Radioactive Waste Management

RWM TECHNICAL NOTE No. 23788023

ASSESSMENT REPORT

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

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This document has been produced by Radioactive Waste Management Limited to advise on waste transport and disposability issues in response to a waste conditioning proposal submitted by Hitachi-GE. The assessment basis adopted by Radioactive Waste Management Limited assumes waste packaging and on site storage for an interim period, transport to an off-site Geological Disposal Facility, emplacement of waste packages underground in a monitored and retrievable form followed by eventual sealing and closure of the facility.

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. Hitachi-GE Nuclear Energy (Hitachi-GE), a strategic global alliance between Hitachi Limited and General Electric (GE) founded in 2007, is proposing to develop and construct nuclear reactors in the UK based on the United Kingdom Advanced Boiling Water Reactor (UK ABWR) design, and is, therefore, requesting assessment of the design under the GDA process.

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 a "*requesting party 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 or spent fuel[3].

In accordance with regulatory guidance, Hitachi-GE has requested that Radioactive Waste Management Limited (RWM), a wholly-owned subsidiary of the NDA, provides advice on the disposability of the higher-activity wastes and spent fuel expected to arise from the operation of the UK ABWR. The assessment of the disposability of the higher-activity wastes and spent fuel from the UK ABWR is based on information on wastes and spent fuel, and proposals for waste packaging supplied by Hitachi-GE, supplemented as necessary by relevant information available to RWM.

This Assessment Report contains comprehensive details of the information supplied to RWM by Hitachi-GE, measures taken by RWM to supplement this information, assessment methods and the detailed conclusions of this GDA Disposability Assessment. 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 RWM within the GDA Disposability Assessment are also presented in a separate Summary Disposability Report[4].

The GDA Disposability Assessment process comprises three main components: a review to confirm the waste and spent fuel properties; an assessment of the compatibility of the proposed waste packages with concepts for geological disposal of higher-activity wastes and spent fuel; and 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 the development of reactor designs and operating regimes, all proposals are necessarily outline in nature. However, this Disposability Assessment has made assumptions to allow the production of a comprehensive and detailed data set describing the intermediate-level waste (ILW) and spent fuel to be generated from operation and decommissioning of a UK ABWR. At a later stage, more

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1 Meeting the Energy Challenge, A White Paper on Nuclear Power, CM 7296, 2008.

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

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

4 RWM, Generic Design Assessment: Disposability Assessment for the UK ABWR, Document Number LL/23383092, 2015.

specific and detailed proposals will be required for endorsing waste packaging proposals through the existing Disposability Assessment (formerly Letter of Compliance) process[5].

The number and type of new build reactors that may be constructed in the UK is currently not defined. The inventory for disposal is used by RWM to plan for the disposal of historical and currently arising wastes, and wastes from new nuclear build reactors[6]. This GDA Disposability Assessment has evaluated the implications of a single UK ABWR and, to illustrate the potential implications for geological disposal of constructing and operating a fleet of such reactors, consideration has also been given to the wastes from a fleet of four UK ABWRs operating for 60 years. This is consistent with current plans to construct two UK ABWRs at Wylfa and two at Oldbury. In order to assess the implications of operation of a fleet of four UK ABWRs, the wastes expected to arise from these reactors has been substituted for wastes in the inventory for disposal from other new nuclear build reactors with an equivalent generating capacity. This illustrative approach is