd	4.92E-07	Tl	6.00E-05
Mg	1.00E-03	Cu	7.33E-03	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	4.50E-04	Ge	7.00E-04	Sn	1.46E-04	Er	7.43E-07	U	3.09E-06
S	3.00E-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		

ILW Steel Radionuclide Inventory (AP04)

Radionuclide activities for the ILW Steel waste stream were estimated by RWMD by conducting neutron activation analyses using FISPACT-2007, and based on the elemental composition of Type 304 stainless steel shown in Table 2, a reactor load factor of 90%, and an estimate of the energy-dependent neutron flux experienced by the significant components of the waste stream [17].

The most irradiated components of the decommissioning ILW are the baffle, the core barrel, and the neutron pads and formers. The baffle is closest to the fuel assembly and sees the highest neutron flux. Although the core barrel is only 0.2m further out from the core boundary than the baffle, it only receives about 10% of the fast neutron flux experienced by the baffle [21].

Since the production of certain activation products show a non-linear dependence upon irradiation, RWMD decided to calculate the activation of the barrel rather than scale the results for the baffle by the ratio of neutron flux. The difference between the barrel and the neutron pads is insignificant, and therefore, two sets of activation calculations were made:

- a high-flux set for the baffle;
- a lower-flux set for the barrel, which was assumed to apply also to the neutron pads.

To calculate reliable activation rates requires the energy dependence of the neutron flux as well as its absolute magnitude. The crudest form of energy dependence that may be used to obtain activation rates is the three-group form that defines the flux in the Thermal (<0.6eV), Epithermal (0.6 eV to 1 MeV) and Fast (>1 MeV) energy ranges. Within each of these three broad ranges typical fine-group forms apply, i.e.

- Thermal – Maxwellian;
- Epithermal – 1/*E*;
- Fast – fission spectrum.

The enhancement used modern neutron fluence / flux data for the energy ranges 0.1 to 1 MeV and >1MeV [21], supplemented by some older published information [22]. The proposed three-group flux data were built up from data for four broad energy groups as shown in Table 3 to Table 5, and these data were used for the FISPACT calculations of radionuclide inventory.

Table 3 Representative neutron flux experienced by the Baffle in an AP1000 for four broad energy groups

Energy Group	Group Flux ($n/cm^2/s$)	Basis of Flux Value
Fast Group (>1 MeV)	7.3E+13	>1MeV Fluence quoted for US 900 MW PWR in Table VIII of [21]
Upper Epithermal Group (0.1 to 1 MeV)	8.5E+13	>0.1MeV Fluence quoted for US 900 MW PWR in Table VIII of [21]
Lower Epithermal Group (0.6 eV to 0.1 MeV)	1.3E+14	Scaled from ratio of Upper to Lower Epithermal fine group flux RWMD have for a typical PWR cladding
Thermal Group (<0.6 eV)	1.5E+13	Shroud flux for Westinghouse PWR as read from Figure 5.1 of [22]

Table 4 Representative neutron flux experienced by the Barrel in an AP1000 for four broad energy groups

Energy Group	Group Flux ($n/cm^2/s$)	Basis of Flux Value
Fast Group (>1 MeV)	6.8E+12	>1MeV Fluence quoted for US 900 MW PWR in Table VIII of [21]
Upper Epithermal Group (0.1 to 1 MeV)	1.1E+13	>0.1MeV Fluence quoted for US 900 MW PWR in Table VIII of [21]
Lower Epithermal Group (0.6 eV to 0.1 MeV)	1.6E+13	Scaled from ratio of Upper to Lower Epithermal fine group flux RWMD have for a typical PWR cladding
Thermal Group (<0.6 eV)	3.0E+12	Core Barrel flux for Westinghouse PWR as read from Figure 5.1 of [22]

Table 5 Representative three-energy-group collapsed flux data for the Baffle and Barrel of an AP1000

Energy Range	Neutron Flux in core centre plane (n/cm^2/s)	
	Baffle	Core Barrel
Fast Group (>1 MeV)	7.3E+13	6.8E+12
Epithermal Group (0.6 eV to 1 MeV)	2.1E+14	2.7E+13
Thermal Group (<0.6 eV)	1.5E+13	3.0E+12

The estimates of activities made using FISPACT were compared with those provided by Westinghouse to provide confidence in the RWMD calculations. The comparison is shown in Table 6. The FISPACT calculations for the two nickel isotopes give results that are very close to the values given in the datasheets. The close match for these significant radionuclides and the reasonable match for Fe-55 and Co-60 provide confidence in the method and neutron flux data used in the FISPACT calculations.

However, the FISPACT calculations estimate the C-14 inventory to be almost two orders of magnitude higher than the Westinghouse datasheet. Following discussion, Westinghouse and RWMD agreed that the RWMD estimate was the appropriate value to use in the GDA Disposability Assessment. The inventory of C-14 associated with decommissioning ILW has been estimated through activation calculations based on an assumed concentration of the relevant pre-cursor species (primarily nitrogen). Westinghouse proposed that the nitrogen content of the stainless steels be taken as 1000ppm. In the absence of other data this is thought to be a conservative assumption.

Table 6 Comparison of ten-year-cooled specific activities for AP1000 Decommissioning Steel ILW calculated by Westinghouse with those produced by RWMD using FISPACT

Radionuclide	Westinghouse total TBq	FISPACT total TBq	FISPACT / datasheet activity ratio
C14	2.27E+00	1.99E+02	87.809
Fe55	3.32E+04	2.38E+04	0.718
Co60	1.10E+04	2.26E+04	2.056
Ni59	2.40E+02	2.59E+02	1.078
Ni63	3.84E+04	4.37E+04	1.137
Nb93m	1.10E+03	3.25E+02	0.296

The specific activity in an average package was calculated by summing the FISPACT activities in the baffle and barrel and dividing by the total volume of the *whole* ILW Steel waste stream, i.e. 19.66 m^3 (153.31 t). Therefore, in an average package, the more-active material from the baffle and barrel is diluted with the less-active material from the 'other internals'. The activities for Fe-55, Co-60, Ni-59, Ni-63 and Nb-93m, were taken directly from the datasheet provided by Westinghouse.

The maximum package was taken to contain 0.9 m^3 of steel from the baffle, with all the activities taken from the FISPACT calculations.

Pressure Vessel Radionuclide Inventory (AP05)

Rather than performing a new set of activation calculations for the pressure vessel stream, which was expected to have a low activity, the radionuclide inventories were estimated by scaling the inventories from the FISPACT data for the barrel (which is the part of the radial shield that is closest to the pressure vessel) by the ratio of the Fe-55 activity in the pressure vessel provided by Westinghouse to the Fe-55 activity in the barrel calculated by RWMD using FISPACT.

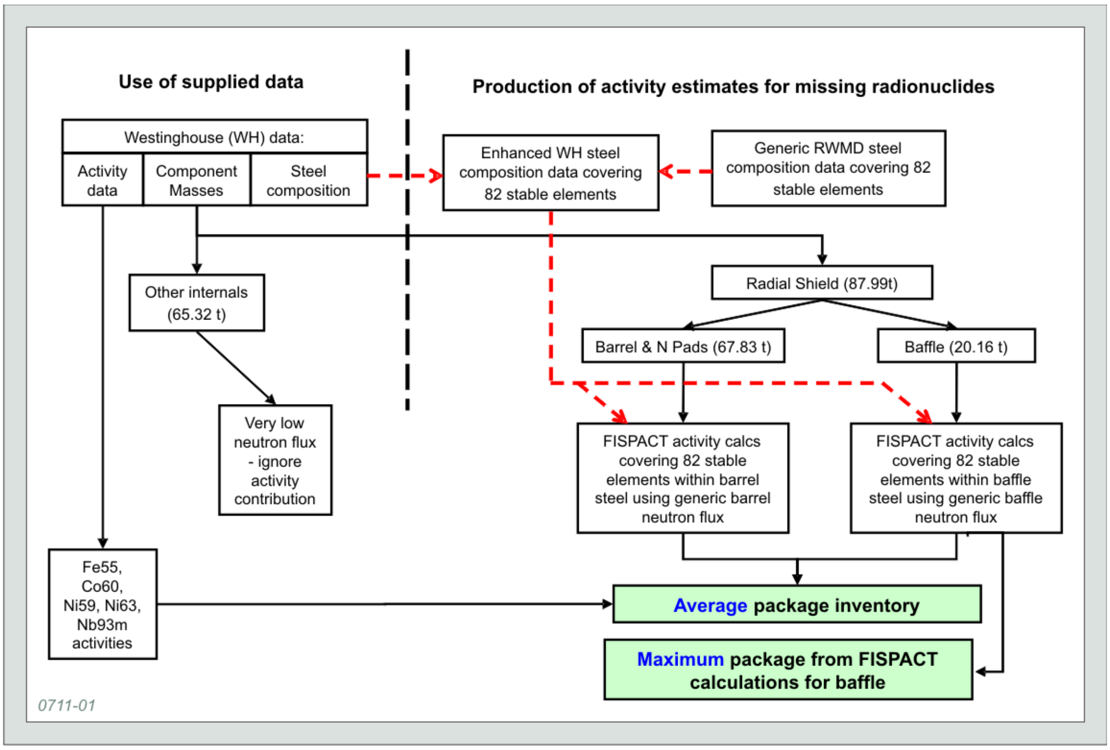

Figure 9 Methods used for estimating radionuclide inventories in average and maximum packages of AP1000 Decommissioning Steel ILW

ILW Package Characteristics

The AP1000 ILW waste package radionuclide-related parameters and waste quantities (package numbers and total packaged volume) are given in Table 7. 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). In the absence of specific information on interim storage plans for operational waste, the conservative assumption of prompt dispatch to the GDF was adopted. 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.

Table 7 AP1000 Operational and Decommissioning ILW Waste Stream Data (1) (2) (3)

Waste Stream (4)	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 Transport Container (mSv/hr)
AP01	$3m^3$ Box	24	78.5	3.95E-06	2.02E+00	5.62E+00	4.21E-01	8.52E-05
AP02	$3m^3$ Drum	1020	2661.0	1.75E-05	7.30E-01	1.07E+00	1.16E-01	6.55E-05
AP03	$3m^3$ Drum	191	498.3	1.75E-05	7.30E-01	1.07E+00	1.16E-01	6.55E-05
AP04	$3m^3$ Box	22	71.4	1.47E-02	1.51E+03	9.14E+01	8.30E+00	3.10E-03
AP05	$3m^3$ Box	43	140.7	5.48E-06	1.04E+00	7.05E-02	7.75E-03	3.56E-06
TOTALS		1300	3449.9					

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=10 for AP01, M≤20 for AP02 & AP03, M ≤9 for AP04, M=2 for AP05.

(3) Dose rate refers to that 1m outside and SWTC-285 in the case of AP01, AP02, AP03 & AP04 and 1m outside an SWTC-70 in case of AP05.

(4) See Section B3.1 for description of AP01 to AP03 waste streams, and Section B3.2 for description of AP04 and AP05 waste streams .

3.3.4 Comparison of AP1000 ILW with Sizewell B ILW

In order to place the information on the radioactivity of the ILW that would arise from an AP1000 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 1,188 MW(e) [23] and an assumed operating life of 40 years, whereas the AP1000's electrical power output is 1,117 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 [13].

Decommissioning ILW is the dominant source of many radionuclides in the estimated inventory for AP1000, with most of this activity being concentrated in the stainless steel waste stream AP04. The radionuclide with the highest total activity in both operational and decommissioning ILW (from AP1000) is Ni-63 and it is estimated that there is approximately 2,000 times more of this radionuclide in the decommissioning waste than in the operational waste. Similar (slightly larger) factors apply to Ni-59 and Co-60. The C-14 content of the AP1000 decommissioning waste at 199 TBq is about 400 times that in the operational waste.

The activity of AP1000 stainless steel decommissioning ILW (stream AP04) is compared with the activity of the equivalent Sizewell B PWR waste [13] (2007 National Inventory stream 3S306) in Table 8. The basis for Table 8 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, AP1000 1.117 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);

- the radionuclides considered in Table 8 are the top 10 most active in the AP1000 wastes for which estimates were also available for the Sizewell B PWR wastes;
- the cell colouration displayed in the sixth column of Table 8 is used to indicate the closeness of the agreement that presents the ratio of AP1000 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 8 Comparison of radionuclide activities for Stainless Steel decommissioning ILW from an AP1000 with equivalent ILW stream from Sizewell B PWR (3S306)

Nuclide	Sizewell B 3S306 (TBq)	AP1000 St_Steel (TBq)	Sizewell B 3S306 (TBq per GW(e).yr)	AP1000 St Steel (TBq per GW(e).yr)	{AP1000 St Steel} / {3S306}
Ni-63	3.35E+04	3.12E+04	7.09E-01	4.66E-01	6.57E-01
H-3	8.77E+01	1.06E+03	1.86E-03	1.58E-02	8.48E+00
Nb-93m	3.84E+02	3.30E+02	8.13E-03	4.92E-03	6.06E-01
Ni-59	3.23E+02	2.40E+02	6.85E-03	3.58E-03	5.22E-01
Co-60	8.04E+02	2.13E+02	1.70E-02	3.18E-03	1.87E-01
C-14	1.21E+02	1.99E+02	2.56E-03	2.96E-03	1.16E+00
Mo-93	1.21E+00	4.49E+01	2.56E-05	6.70E-04	2.62E+01
Fe-55	1.64E+02	1.66E+01	3.48E-03	2.47E-04	7.10E-02
Nb-94	4.04E+00	4.58E+00	8.56E-05	6.83E-05	7.98E-01
Tc-99	1.21E-01	1.14E+00	2.57E-06	1.70E-05	6.62E+00

As can be seen from Table 8, with the exception of H-3 and Co-60, the activities of the six radionuclides with the highest activities are similar (within a factor of three). Like H-3 the total activity of Mo-93 and Tc-99 is considerably higher in the AP1000 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 AP1000 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 AP1000 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 AP1000 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 the AP1000 decommissioning waste streams is comparable with that for Sizewell B.

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

3.4.1 Description of Spent Fuel

The AP1000 fuel design is based on the 17x17 XL (14 foot) design used successfully at plants in the US and Europe. The core of an AP1000 consists of 157 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 Zirlo tubes, made up of 264 fuel rods, 24 control rod guide thimbles and a central instrument thimble, as illustrated in Figure 10. Zirlo is an advanced alloy of zirconium with a typical major element composition by mass of Zr – 97.4%, Nb – 1.2%, Sn – 1.1% and trace iron and oxygen. Zirlo 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.

The rods are held in bundles by 10 spacer grids distributed at roughly uniform intervals up the 4.6m free height of the rods (5 additional grids consisting of 4 intermediate flow mixing grids and one so-called 'P-Grid' are also part of the fuel assembly). 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 4.795m. The 25 guide thimbles are joined to the grids and the top and bottom nozzles. Twenty four of the guide thimbles are the locations for the rod cluster control assemblies (RCCAs – the control rods) or burnable poison rods. The remaining central thimble may contain neutron source rods, or 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 eight spacer grids and four intermediate flow mixing grids distributed along the fuelled section of the assembly are made from low neutron capture Zirlo, whereas the top and bottom spacer grids and P grid are made from the nickel alloy Inconel 718.

The AP1000 fuel assembly and fuel rod are illustrated in Figure 10 and some additional dimensional information is provided in Table 9.

The fuel rods consist of uranium dioxide (UO_2) pellets, typically 4.5% enriched in U-235, stacked in a Zirlo cladding tube plugged and seal welded to encapsulate the fuel.

The stack of UO_2 pellets extends over a height of 4.267m 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.

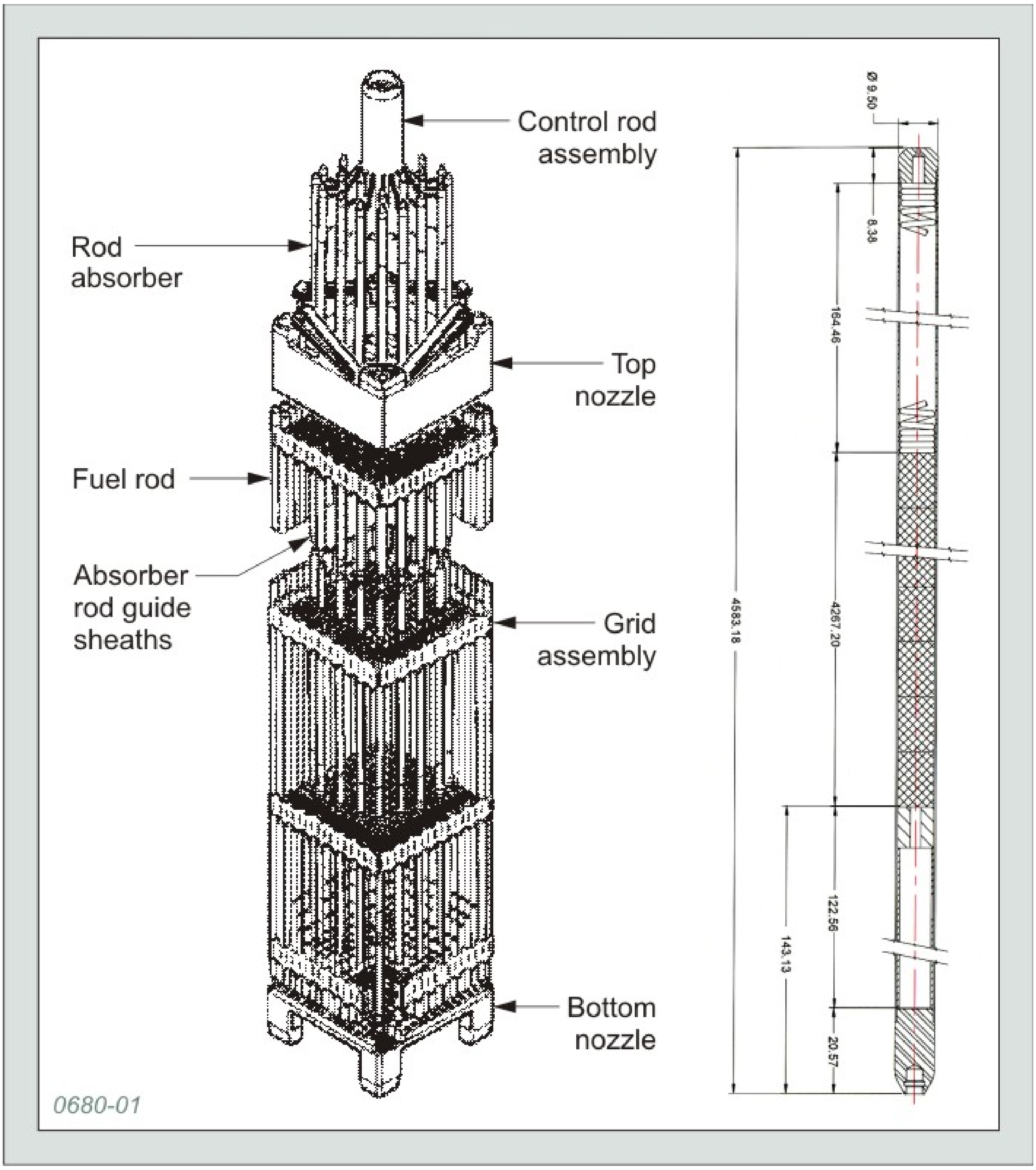

Figure 10 **Components of an AP1000 fuel assembly and separate control rod assembly (left) and a single AP1000 fuel rod (right)**

Table 9 Dimensional information for AP1000 fuel assemblies and rods

Fuel Assembly	
External maximum section (mm x mm)	214 × 214
Maximum length (mm)	4795
Active length (mm) (Average, at 20 °C)	4267
Fuel Rod	
Number of fuel rods	264
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 Westinghouse design of fuel, the disposal canister would be required to be 5.2m in length (Figure 11). 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 overpack. 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 the high neutron capture material ‘such as Kobesh’.

Although the quantitative analyses conducted in the GDA Disposability Assessment for the AP1000 are based on certain disposal concept assumptions, the implications of alternative disposal concepts also have been considered.

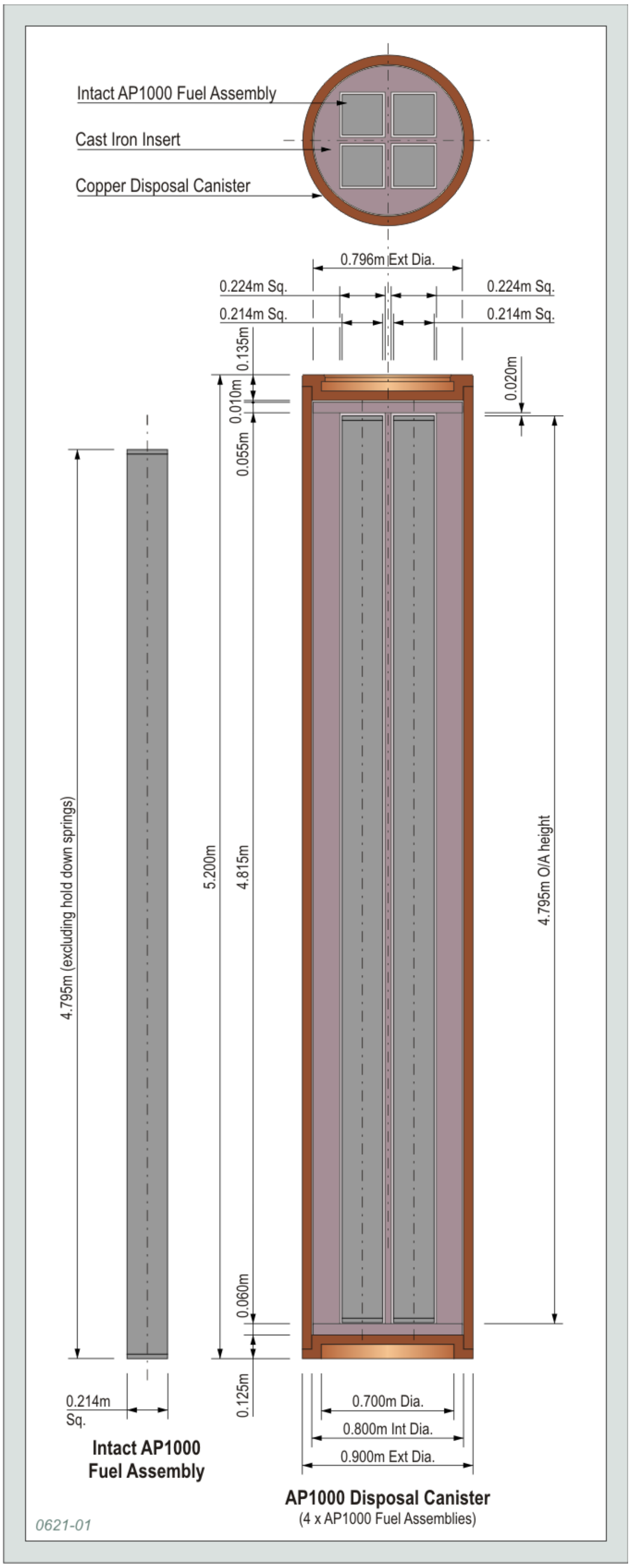

Figure 11 Illustration of an AP1000 spent fuel disposal canister

3.4.3 Spent Fuel Package Numbers and Characteristics

Description of Packages

The GDA Disposability Assessment for the AP1000 assumes that 64 fuel assemblies will be generated every 18 months of reactor operation, which, for an assumption of 60 years operation, results in a total of 2,560 assemblies requiring disposal, i.e. 640 canisters.

The RCCAs described in Section 3.4.1 were not included in the initial disposal inventory supplied by Westinghouse. Although these wastes may have high specific activity, they will not be of large volume, and, therefore, are not expected to affect disposability of wastes from an AP1000. These components could be managed as either 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 4.795m (Figure 10), so the raw waste volume associated with 2,560 fuel assemblies is 562m^3. Regarding packaged volume, the envelope volume of a canister capable of accommodating four fuel assemblies is 3.33m^3, and the packaged volume of the waste consisting of 640 canisters is therefore 2,131m^3.

Westinghouse proposed that the concentration of chlorine impurities in the fuel was 5ppm for UO_2 and 20ppm for Zirlo. Both concentrations are considered to represent conservative assumptions.

The component mass estimates for an AP1000 fuel assembly are provided in Table 10. Table 11 presents the mass data for each fuel assembly and per disposal canister, summed for each material type.

Table 10 Estimates of component mass for an AP1000 fuel assembly

Component	Material	Mass per Assembly (kg)
UO_2	UO_2	6.125E+02
Cladding	Zirlo	1.246E+02
Zirlo Grids & Guide Tubes etc	Zirlo	2.930E+01
Inconel Grids	Inconel 718	2.070E+00
Nozzles Springs	Inconel 718	1.290E+00
Nozzles Steel	St Steel Type 304	1.457E+01
Lower Plenum standoff tube	Zirlo	2.310E+00
Plenum Springs	Inconel 718	1.820E+00
Alumina Insulating Pellets	Al_2O_3	5.650E-01
Total - whole fuel Assembly		**7.890E+02**

Table 11 Material mass breakdown for an AP1000 fuel assembly and for a copper canister (assuming four assemblies per canister)

Material	Mass per Assembly (kg)	Mass per Canister (kg)
UO_2	6.13E+02	2.450E+03
Zirlo	1.56E+02	6.25E+02
St Steel Type 304	1.5E+01	5.8E+01
Inconel 718	5E+00	2.1E+01
Al_2O_3	6E-01	2E+00
Total	**7.89E+02**	**3.156E+03**

Radionuclide Inventory

One-year-cooled radionuclide inventory data for AP1000 spent fuel irradiated to 65 GWd/tU were estimated by RWMD based on the following approach:

- the radionuclide and stable isotope content of the UO_2 was calculated using ORIGEN-S [24] based on cross-section libraries appropriate to a 17x17 Westinghouse fuel assembly. The calculations included the generation of the activation products C-14 and Cl-36 from the neutron activation of the O, N and Cl content of the UO_2;

- the radionuclide content of the Zirlo and Inconel cladding and assembly structural components were derived from an existing PWR fuel inventory spreadsheet, whose specific activity data had come from FISPACT-97 [25] calculations.

Following discussions between Westinghouse's fuel modelling specialists and RWMD, it was agreed that the 65 GWd/tU burn-up of an AP1000 should be accumulated in a sequence of five irradiation cycles each separated by a 17-day shutdown interval as specified in Table 12.

Table 12 Assumed irradiation cycle for estimation of AP1000 spent fuel inventory

Cycle Number	Duration (days)	Average Fuel Rating (MW/tU)
1	510	45.7314
2	510	44.3392
3	510	15.2020
4	510	14.1427
5	307	13.3492

Table 13 gives the starting composition of the UO_2 applied in the ORIGEN-S calculations. The nitrogen concentration in the UO_2 equates to 75ppm and is consistent with the maximum allowable concentration defined in Westinghouse's specification for UO_2 as described in its submission.

The composition of chlorine shown in Table 13 is consistent with a chlorine concentration in the UO_2 of 5ppm, which is consistent with the 95th percentile upper chlorine concentration derived for overseas LWR UO_2 as part of the Nirex Chlorine-36 project [26]. This level is somewhat below the 25ppm maximum allowable concentration defined in Westinghouse's specification for UO_2. It is noted that the Westinghouse specification limits for nitrogen and chlorine in UO_2 are the same as the ASTM Standard Specification for Sintered UO_2 [27]. It is clear that trace impurity levels, particularly of chlorine, are subject to considerable uncertainty and the actual levels found are likely to be dependent upon the supplier of the UO_2. Unpublished data on the chlorine concentration in the UO_2 used for Sizewell B fuel suggests that its upper bound concentration is likely to be much closer to 5ppm rather than 25ppm.

Table 13 Starting composition of UO_2 applied in ORIGEN-S calculations of the radionuclide inventory of AP1000 spent fuel

Isotope / Element	Atomic Wt	Number of atoms per tU	ORIGEN input Concentration (g/tU)
U234	234	1.363505E+23	**5.298E+01**
U235	235	1.140160E+26	**4.449E+04**
U238	238	2.417082E+27	**9.553E+05**
Oxygen	16.00	5.062469E+27	**1.345E+05**
Nitrogen	14.00	3.657490E+24	**8.503E+01**
Chlorine	35.45	9.636321E+22	**5.673E+00**

The inventory calculations for the cladding and structural components of the spent fuel considered the structural materials within the high-flux fuelled region of the assembly only. Upper bound elemental concentration data provided by Westinghouse were used where available. For elements that Westinghouse was unable to provide data for, upper bound concentrations that had been developed by RWMD for Zircaloy-4 and Inconel 718 and used in previous studies were applied. The combined Westinghouse / RWMD upper bound element concentration data to be applied is shown in Table 14 (Westinghouse data highlighted by a blue background).

Table 14 Upper bound elemental mass fractions used in estimation of AP1000 spent fuel cladding and structural material inventory (continued over two pages)

Element	Zirlo	Inconel 718
H	2.0000E-05	7.0000E-05
Li	1.6276E-06	1.9255E-06
Be	6.0000E-04	6.0000E-04
B	5.0000E-07	6.0000E-05
C_	1.2000E-04	4.5000E-04
N	5.0000E-05	1.0882E-03
O	1.3500E-03	1.5000E-03
F	6.0000E-05	1.0000E-03
Ne	1.2700E-05	1.2700E-05
Na	2.0000E-05	2.9922E-05
Mg	2.0000E-05	1.0000E-03
Al	7.5000E-05	6.0000E-03
Si	8.0000E-05	3.5000E-03
P	1.5000E-05	1.0000E-04
S	3.5005E-04	1.0000E-04
Cl	2.0000E-05	2.2618E-07
Ar	1.0000E-03	1.0000E-03
K	4.5000E-05	6.4264E-05
Ca	3.0000E-05	4.0235E-05
Sc	5.0000E-04	1.8030E-07
Ti	5.0000E-05	1.1500E-02
V	5.0000E-05	1.1041E-03
Cr	1.0000E-04	2.1000E-01
Mn	5.0000E-05	3.5000E-03
Fe	1.3000E-03	1.9300E-01
Co	5.0000E-05	1.0000E-02
Ni	7.0000E-05	5.5000E-01
Cu	5.0000E-05	1.5000E-03
Zn	1.5278E-05	1.4552E-03
Ga	1.0000E-03	4.2698E-04
Ge	7.0000E-04	7.0000E-04
As	5.0000E-04	7.7784E-04
Se	9.0000E-06	4.4607E-05
Br	1.6000E-04	5.3128E-06
Kr	3.3000E-06	3.3000E-06
Rb	4.5000E-05	1.2081E-05
Sr	1.0000E-03	1.0517E-05
Y_	1.0000E-03	8.0915E-06
Zr	9.7435E-01	1.5793E-05
Nb	1.2000E-02	5.2000E-02
Mo	5.0000E-05	3.3000E-02
Ru	1.0000E-07	1.0000E-07
Rh	4.8825E-07	5.6303E-06

Element	Zirlo	Inconel 718
Pd	1.0000E-06	1.0264E-06
Ag	1.0000E-05	1.5713E-05
Cd	5.0000E-07	4.9080E-06
In	1.7682E-06	6.6152E-07
Sn	1.1000E-02	1.4626E-04
Sb	1.0000E-04	8.2953E-05
Te	2.0000E-07	2.0000E-07
I	3.0000E-05	3.0000E-05
Xe	3.9500E-07	3.9500E-07
Cs	4.5000E-05	4.7994E-07
Ba	1.0000E-03	7.7470E-04
La	1.0000E-03	7.0228E-07
Ce	1.0000E-03	7.7447E-04
Pr	5.5000E-04	5.5000E-04
Nd	6.3636E-07	3.4035E-07
Sm	1.1818E-07	4.0174E-07
Eu	8.1882E-07	9.4265E-08
Gd	1.0909E-07	5.3379E-07
Tb	9.0000E-05	4.7200E-06
Dy	1.0000E-07	9.6995E-07
Ho	1.0000E-07	3.1099E-07
Er	1.0909E-07	6.1632E-07
Tm	1.0909E-07	2.0926E-06
Yb	2.7000E-04	2.8761E-06
Lu	1.0909E-07	1.3612E-06
Hf	1.0000E-04	4.9816E-06
Ta	2.0000E-04	5.0000E-04
W	1.0000E-04	4.6253E-04
Re	1.0909E-07	2.6465E-05
Os	1.0000E-07	1.7500E-07
Ir	1.3219E-06	2.2232E-06
Pt	5.0000E-07	5.0000E-07
Au	5.0000E-07	5.0000E-07
Hg	5.0000E-05	5.0000E-05
Tl	6.0000E-05	6.0000E-05
Pb	1.0000E-04	5.0000E-06
Bi	2.0000E-05	1.5641E-07
Th	2.6240E-06	6.0903E-06
U	0.0000E+00	0.0000E+00

Note: Westinghouse data are highlighted by a blue background.

To generate the overall AP1000 spent fuel inventory for input into DIQuest the one-year-cooled ORIGEN-S calculated inventory of the UO_2 was added to one-year-cooled cladding and fuel structural inventories coming from the PWR fuel inventory spreadsheet that had previously been calculated by FISPACT-97. The spreadsheet contained inventory data appropriate to a 61 GWd/tU irradiation. For application to this higher burn-up case all the radionuclide activities were increased by a factor 65/61. Such an adjustment is valid for most long-lived activation products as they tend to be generated by a simple one step activation process.

This one-year-cooled combined UO_2 and structural material inventory data were imported into DIQuest so that various waste package and radionuclide related parameters, such as heat output could be derived. The DIQuest import used the radionuclide inventory (specific activities) for only the fuelled section of the assembly, because this provides the overwhelming majority of the activity. The fuelled section has a volume of 0.20 m^3 per fuel assembly, giving 0.78 m^3 per disposal canister (waste package) and hence a total raw waste volume for DIQuest purposes of 500 m^3.

Package data are summarised in Table 15. The information in **Error! Reference source not found.** 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 15 AP1000 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}
Spent Fuel	Disposal Canister	640	2131.00	1.06E+03	3.39E+03	1.05E+06	1.43E+03	1.18E-01	2.24E+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 Westinghouse is designing and planning for a 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 Westinghouse 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 AP1000 at a load factor of 93% would be 6.93E+04 GWd. The 2,560 AP1000 fuel assemblies would contain 1,383 tU. Therefore, assuming that 2,560 fuel assemblies are generated over the lifetime of a reactor implies that the average burn-up of the assemblies is 50.1 GWd/tU. In calculating the total spent fuel inventory for the post-closure performance assessments, it was assumed that all 2,560 spent fuel assemblies had been irradiated to 65 GWd/tU, rather than 50.1 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.

3.4.4 Comparison of AP1000 Spent Fuel with Sizewell B PWR Spent Fuel

Fuel used to generate heat in an AP1000 would be expected to experience higher burn-ups than existing commercial reactors in the UK, for example the PWR at Sizewell B. Higher burn-up results in efficiency savings for the operator. For a similar quantity of electricity produced an AP1000 would create a smaller volume of spent fuel.

For example, an AP1000 operating for 60 years at 1.117 GW(e) would produce 2,560 spent fuel assemblies, which is equivalent to 38.2 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 [28], 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 AP1000 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 AP1000 with radionuclide inventories for spent fuel from the 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[12]. The data for the Sizewell B PWR are derived from the Low Burn-up PWR data presented in [29], 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 [14].

The only comparison of AP1000 and Sizewell B spent fuel inventories that could readily be made involves AP1000'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, appropriate information was not available for such comparison at the time of this assessment. Since the burn-up assumed for AP1000 spent fuel is about twice that assumed for the Sizewell B spent fuel, for many radionuclides the ratio of AP1000 to Sizewell B fuel activities is about two, as shown in Table 16. Ratios a little below and above two reflect non-linearity 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, green and orange shadings have been used in Table 16 to identify the cause of the apparently anomalous activity ratios.

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[12] Ninety 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.

Table 16 Comparison of radionuclide activities for spent fuel from an AP1000 with spent fuel from Sizewell B

Nuclide	Sizewell B SF TBq per Canister	AP1000 SF TBq per Canister	Ratio AP1000 : SXB
C-14	6.45E-02	3.30E-01	5.1
Cl-36	8.31E-04	3.63E-03	4.4
Ni-59	9.08E-04	1.55E-01	171
Se-79	3.18E-02	1.08E-02	0.34
Sr-90	6.75E+02	1.16E+03	1.7
Tc-99	1.03E+00	1.92E+00	1.9
Sn-126	5.67E-02	8.79E-02	1.5
I-129	2.39E-03	4.33E-03	1.8
Cs-135	3.02E-02	8.08E-02	2.7
Cs-137	1.02E+03	1.98E+03	1.9
U-233	1.23E-05	5.16E-05	4.2
U-234	1.33E-01	1.70E-01	1.3
U-235	1.53E-03	7.36E-04	0.48
U-236	2.15E-02	3.25E-02	1.5
U-238	2.46E-02	2.44E-02	1.0
Np-237	3.28E-02	6.50E-02	2.0
Pu-238	9.09E+01	4.16E+02	4.6
Pu-239	2.50E+01	3.03E+01	1.2
Pu-240	3.61E+01	6.39E+01	1.8
Pu-241	1.23E+02	2.03E+02	1.6
Pu-242	1.24E-01	4.34E-01	3.5
Am-241	2.83E+02	4.92E+02	1.7
Am-242m	7.32E-01	1.89E+00	2.6
Am-243	1.14E+00	7.25E+00	6.3

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, Westinghouse 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 AP1000 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 AP1000 inventories being about twice the Sizewell B inventories because of the two-fold higher irradiation). For example, for the calculations Westinghouse indicated that chlorine concentrations of approximately 5ppm and 20ppm for the UO_2 and Zirlo 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-4 were assessed to be approximately 5ppm and 1.7ppm respectively [30],[31]).

The large (factor of 171) activity ratio calculated for Ni-59 arises from the extra activity induced in the nickel rich Inconel 718 top and bottom grids of the AP1000 assembly. The calculations performed for the Sizewell B fuel did not include any Inconel fuel structural components.

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 AP1000 used a half-life for the radionuclide of about 2.95E+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 spent fuel calculations used to develop the estimates.

Blue cells: U-235. The lower activity of U-235 present in the AP1000 spent fuel is relatively straightforward to explain, it is merely a feature of the higher burn-up experienced by the AP1000 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.

Orange cell: U-233. When the typical mix of uranium isotopes in PWR fuel is irradiated, U-233 arises predominantly from the decay of Np-237 which has been produced by neutron capture in U-235 and U-238. The long half-life of Np-237 (~2E+6 yrs) means that on the timescale of 90 years cooling only a small fraction of the Np-237 inventory decays to U-233 so the inventory of U-233 is quite small. If Th-232 is present as an impurity in the fuel materials then U-233 may also arise directly by neutron capture in this thorium isotope. Because the rate of production of U-233 from Th-232 activation is relatively high, even trace amount of Th-232 in the fuel materials can lead to a substantial increase in the arisings of U-233. In the case of the AP1000 spent fuel calculations the Zirlo and Inconel 718 cladding and fuel structural materials were assumed to contain 2.6ppm and 6.1ppm thorium respectively. However, the Sizewell B fuel inventory calculations did not consider the presence of thorium impurities. The combination of the higher fuel burn-up and the extra U-233 production from Th-232 impurities explain why the U-233 inventory for the AP1000 spent fuel is about four times larger than that for the Sizewell B spent fuel.

Given the pessimisms associated with the per canister inventories, it can be concluded that the radionuclide characteristics of spent fuel from an AP1000 are consistent with those from Sizewell B PWR.

4 ASSESSMENT OF AP1000 OPERATIONAL AND DECOMMISSIONING ILW

In this section the assessment of Westinghouse's packaging proposals for ILW is discussed 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 Westinghouse, 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 Westinghouse'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 Westinghouse'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 AP1000 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 and would be expected to be addressed by future operators in a Letter of Compliance assessment process.

4.1 Waste Package Properties

4.1.1 Waste Container

Context

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.

The waste packages proposed by Westinghouse for their operational and decommissioning wastes are NDA standard 3m^3 Drums and 3m^3 Boxes.

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 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 Westinghouse submission. The results of the evaluation are provided 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.

Results and Implications

The $3m^3$ Box and $3m^3$ Drum packaging options proposed by Westinghouse are standard containers. The case for compliance with GWPS waste container criteria should be readily made and is 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 Check-list criteria for the different waste containers proposed by Westinghouse for the packaging of ILW

Waste Container	$3m^3$ Drum (Operational Waste)	$3m^3$ Box (Operational Waste)	$3m^3$ Box (Decommissioning Waste)
Dimensions within a defined envelope	Yes – as defined in WPS 320	Yes – as defined in WPS 310 or WPS 315	Yes – as defined in WPS 310 or WPS 315
Standardised lifting features	Twistlock fittings on top face of container	Twistlock fittings on top face of container	Twistlock fittings on top face of container
Gross mass [package specific mass limit]	4.36 tonnes [package specific mass limit of 8 tonnes]	4.56 tonnes [package specific mass limit of 12 tonnes]	10.72 tonnes [package specific mass limit of 12 tonnes]
Defined identifier format and location	Alpha-numeric identifier in machine readable format in four positions on drum body	Alpha-numeric identifier in machine readable format in four positions on box body	Alpha-numeric identifier in machine readable format in four positions on box body
Physical containment provided by container body, lid and sealing system	Stainless steel containment system with bolted lid incorporating either an elastomer seal or a labyrinth lid to prevent the loss of particulate material.	Stainless steel containment system with bolted lid incorporating either an elastomer seal or a labyrinth lid to prevent the loss of particulate material.	Stainless steel containment system with bolted lid incorporating either an elastomer seal or a labyrinth lid to prevent the loss of particulate material.
Standardised stacking characteristics	Designed for 7 high stacking with similar packages, each at 8 tonne gross mass (48 tonne compressive stack load)	Integral stacking posts, designed for 7 high stacking with similar packages, each at 12 tonnes gross mass (72 tonne compressive load)	Integral stacking posts, designed for 7 high stacking with similar packages, each at 12 tonnes gross mass (72 tonne compressive load)
Filtered venting where necessary	Filtered vent	Filtered vent	Filtered vent

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 waste package specification [7];

- *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 proposals for packaging of ILW include outline descriptions of the means of conditioning and immobilising activity associated with the waste. Detailed descriptions and supporting evidence as to the properties of the proposed wasteforms have not been presented by Westinghouse, 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 proposed use of cement grout for waste conditioning conforms to existing practices for similar wastes in the UK and would be expected to produce wasteforms that could meet existing RWMD specifications. The proposal to use RWMD standard waste containers is also likely to enable compliance with the existing standards and specifications. However, Westinghouse did not identify candidate grouts. Details of specific grouts, their properties and formulation development will be required in future LoC submissions.

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 Table 18 and Table 19. The key points are summarised below.

Operational Waste

The operational ILW wasteforms (Table 18) exhibit characteristics very similar to other operational waste steams that are already covered by Letters of Compliance. In principle, production of wasteforms with the necessary integrity should be readily achievable. It is expected that the issues below would be addressed by the operator as part of future submissions for operational ILW under the LoC process. This should include the development of understanding of wasteform evolution and performance under normal and accident conditions.

Table 18 Wasteform characteristics: Operational ILW

Waste stream	Filters	Primary resins[1,2]	Mixed resins[3]
Conditioning proposal	Conditioned with cement grout in $3m^3$ Box.	Conditioned with cement grout in $3m^3$ Drum.	Conditioned with cement grout in $3m^3$ Drum.
Physical immobilisation	Waste is infiltrated with cement grout to form a solid product. Infiltration of filter elements not known but, based on experience, appropriate immobilisation is likely to be achievable. It will be necessary to demonstrate that free liquids will not be present in the filters and that grout infiltrates the filters and immobilises particulates successfully and minimises voidage.	Waste is intimately mixed with cement grout to form a solid product. Based on experience this is likely to be acceptable. Measures to avoid segregation of lower density materials, including activated carbon, from grout may need to be considered. Capping grout likely to be required.	Waste is intimately mixed with cement grout to form a solid product. Based on experience this is likely to be acceptable. Measures to avoid segregation of lower density materials, including activated carbon, from grout may need to be considered. Capping grout likely to be required.
Mechanical/ physical properties	No data have been provided. Intimate grouting of filters closely resembles wasteforms that have been considered previously and experience suggests that satisfactory mechanical and physical properties can be achieved.	No data have been provided but mechanical and physical properties are likely to be acceptable. Presence of borate can retard setting of cements and will require appropriate grout formulation development. Alternative approaches could be considered.	No data have been provided but mechanical and physical properties are likely to be acceptable. Presence of borate can retard setting of cements and will require appropriate grout formulation development. Alternative approaches could be considered.
Chemical containment	This wasteform closely resembles wasteforms that have been considered previously. Experience suggests that this waste is unlikely to affect chemical containment and is likely to acceptable.	Long-term degradation of organic materials and effect on pH buffering within backfilled vaults has some associated uncertainties. However, presence of the immobilisation grout will provide mitigation for such effects. Although the degradation products of ion-exchange resins may include species that could complex radionuclides, current experience suggests that such effects not to	Long-term degradation of organic materials and effect on pH buffering within backfilled vaults has some associated uncertainties. However, presence of the immobilisation grout will provide mitigation fo such effects. Although the degradation products of ion-exchange resins may include species that could complex radionuclides, current experience suggests that such effects not to

Waste stream	Filters	Primary resins[1,2]	Mixed resins[3]
		be significant although the nature of any degradation products should be investigated. It will be necessary to confirm that any oil associated with the activated carbon is retained within the wasteform. On balance, experience suggests that the wasteform is likely to be acceptable.	be significant although the nature of any degradation products should be investigated. It will be necessary to confirm that any oil associated with the activated carbon is retained within the wasteform. On balance, experience suggests that the wasteform is likely to be acceptable.
Hazardous materials	No data provided. However, no hazardous materials, with the exception of some common chemo-toxic elements, are likely to be present in the waste.	No data provided. However, no hazardous materials, with the exception of some common chemo-toxic elements, are likely to be present in the waste.	No data provided. However, no hazardous materials, with the exception of some common chemo-toxic elements, are likely to be present in the waste.
Gas generation	No data provided. Bulk gas generation not likely to be a significant issue for these wastes and the use of a vented container and typical permeability grouts are likely to be acceptable. Radiolysis of porewater would be expected to be the major source of gas and estimates would be expected as part of future LoC submissions.	No data provided. Bulk gas generation not likely to be a significant issue for these wastes and the use of a vented container and typical permeability grouts are likely to be acceptable. Radiolysis of porewater would be expected to be the major source of bulk gas and estimates would be expected as part of future LoC submissions. Volatile amines may be released from degradation of anion-exchange resins which may be of concern during transport and operations	No data provided. Bulk gas generation not likely to be a significant issue for these wastes and use of a vented container and typical permeability grouts are likely to be acceptable. Radiolysis of porewater would be expected to be the major source of bulk gas and estimates would be expected as part of future LoC submissions. Volatile amines may be released from degradation of anion-exchange resins which may be of concern during transport and operations.
Wasteform evolution	No issues are expected for filters infiltrated by a cement grout. Intimate grouting of filters closely resembles wasteforms that have been considered previously and experience suggests that satisfactory wasteform evolution	Long-term evolution and impact on wasteform performance requires further consideration but it is expected to be acceptable. The degradation of some cation ion-exchange resins has the potential to release sulphate	Long-term evolution and impact on wasteform performance requires further consideration but it is expected to be acceptable. The degradation of some cation ion-exchange resins has the potential to release sulphate

Waste stream	Filters	Primary resins[1,2]	Mixed resins[3]
	would be expected.	which can affect evolution of the wasteform.	which can affect evolution of the wasteform.

Notes
1 This stream also includes inorganic ion-exchange materials.
2 This stream will also include spent activated carbon.
3 The mixed resin stream comprises the primary resins waste stream plus condensate polisher spent resins and any steam generator blow down material.

Decommissioning ILW

The decommissioning ILW wasteforms (Table 19) exhibit characteristics very similar to other decommissioning waste steams 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 and current expectations that these will be low within a grouted wasteform. The significant C-14 content in the ILW steel may have an impact on risk from the release of gas after facility closure.

Table 19 Wasteform characteristics: Decommissioning ILW

Waste stream	ILW Steel[1]	Pressure vessel ILW[2]
Conditioning proposal	Conditioned with cement grout in $3m^3$ Box.	Conditioned with cement grout in $3m^3$ Box.
Physical immobilisation	Waste is infiltrated with cement grout to form a solid product. Waste is immobilised.	Waste is infiltrated with cement grout to form a solid product. Waste is immobilised.
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 decommissioning wastes are unlikely to contain materials that can affect chemical containment adversely.	Steel decommissioning wastes are unlikely to contain materials that can affect chemical containment adversely.
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 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 but rates of radiolytic gas generation will be	No data provided. Low corrosion rates expected. Experience suggests that typical corrosion

Waste stream	ILW Steel[1]	Pressure vessel ILW[2]
	higher than for the reactor vessel steel. Experience suggests that typical bulk gas generation rates and the use of standard permeability grouts in a vented $3m^3$ Box will be acceptable. The significant C-14 content may have an impact on risk from the post-closure gas pathway.	rates and the use of standard permeability grouts in a vented $3m^3$ Box will be acceptable.
Wasteform evolution	No issues expected. Wasteform is expected to evolve in a slow and predictable manner. Confirmation of expected slow corrosion rates for these materials in cement grout will be required to confirm adequate long-term performance of wasteform with respect to expansive corrosion.	No issues expected. Wasteform is expected to evolve in a slow and predictable manner. Confirmation of expected slow corrosion rates for these materials in cement grout will be required to confirm adequate long-term performance of wasteform with respect to expansive corrosion.

Notes
1 Type 304 stainless steel.
2 Ferritic steel with small amount of stainless steel cladding.

4.1.3 Waste Package Performance

The Waste Package Performance assessments considered the performance of the proposed waste packages under accident conditions [34]. 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 a 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.

Using these methods, impact and fire accident release fractions were estimated for the waste packages proposed for the operational ILW ($3m^3$ Boxes and $3m^3$ Drums) and for the decommissioning ILW ($3m^3$ Boxes) [34].

In the following paragraphs each of the proposed waste package/ wasteform combinations are compared against the above criteria and results presented in Tables 20 to 23. It should be noted that the values for calculated dose given in Table 20 to 23 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

The filter and ion exchange resin operational wastes have been defined and will be directly immobilised in standard waste containers. The waste containers are well-known and there has been extensive testing of the mechanical and thermal performance of metal in grout and ion exchange resin 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 as described above using the descriptions supplied by Westinghouse and supplementing this where possible by the use of UK generic test and modelling results for generic waste packages. 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 Table 20 and Table 21. 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 20 Waste package impact performance: Operational ILW

Waste stream	Filters grouted in $3m^3$ Box	Ion exchange resins grouted in $3m^3$ Drum
Release of radionuclides and hazardous materials are low and predictable, exhibit progressive release behaviour and no cliff-edge effects	Yes, based on impact modelling and drop testing of a generic $3m^3$ Box waste package.	Yes, based on drop testing and impact modelling of a generic $3m^3$ Drum waste package.
Both barriers play effective role in minimising releases	Yes, based on modelling of a generic $3m^3$ Box waste package combined with break-up data for metal in grout.	Yes, based on modelling of a generic $3m^3$ Box waste package combined with break-up data for ion exchange resin in grout.
Capable of being dropped from 0.3m with no release	Yes, based on impact modelling of a generic $3m^3$ Box waste package.	Yes, based on impact modelling of a generic $3m^3$ Box waste package.
Activity release consistent with regulatory dose limits to workers and members of the public	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 WPS/710. Predicted maximum dose of 0.02 mSv (AP01).	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 WPS/720. Predicted maximum dose of 0.001 mSv (AP02) and 0.001 mSv (AP03).

Table 21 Waste package fire performance: Operational ILW

Waste stream	Filters grouted in $3m^3$ Box	Ion exchange resins grouted in $3m^3$ Drum
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 $3m^3$ Box waste package it is predicted that releases will be low and predictable except for I-129. (See note following this table)	Yes, based on thermal modelling of a generic $3m^3$ Drum waste package it is predicted that releases will be low and predictable except for I-129. (See note following this table)
Both barriers play effective role in minimising releases	Yes, based on modelling of a generic a generic $3m^3$ Box waste package combined with small-scale furnace test data on releases from metal in grout.	Yes, based on modelling of a generic $3m^3$ Drum waste package combined with small-scale furnace test data on releases from ion exchange resin in grout.
Activity release consistent with regulatory dose limits to workers and members of the public	Scoping calculations indicate higher than guidance values of Table 2 WPS/710. Predicted maximum dose of 16.4 mSv (AP01).	Scoping calculations indicate higher than guidance values of Table 2 WPS/720. Predicted maximum dose of 13.2 mSv (AP02) and 9.7 mSv (AP03).

Note: The high releases are due solely to the I-129 content in the wasteform. This radionuclide has a chemical form that is potentially very volatile (i.e. can form gaseous compounds). The above predicted releases (mSv) can be significantly reduced if the actual chemical form of I-129 is identified and applied to the release calculations with less volatile characteristics.

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 Westinghouse 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 were 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 Table 22 and Table 23. Specific modelling of the waste items within the container would be required to improve on these assumptions in support of future LoC submissions.

Table 22 Waste package impact performance: Decommissioning ILW

Waste stream	Activated steels 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 and drop testing of a generic $3m^3$ Box waste package.
Both barriers play effective role in minimising releases	Yes, based on modelling of a generic $3m^3$ Box waste package combined with break-up data for metal in grout.
Capable of being dropped from 0.3m with no release	Yes, based on impact modelling of a generic $3m^3$ Box waste package.
Activity release consistent with regulatory dose limits to workers and members of the public	Yes. Initial calculations based on faults in the UILW vaults, consistent with guidance values of Table 2 WPS/710. Predicted maximum dose of 1.11 mSv (AP04) and <0.001 mSv (AP05).

Table 23 Waste package fire performance: Decommissioning ILW

Waste stream	Activated steels 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 $3m^3$ Box waste package it is predicted that releases will be low and predictable except for Cl-36, Se-79 and C-14. (See note following this table)
Both barriers play effective role in minimising releases	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 limits to workers and members of the public	Scoping calculations indicate higher than guidance values of Table 2 WPS/710. Predicted maximum dose of 30.01 mSv (AP04) and <0.001 mSv (AP05).

Note: The waste 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 AP04 have chemical forms that are potentially very volatile (i.e. can form gaseous compounds): Cl-36, Se-79 and C-14. The above predicted releases (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 Design Impact

Context

The GDA Disposability Assessment for the AP1000 has considered implications for GDF design of disposing of ILW from an AP1000, and the scale of the impact of the additional ILW from operation and decommissioning of an AP1000 on the projection of the GDF area on the land surface (the "footprint"). This analysis is based on the ILW GDF design presented by RWMD in [35]. 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 [36]. 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 [37] assumed that operational and decommissioning ILW would be emplaced in unshielded ILW (UILW) vaults. The $3m^3$ Boxes and $3m^3$ Drums that are proposed for packaging of AP1000 operational and decommissioning ILW are UK standard packages, and, therefore, would not present any new issues for handling, stacking, lifting and identification.

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. In all cases the volumes of ILW generated by the operation of an AP1000 are small compared to the volume of legacy ILW. Operation of a single AP1000 would require an additional length of UILW vault of approximately 65 m (7 m for the $3m^3$ Boxes and 58 m for the $3m^3$ Drums) [37]. This represents approximately 1% of the area required for the legacy ILW, per AP1000 reactor, and less than 10% for the illustrative fleet of nine AP1000 reactors.

4.2.2 Transport System

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 [38] as incorporated into UK transport legislation. In support of this work RWMD has produced a Generic Transport Safety Assessment (GTSA) [39] 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 required 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 AP1000 have been tested following the above approach. The transport safety assessment has addressed [40]:

- transport of AP01, AP04 and AP05 ILW in $3m^3$ Boxes as Type B Packages;
- transport of AP02 and AP03 ILW in $3m^3$ Drums 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 [40]. 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. Selected waste packages were [40]:

- $3m^3$ Box: AP01 Primary Circuit Filters;
- $3m^3$ Drum: AP02 Organic Primary Resins;
- $3m^3$ Box: AP04 ILW Steel.

Results and Implications

A range of issues have been identified through the transport assessment [40] 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.

Operational ILW

The proposal to use RWMD standard waste containers for operational ILW ($3m^3$ Box and $3m^3$ Drum), and the requirement for such packages to be transported in a shielded transport overpack has been assessed to eliminate potential challenges to the dose-rate limits set out in the IAEA Transport Regulations.

The $3m^3$ Box and $3m^3$ Drum waste packages proposed by Westinghouse are expected to meet all requirements for safe transport as defined in IAEA transport regulations. No issues were identified for operational ILW. For the AP01 Primary Circuit Filters, the AP02 Organic Primary Resins and AP03 Organic Secondary Resins the grouted wasteforms and packaging provide adequate shielding and containment under all conditions assessed.

Decommissioning ILW

The proposed decommissioning ILW packages comprise metal items immobilised into standard containers using a cement grout. Transport of decommissioning steel is assumed to be undertaken 40 years after final reactor shutdown, which allows the radioactivity and heat output of the ILW to reduce below transport limits. The packaging proposals for decommissioning ILW 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 $3m^3$ Box packaging option for AP04 ILW Steel decommissioning waste 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. Transport of AP04 ILW Steel wastes packaged in a $3m^3$ Box assumes use of a SWTC-285 in order to meet dose rate limits for Type B transport packages. For the maximum inventory, external dose rates may exceed IAEA Transport Regulations limits at the calculated time of 40 years decay. As has been noted previously, a conservative approach has been adopted for the calculation of the inventory and therefore these dose rates may not occur in practice. If this is an issue then management arrangements applied to loading of the waste into the $3m^3$ Box, or further storage before transport, may be required.

For containment under normal conditions, the H-3, C-14 and Ar-39 contents of AP04 ILW Steel may lead to increases in excess of the 10^{-6} A_2/hr release criteria defined in the IAEA Transport Regulations [38]. Tritium is derived from beryllium, and crustal abundances had been assumed for concentrations of beryllium in precursor steel, which is highly-conservative. C-14 is an activation product and release of C-14 would require corrosion of the steel (a similar argument applies to Ar-39). Therefore, in future LoC interactions the operator will need to demonstrate that the grade of steel used in the reactor does not result in concentrations of H-3, C-14 and Ar-39 activation products in AP04 ILW Steel that threaten containment during transport.

All transport packages need to be accompanied by a Design Safety Report which is used to demonstrate compliance with IAEA Transport Regulations and to define acceptable contents, usually via a “contents specification”. Such a document has been produced for the SWTC [41], and future LoC interactions would be informed by the contents specification to demonstrate compliance with transport regulations.

Criticality Safety

IAEA regulations on the safe transport of radioactive waste [38] 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 0.03g in AP02. 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 [38]. 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 AP1000.

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 AP1000 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 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) [42]. 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 AP1000 GDA 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 the necessary operations to unload the SWTC are completed and the $3m^3$ Box or $3m^3$ Drum is transferred to the emplacement location in the disposal vault.

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 waste 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 AP1000 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 [43]. The estimated doses were then compared to targets for design basis fault sequence mitigated doses currently being considered by RWMD. These targets are reproduced in Table 24.

Table 24 Targets for design fault sequence mitigated doses used in the AP1000 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 ROSA toolkit assessments are summarised here in terms of the waste type, based on the discussion in the AP1000 Operational Safety assessment [43]:

- operational ILW (representative streams AP01 and AP02):
  - Both on-site (worker) and off-site (public) protected doses for impact accidents were below the BSO of 0.1 mSv and 0.01 mSv respectively for the Primary Circuit Filters and for the Organic Primary Resins. The operational ILW packages were therefore judged acceptable with regard to impact faults.
  - For fire accidents, on-site (worker) protected doses are below the BSO. Application of the ROSA Toolkit results in high off-site (public) protected doses for the AP02 Organic Primary Resins, with doses dominated by I-129. The activity of I-129 in the resins is the result of applying a scaling factor of 4.1×10^{-5} to the assumed activity of Cs-137 in the waste. The scaling factor is the geometric mean of published values but includes large numbers that represent extreme cases, and reduction of the scaling factor would lead to the calculation of acceptable doses. In future LoC interactions, the operator will need to provide further information and justification for the scaling factors used to derive I-129 inventories.
- decommissioning ILW (representative stream AP04):
  - (Off-site public) protected doses are generally below the most stringent BSL with the exception of those for faults involving thermal challenges to the AP04 ($3m^3$ Box) packages which gave public protected doses above the 1 mSv BSL with a maximum predicted dose of 7.5 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, 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.

- (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.

Operational safety assessment for ILW from an AP1000 did not identify any issues that challenge the disposability of these wastes [43]. Both worker and public mitigated doses for operational ILW and decommissioning ILW packages are below the required standards indicating acceptable performance. 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 a more detailed estimate of the radionuclide inventory, especially for AP02 Organic Primary Resins.

Operational Safety under Normal Conditions

IAEA Regulations for the safe transport of radioactive materials [38] 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 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.

Handling and emplacement of the unshielded packages containing operational ILW or decommissioning ILW, provided they are transported in SWTCs with 285 mm of shielding (70 mm shielding for AP05), is unlikely to contribute significantly to operational doses, owing to the remote handling philosophy adopted within the GDF design for such packages.

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 [43]. This reflects the nature of the wastes and the small quantities of potentially gaseous radionuclides in the assessment inventories.

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 [44]. 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 AP1000 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 protected accidental 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 AP1000 through the LoC assessment process.

4.2.4 Environmental Issues

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 AP1000 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 AP1000 ILW.

It is noted that Westinghouse proposes a forty-year deferral period before starting final decommissioning. 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 Assessment

Context

The Security assessment 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;
- 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 AP1000 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 AP1000 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 AP1000 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 Nuclear Material 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 AP1000 would require physical protection to no higher than Category IV standards [45] 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 AP1000 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 gradually 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) [46]. 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 AP1000 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 also considered impacts due to chemotoxic species contained in the ILW from a single AP1000. These issues have been considered by comparison of the wastes for each AP1000 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 nine AP1000 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 AP1000 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 AP1000 operational ILW is 2,727 m^3 [47], which is small compared to the total conditioned waste volume of 168,000 m^3 assessed in the GPA. Similarly, the number of packages, 1,235 [47], is also small compared to the 285,000 packages assessed in the GPA. For a fleet of nine AP1000s, the conditioned waste volumes and package numbers are approximately 15% and 4% of the quantities assessed in the GPA respectively. Furthermore, for a fleet of nine AP1000s, the contribution to the total inventory of each radionuclide assessed in the GPA is less than 0.1% for all radionuclides except I-129 and Be-10. The combined operational inventory of I-129 is 5% of the legacy waste inventory. As described in Section 3.3.3, the quantities of I-129 in the resins used conservative scaling factors. However, even the conservative values used in the GDA Disposability Assessment for the AP1000, do not result in risks above the risk guidance level.

Recognising the requirements to refine inventory data and confirm the viability of packaging proposals identified previously, the additional calculated risk for the disposal of ILW from a single AP1000 in a site of the type described would be consistent with meeting regulatory targets. The consideration of a fleet of nine reactors would not alter this conclusion.

The post-closure safety assessment [47] has identified one issue regarding the organic content of the operational waste which would need to be considered in a future Letter of Compliance interaction with the operators.

Assessment of the characteristics of the operational ILW waste packages noted that the quantity of organic material in the resin wastes from a fleet of nine AP1000 reactors was approximately 40% of the quantity in legacy wastes. Furthermore, the Westinghouse packaging proposals result in a higher organic loading per waste package for resin wastes than the GPA average (~700 kg/m^3 compared to an average of ~50 kg/m^3 in legacy wastes). The total quantity of organic material in the AP1000 intermediate level waste is not a major issue, however, the concentration in waste packages is at a high level (factor of 14 greater than that of an average waste package). Were this organic material to be released and interact with neighbouring packages, it could have deleterious effects on the solubility and sorption of key radionuclides such as uranium-238 which may be present in neighbouring waste packages. This issue can be addressed through greater knowledge of the form of the organic material and through consideration as part of deliberations of the GDF waste emplacement strategy. This issue would require further consideration by operators and RWMD under a future Letter of Compliance interaction.

Decommissioning ILW

As with operational ILW, the conditioned volume of decommissioning ILW, at 172 m^3, 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 AP04 and AP05 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, AP04 (ILW Steel), which is stainless steel waste associated with the reactor pressure vessel internals, has high specific activity for a range of radionuclides, in particular C-14. This waste stream contains 199TBq of carbon-14. The acceptable release rate, via the gas pathway, calculated by RWMD for carbon-14 for GDF conditions is 0.02 TBq/y. Further consideration/examination should be given to the assumed inventory of carbon-14 and its release rate from the steel matrix. In particular, it will be necessary to determine the fraction of the carbon-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 carbon -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 AP04 ILW Steel is 2 Wm^{-3}. Although this is higher than the heat output of an average 3m^3 Box for legacy waste, it is significantly less than the Waste Package Specification limit for a 3m^3 Box waste package of 200 W.

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 AP1000 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 contains more than 5g of fissile material, no decommissioning ILW stream contains more than 28g of fissile material, and there is a 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 substantially less than a minimum critical mass.

Summary

The operational and decommissioning ILW from an AP1000 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 C-14 content of steels and its impact on risk from the gas pathway, organic content of operational waste streams and impact on the long-term safety case.

4.4 Summary of the Disposability of AP1000 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, proposals for the packaging of operational ILW and 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 AP1000. 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 AP1000 and of existing PWRs, and the expectation that similar operating regimes would be applied.

For operational ILW, the conditioned waste volume (2,727 m^3) and number of packages (1,235) 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 0.1% for all radionuclides except I-129 and Be-10. The combined operational waste inventory of I-129 is calculated at 5% of the inventory for legacy wastes. This is a conservative estimate and can be reduced with improved data on the frequency of fuel cladding failure.

For decommissioning ILW, the total activities of the six radionuclides with the highest activities in AP1000 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 8). 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 1000ppm. 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 proposed operational ILW packages use standard RWMD waste containers which would provide compliance with many aspects of the existing standards and specifications.

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 AP1000 operational and decommissioning ILW on the size of a Geological Disposal Facility has been assessed. It has been concluded that the necessary increase in the 'footprint area' is small, corresponding to approximately 65m of vault length for each AP1000. This represents approximately 1% of the area required for the legacy ILW, per AP1000 reactor, and less than 10% for the illustrative fleet of nine AP1000 reactors. This is in line with previous estimates for potential new build reactor designs.

4.4.5 Transport Safety

The proposal to use RWMD standard waste containers for operational ILW ($3m^3$ Box and $3m^3$ Drum), and the requirement for such packages to be transported in a shielded transport overpack has been assessed to eliminate potential challenges to the dose-rate limits set out in the IAEA Transport Regulations.

The transport safety assessment has identified that carbon-14 has the potential to challenge safety limits for decommissioning ILW. 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.

4.4.6 Operational Safety

The operational safety assessment for ILW from an AP1000 did not identify any issues that challenge the disposability of these wastes. In some cases, doses estimated for operational ILW and 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, 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 AP1000 has been recognised.

4.4.8 Security and Safeguards

The ILW to be disposed of from operation of an AP1000 present no security or safeguards issues of significance.

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 AP1000 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.

The organic loading of the operational waste is a factor of 14 greater than that of an average waste package. This organic material could interact with neighbouring packages and result in deleterious effects on the solubility and sorption of key radionuclides such as uranium-238 which may be present in neighbouring waste packages. This issue would require further consideration by operators and RWMD under a future Letter of Compliance interaction.

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 AP1000 in a site of the type described would be consistent with meeting the regulatory guidance level. The consideration of a fleet of nine reactors would not alter this conclusion.

5 ASSESSMENT OF AP1000 SPENT FUEL

In this section, we discuss the assessment of Westinghouse's 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 AP1000.

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 and would be expected to be addressed by future operators in a Letter of Compliance assessment process.

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. Initially, fuel is cooled in a water-filled pool whilst the short-lived radioactivity decays. In many PWR power plants around the world, fuel is later transferred to dry storage which may be vault-storage or cask-storage, for the remainder of the interim storage period.

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[13]. 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 insert, 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[14] 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 fuel assembly burn-up of 50 GWd/tU.

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[13] 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.

[14] In an AP1000, fuel burn-up is accumulated in an irradiation "history" assumed to consist of four irradiation cycles of 510 days duration and a fifth cycle of 307 days duration, with 17 days with the reactor shutdown for maintenance purposes.

Thermal Modelling

Four different calculations have been performed; two for the 65 GWd/tU 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 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 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 12 shows the temperature transient for a disposal canister containing four 100-year cooled maximum burn-up assemblies (in Figure 12, 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 13 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 12 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 reference assumption for this stage of assessment.

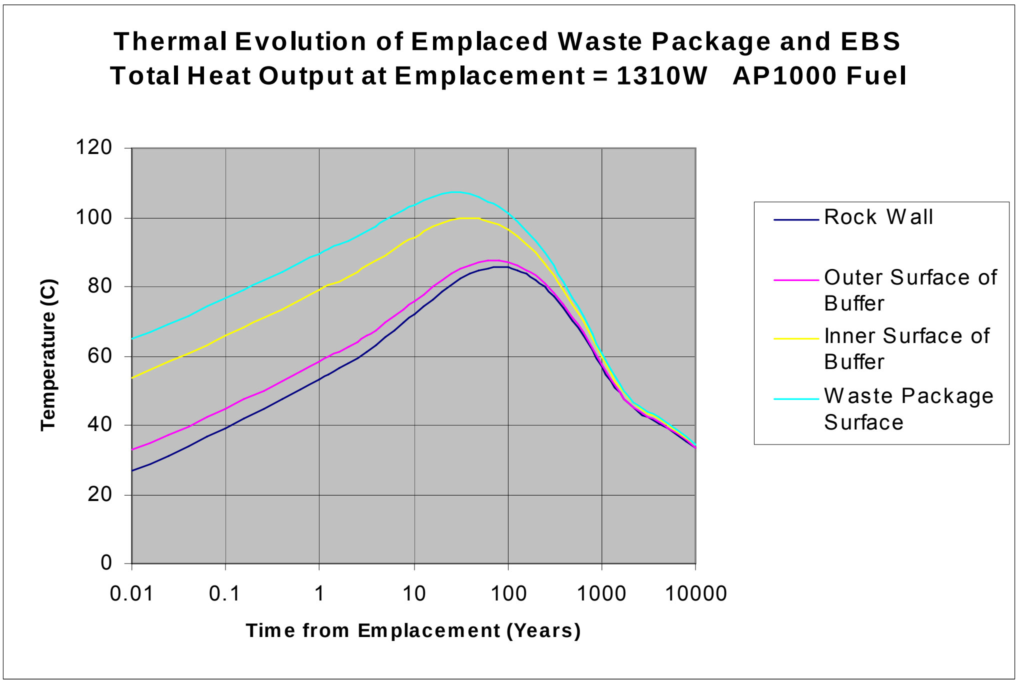

Figure 12 Near-field temperature history for 65 GWd/tU base case following emplacement of spent fuel waste packages; buffer inner surface refers to the part of bentonite adjacent to the disposal canister, and buffer outer surface refers to the part of bentonite adjacent to the near-field rock

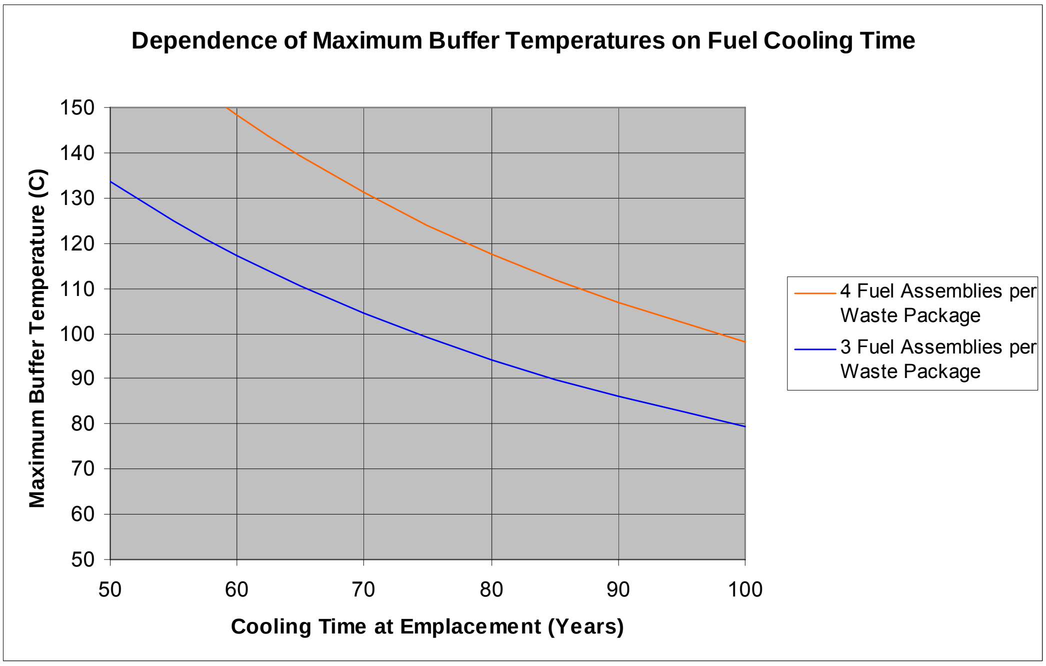

Figure 13 Interim storage cooling times for 65 GWd/tU base case required to attain required temperatures in the inner surface of bentonite buffer

5.2 Spent Fuel Disposal Package Properties

5.2.1 Wasteform

Context

The provision of a hermetically sealed, durable copper or steel waste container 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 AP1000 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 Zirlo 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 AP1000 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 important 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 taken 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 [8] 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 qualitative 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 5 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 waste 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 insert and, during transport, the shielded transport container. Evaluation of disposal package performance under such accident conditions has been undertaken by RWMD [34] 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 [34]. 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 AP1000 spent fuel on the design of the disposal facility [37].

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 AP1000 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 considered the impact on the GDF of a single AP1000 based on the assumption that the spent fuel is packaged prior to consignment. The impact of a fleet of nine AP1000s has also been considered [37].

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 two 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 [62]. As the MRWS process progresses, RWMD will review the design based on information relevant to a specific site and specific setting.

Results and Implications

For a packaging assumption of four fuel assemblies per canister, the 2,560 fuel assemblies would require 640 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 AP1000 [37]. Other design impacts associated with this change include [37]:

- 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, 15 disposal tunnels would be required for disposal of the 640 spent fuel disposal canisters from an AP1000. The area required for 15 disposal tunnels is approximately 0.11 km^2 [37]. The disposal tunnels required for emplacement of spent fuel from operation of a fleet of nine AP1000s would require 1 km^2 (the overall GDF footprint would in fact be slightly greater, due to a need for extra underground supporting infrastructure). This represents approximately 6% of the area required for the legacy HLW and spent fuel per reactor, and approximately 55% for the illustrative fleet of nine AP1000 reactors.

The Westinghouse proposals did not include any information regarding waste package identification markings [37]. 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 AP1000 that commenced operation around 2020, disposal of spent fuel from the reactor would commence in approximately 2120. Current plans for operation of the GDF anticipate completion of the disposal of legacy wastes in 2128. Therefore, interim storage of the first AP1000 spent fuel for ~100 years is consistent with current expectations regarding GDF operation. Although the early fuel discharge from an AP1000 could be disposed within the period of GDF operation for legacy spent fuel, the majority of spent fuel waste from operation of an AP1000 would arise after year 2128, which is the currently assumed closure date for the GDF. For example, the first reactor to commence operation in 2020 would not discharge its final spent fuel until 2080, and allowing for 100 years interim storage would mean this last fuel would not be available for disposal until 2180. This final disposal time could extend for a further 20 years under the assumption that it would take 20 years for a fleet of nine AP1000 to be constructed in the UK.

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 [38] 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 [63]. 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 AP1000 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 AP1000 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 [38]). Based on the data for AP1000 spent fuel (Section 3.4, Table 15), this limit would be challenged by the A_2 activity content of AP1000 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 [38]. 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 the 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 AP1000. Therefore, the design of the DCTC is expected to be sufficient to meet the requirements for containment under accident conditions. In any 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 waste 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 [64]. The mass of the DCTC loaded with a disposal canister containing four AP1000 spent fuel assemblies is estimated to be approximately 45t, which is compatible with existing design assumptions for transport by rail.

Transport Operational Risks

The additional transport movements associated with transport of AP1000 spent fuel to a GDF have been compared with the generic transport risk assessment [40], 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 [65]. 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 AP1000 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 [66].

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 AP1000 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 [67] 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 [68]. 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 AP1000 [43] 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 [68] 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 [69]. 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 24. 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 24.

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 AP1000 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 stages of operation the dose will be controlled by provision of shielding sufficient for the protection of workers to the requisite standard.

Criticality

The disposal packages containing AP1000 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 AP1000 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 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 quantity of spent fuel. 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 5.2, the disposal of spent fuel arising from a single AP1000 in a geological disposal facility will have an associated impact on the GDF footprint. It is estimated that an extra 400,000 m^3 of rock would be excavated if AP1000 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 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 AP1000 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 24.5kg plutonium (Table 15). Trace quantities of U-233 and thorium would also be present. AP1000 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 AP1000 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 the first 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 AP1000 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) [70], 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 safety 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 [71]. The post-closure safety assessment of AP1000 spent fuel was undertaken by considering whether the disposal of AP1000 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 AP1000 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 AP1000 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 [72] 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.

5.4.1 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 AP1000, 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 [73], which indicates an initial period of rapid aerobic corrosion, resulting in 11mm of penetration, followed by 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 [74].

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 [75]. Copper canisters with this performance have been adopted in the Canadian, Finnish and Swedish disposal programmes.

Results for Groundwater Pathway

Figure 14 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.

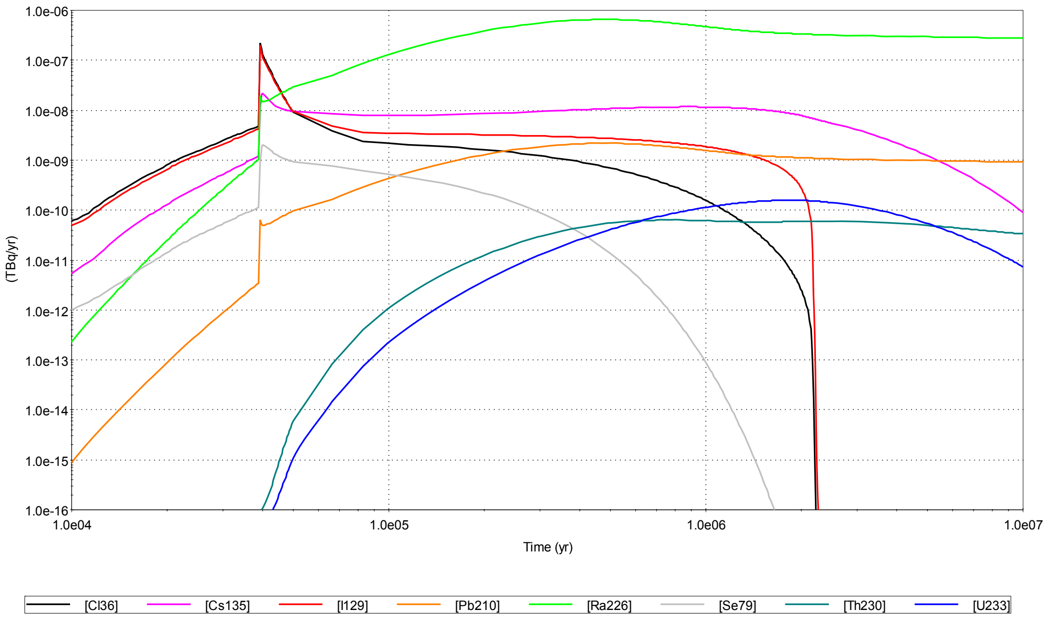

Figure 14 Near-field fluxes from a single steel overpacked disposal package containing four AP1000 spent fuel assemblies

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) is 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 14. 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 15. The peak risk for steel canister is calculated to be 5.6×10^{-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 3.2×10^{-7} per year for the lifetime arisings of a fleet of nine AP1000 reactors each generating a lifetime total of 640 canisters is calculated. This is below the risk guidance level [70]. Therefore, the post-closure assessment has not identified any post-closure safety issues for the groundwater pathway.

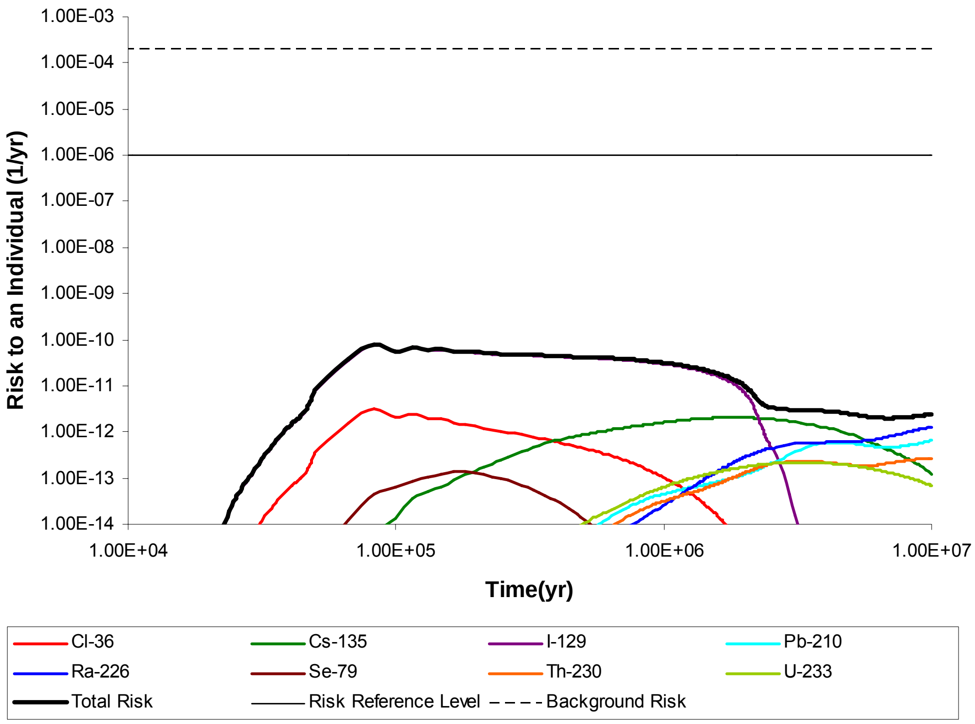

Figure 15 Total risk from a single AP1000 disposal package containing 4 spent fuel assemblies, assuming a steel canister

Results for the 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 5,760 canisters (i.e. equivalent to the spent fuel from nine AP1000s) is estimated to lead to the production of 403 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 [76]. 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 [62] 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.

Results for Criticality Safety

AP1000 spent fuel would contain about 22kg 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 [71]. Therefore, the quantity of fissile material in AP1000 spent fuel therefore lies between the quantity in UK AGR and Sizewell B PWR spent fuel. Reference [71] 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 adjacent canisters failing and, therefore, a low probability of fissile material from more than one canister accumulating together.

The potential for fissile material accumulation out of the canisters and post-closure criticality 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 [77]. 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 AP1000 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 AP1000 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 nine AP1000 reactors, encased 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 [78]. 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 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 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 AP1000 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 AP1000 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 Westinghouse for the GDA Disposability Assessment are highlighted in this section.

5.5.2 Inventory

The GDA Disposability Assessment for the AP1000 has shown that the principal radionuclides present in AP1000 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 AP1000 and existing PWRs, and the expectation that similar operating regimes would be applied.

Westinghouse has indicated that the GDA Disposability Assessment for the AP1000 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 AP1000 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

AP1000, 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 AP1000 operating for 60 years at 1.1 GW(e) would produce 2,560 spent fuel assemblies, which is equivalent to 38.8 SF 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 SF assemblies, 46.9 SF 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 AP1000, particularly in comparison with the spent fuel expected from Sizewell B.

For AP1000 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 AP1000 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 AP1000 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 AP1000 has assumed that spent fuel would be overpacked for disposal. Under this concept, spent fuel would be overpacked 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 AP1000 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 AP1000 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 AP1000 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 AP1000 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 AP1000 spent fuel on the size of the GDF has been assessed. The assumed operating scenario for an AP1000 (60 years operation) gives rise to an estimated 640 disposal canisters, requiring an area of approximately 0.11 km^2 for the associated disposal tunnels. A fleet of nine such reactors would require an area of approximately 1 km^2, together with associated service facilities. This represents approximately 6% of the area required for the legacy wastes HLW and spent fuel, per reactor, and approximately 55% for the illustrative fleet of nine 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 development of designs of suitable reusable shielded transport overpacks has commenced although is at an early stage of development. Consequently, although the AP1000 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 AP1000 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.

5.5.7 Environmental Considerations

No environmental issue that challenge the viability of the disposal of spent fuel from an AP1000 has been recognised.

5.5.8 Security and Safeguards

No security or safeguards issues were identified for AP1000 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 is 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 nine AP1000 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 risk 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 AP1000. 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 Westinghouse. 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 the Geological Disposal Facility, and the longer-term performance of the system have been considered, together with the implications for the size and design of the Geological Disposal Facility.

RWMD has concluded that sufficient information has been provided by Westinghouse 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 AP1000 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 AP1000, 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 AP1000 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[15]. 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[16].

In the case of higher activity waste coming forward from the AP1000 it is expected that the GDA Disposability Assessment commissioned by Westinghouse 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.

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[15] 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

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

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 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, numerous 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 numerous areas for future development is entirely consistent with expectations at this stage, due to the preliminary nature of the proposals for the packaging of waste and spent fuel considered in the GDA Disposability Assessment and the relatively high-level assessments performed.

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 AP1000 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.

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;
- provide information on procedures used to store waste prior to consignment to the GDF.

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 estimates for the quantity of organic material in the waste packages;

- provide information on the types of resins present in the wastes;

- provide information on the grade and composition of stainless steel used in an AP1000, taking account of the nitrogen impurities in the steel and provide information on the nature of tritium, C-14 and Ar-39 in activated metals;

- provide more detailed information on the chemistry of the wastes, including toxic element content;

- confirm that the contents of waste packages meet the “contents specifications”, for example that masses of both deuterium and beryllium in the waste packages are less than 1.8g and that the specific limitations on quantities of graphite, exotic fissile materials, moderating materials and favourable sites for sorption of fissile material will be met;

- 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 further information and justification for the scaling factors used to derive I-129 inventories;

- 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 Processes

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:

- demonstrate that grout used for conditioning of waste infiltrates the waste and immobilises particulates successfully, and that wastes are retained in the body of the wasteform, for example confirm that free liquids will not be present in the filters and demonstrate that grout infiltrates the filters, immobilises particulates successfully and minimises voidage;

- develop appropriate waste conditioning process envelopes, demonstrate that the plant operational envelope falls within this, and establish acceptable evolution and performance of the resulting wasteforms, for example develop an appropriate formulation envelope for Organic Primary and Secondary Resins that considers the presence of borate within the wastes;

- consider the use of alternative approaches to grouting waste, such as the use of organic polymers as an alternative to the use of cementitious grouts for conditioning;

- demonstrate that the packaging of AP04 steel ILW has appropriately considered the distribution of radioactivity associated with the waste, and that dose rates are not affected by placing steel close to the edge of the packages;

- provide data on the mass transport, thermal conductivity, and gas generation and pressurisation properties of the wasteforms.

B.1.4 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:

- 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.

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 Zirlo;

- 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.

7 REFERENCES

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 RWMD, *Generic Design Assessment, Summary of Disposability Assessment for Wastes and Spent Fuel arising from Operation of the Westinghouse Advanced Passive Pressurised Water Reactor (AP1000),* September 2009.

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

6 RWMD, *Disposability Assessment of Solid Waste Arisings from New Build,* April 2008.

7 United Kingdom Nirex Limited, *Generic Waste Package Specification*, Nirex Report No. N/104 (Issue 2), March 2007.

8 United Kingdom Nirex Limited, *Specification for Waste Packages Containing Vitrified High Level Waste and Spent Nuclear Fuel*, Nirex Report No. N/124, December 2005.

9 United Kingdom Nirex Limited, *Generic Disposal System Specification*, Nirex Report No. N/075, July 2003.

10 United Kingdom Nirex Limited, *Generic Repository Concept Specification for the Reference Concept for High Level Waste and Spent Fuel, Volume 1 - Specification*, Nirex Technical Note Ref: 491290, March 2007.

11 United Kingdom Nirex Limited, *Overview Report on Repository Concept and Programme Development for High Level Waste and Spent Nuclear Fuel in the United Kingdom*, Nirex Report No. N/125, September 2005.

12 SKB, *Long-term Safety for KBS-3 Repositories at Forsmark and Laxemar – A First Evaluation, Main Report of the SR-Can Project*, SKB Technical Report TR-06-09, October 2006.

13 Defra/NDA, *The 2007 UK Radioactive Waste Inventory*, Defra/RAS/08.001, NDA/RWMD/003, 2008.

14 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.

15 https://www.ukap1000application.com/PDFDocs/UK%20AP1000%20Safety,%20Security,%20and%20Environmental%20Report/Chapter%201/1-2_r1.pdf.

16 http://www.nrc.gov/reading-rm/basic-ref/students/animated-pwr.html, viewed on 23 June 2009.

17 NDA RWMD, *Nature and Quantity Assessment and Data Summary Sheets for Operational and Decommissioning Waste from a New Build AP1000 Reactor*, February 2009, NDA Document Ref: LL9287986.

18 Defra and Nuclear Decommissioning Authority, *The 2007 UK Radioactive Waste Inventory*, Report Dera/RAS/08.002 and NDA/RWMD/004, March 2008.

19 United Kingdom Nirex Limited. *The Identification of Radionuclides Relevant to Long-Term Waste Management in the United Kingdom*, Nirex Report No. N/105, November 2004.

20 NDA RWMD, *Scaling Factors for Radionuclides in PWR Operational Wastes*, NDA Technical Note (draft), Ref: LL8292332, October 2008.

21 IAEA, *Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: PWR Vessel Internals*, IAEA–Tecdoc-1119, October 1999.

22 J C Evans *et al.*, *Long-Lived Activation Products in Reactor Materials*, NUREG/CR-3474, 1984.

23 Nuclear Engineering International, *World Nuclear Industry Handbook*, ISBN 978-1-903077-67-2, 2009.

24 I C Gauld et al *ORIGEN-S: SCALE System Module To Calculate Fuel* Depletion*, Actinide Transmutation, Fission Product Buildup And Decay, And Associated Radiation Source Terms*, ORNL/TM-2005/39 Version 5.1, November 2006.

25 R A Forrest, J Ch Sublet, *FISPACT-97 User Manual*, UKAEA FUS 398, May 1997.

26 United Kingdom Nirex Limited, *The Measurement of Chlorine in Uranium and Uranium Dioxide*, Nirex Report NR774, March 1997.

27 *Standard Specification for Sintered Uranium Dioxide*, ASTM C776-94, October 1994.

28 United Kingdom Nirex Limited, *Outline Design for a Reference Repository Concept for UK High Level Waste/Spent Fuel*, Nirex Technical Note Ref: 502644, September 2005.

29 Electrowatt-Ekono (UK) Limited, *Radionuclide Content of a Range of Irradiated Fuels.* Contractor's Report to United Kingdom Nirex Limited, Nirex Technical Note Ref: 395337, September 2002.

30 J. Harrison, *The Chlorine-36 Content of Radioactive Wastes from Reactor Systems,* Nirex Document T/REP/20126/P/03, March 1997.

31 J. Harrison, *Measurement of the Chlorine Content of Non-Ferrous Materials*, Nirex Document T/REP/20118/P/04, March 1997.

32 RWMD, *Guidance on Environmental Conditions during Storage of Waste packages,* NDA Document WPS/630, June 2008.

33 NDA RWMD, *Wasteform Evaluation Pre-Conceptual LoC Submission for New Build AP1000 Reactor*, NDA Memorandum Ref: LL9682220, February 2009.

34 RWMD, *Impact and Fire Evaluations for New Build AP1000,* NDA document Ref LL9935811, April 2009.

35 United Kingdom Nirex Limited*, Generic Repository Design*, Nirex Report No. N/077, July 2003, updated 2005.

36 Defra, BERR, Welsh Assembly Government, Department of the Environment Northern Ireland, *Managing Radioactive Waste Safely: A Framework for*

Implementing Geological Disposal, Cm 7386. ISBN 978-0-10-173862-0. June 2008.

37 NDA RWMD, *Geological Disposal Facility – Design Impact Assessment for Generic Disposability Assessment Westinghouse Report,* NDA Technical Note Ref LL10022831, March 2009.

38 IAEA, *Regulations for the Safe Transport of Radioactive Material, 2005 Edition, Safety Requirements*, IAEA Safety Standards Series No. TS-R-1, August 2005.

39 United Kingdom Nirex Limited, *Generic Transport Safety Assessment*, Nirex Report No. N/078, July 2003.

40 Areva Risk Management Consulting, *Transport Safety Assessment of AP1000 New Build Reactors (GDA Disposability Assessment),* Areva Risk Management Consulting Report R09-061 (W), June 2009.

41 RMC, Contents Activity Limits of 3m^3 Box Transported in SWTC-285 Transport Container, (R04-169 (C) Issue A), October 2005.

42 United Kingdom Nirex Limited, *Generic Operational Safety Assessment*, Nirex Report No. N/079, July 2003.

43 Scott Wilson, *New Build AP1000 ILW and Spent Fuel Wastes,* Scott Wilson Operational Safety Assessment, June 2009, NDA Document Ref LL 10394611.

44 Galson Sciences Limited, *Revision of the General Criticality Safety Assessment*, GSL Report 0205-1, Version 1, Galson Sciences Limited, Oakham, March 2007.

45 Office for Civil Nuclear Security (Department of Trade and Industry), Information Concerning the Use, Storage And Transport Of Nuclear And Other Radioactive Material, CWP/G8, Revised: July 2006.

46 United Kingdom Nirex Limited, *Generic Post-closure Safety Assessment*, Nirex Report No. N/080, July 2003.

47 Serco, *Post-closure Assessment of AP1000 New Build Reactor Wastes,* Serco Report SERCO/TAS/003408/001, May 2009.

48 Defra, BERR, Welsh Assembly Government, Department of the Environment Northern Ireland, *Managing Radioactive Waste Safely: A Framework for Implementing Geological Disposal*, Cm 7386. ISBN 978-0-10-173862-0. June 2008.

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

50 Paxton, H.C. and Pruvost, N.L., *Critical Dimensions of Systems Containing* ^{235}U, ^{239}Pu *and* ^{233}U, LA-10860-MS, Revision, 1986.

51 JNC, H12: *Project to Establish the Scientific and Technical Basis for HLW Disposal in Japan, Second Progress Report on Research and Development for the Geological Disposal of HLW in Japan, Supporting Report 2: Repository Design and Engineering Technology*, JNC TN1410 2000-003, December 2000.

52 Wersin, P., Johnson, L.H. and McKinley, I. G., *Performance of the bentonite buffer at temperatures beyond 100°C: A Critical Review*, Physics and Chemistry of the Earth, Parts A/B/C, 32, 8-14, pp. 780-788, 2006.

53 Choi, H-J and Choi, J., *Double-layered buffer to enhance the thermal performance in a high-level radioactive waste disposal system*, Nuclear Engineering and Design 238, 2815-2820, 2008.

54 Kawamura, H., McKinley, I. and Neall, F., *Practical and safe implementation of disposal with prefabricated EBS modules*, International Technical Conference on Practical Aspects of Deep Radioactive Waste Disposal Session 2 - Paper No 8, 2008.

55 Nirex, *Preliminary Post-closure Assessment of a Reference Repository Concept for UK High-level Waste/Spent Fuel*. Technical Note, document number 503356, March 2006.

56 U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, Standard Review Plan for Spent Fuel Dry Storage Facilities Final Report, NUREG-1567, March 2000.

57 Grambow, B., Lemmens, K., Minet, Y., Poinssot, C., Spahiu, K., Bosbach, D., Cachoir, C., Casas, I., Clarens, F., Christiansen, B., de Pablo, J., Ferry, C., Giménez, J., Gin, S., Glatz, J.P., Gago, J.A., Gonzalez Robles, E., Hyatt, N.C., Iglesias, E., Kienzler, B., Luckscheiter, B., Martinez-Esparza, A., Metz, V., Ödegaard-Jensen, A., Ollila, K., Quiñones, J., Rey, A., Ribet, S., Rondinella, V.V., Skarnemark, G., Wegen, D., Serrano-Purroy, D. and Wiss, T., *RTD Component 1: Dissolution and release from the waste matrix: Final Synthesis Report*, NF-Pro Deliverable D1.6.3, May 2008.

58 Johnson, L., Poinssot, C., Ferry C., and Lovera, P., *Estimates of the Instant Release Fraction for UO2 and MOX Fuel at t=0. A Report of the Spent Fuel Stability (SFS) Project of the 5th Euratom Framework Program,* Nagra Technical Report 04-08, November 2004.

59 HSE, *Safety Assessment Principles for Nuclear Facilities*, 2006 Edition, Revision 1, January 2008.

60 Knuutila, A, *Waste Package Overpack Integrity Assessment in Handling Accidents*, September 2001.

61 Frazer Nash, *Impact Performance of HLW Waste Package, Preliminary Impact Modelling,* FNC 34118-13/35080R Issue 1, January 2009.

62 Defra, BERR, Welsh Assembly Government, Department of the Environment Northern Ireland, *Managing Radioactive Waste Safely: A Framework for Implementing Geological Disposal*, Cm 7386. ISBN 978-0-10-173862-0. June 2008.

63 Nuclear Technologies Plc, Performance of Gamma and Neutron Transport Calculations to Generate Dose Rates for Canisters of Vitrified HLW and Spent Fuel, NT/722500211/R402, January 2009.

64 RM Consultants, *Study of the Transport of UK High Level Waste and Spent Fuel,* R08-099(C), RWMD Reference LL/8387139, August 2008.

65 United Kingdom Nirex Limited, *Radiological Protection Policy Manual (RPPM)*, Reference NA/98/002, December 1998.

66 Agrenius, L., *Criticality safety calculations of storage canisters*, SKB Report TR-02-17.

67 United Kingdom Nirex Limited, *Operational Safety Assessment of a Reference Repository Concept for UK High Level Waste/Spent Nuclear Fuel*, Nirex Technical Note Ref: 481845, November 2005.

68 Scott Wilson, *Operational Safety Assessment for Vitrified High Level Waste (Preliminary),* Rev. V2, February 2009.

69 RWMD, *Nature and Quantity Assessment and Data Summary Sheets for Spent Fuel Waste from a New Build AP1000,* NDA Document Ref LL 9937805, April 2009.

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

71 United Kingdom Nirex Limited, *Preliminary Post-Closure Assessment of a Reference Repository Concept for UK High-Level Waste/Spent Fuel*, Nirex Technical Note Ref: 503356, March 2006.

72 Kossik, R. Miller, I. and Knopf, S., *GoldSim Graphical Simulation Environment: User's Guide*, GoldSim Consulting Group, February 2002.

73 Johnson, L. and King, F., *Canister options for the disposal of spent fuel,* Nagra Technical report 02-11, April 2003.

74 Oversby, V.M., *Uranium dioxide, SIMFUEL, and spent fuel dissolution rates a review of published data*, SKB Technical report TR-00-22, October 1999.

75 King, F., Ahonen, L., Taxén, C., Vuorinen, U. and Werme, L., *Copper Corrosion Under Expected Conditions in a Deep Geologic Repository*, SKB Technical Report TR-01-23, 2001.

76 United Kingdom Nirex Limited*, Generic Post-Closure Performance Assessment*, Nirex Report No. N/080, July 2003.

77 Behrenz, P. and Hannerz, K., *Criticality in a Spent Fuel Repository in Wet Crystalline Rock*, KBS Teknisk Rapport 108, Swedish Nuclear Fuel and Waste Management Company, Stockholm, Sweden, 1978.

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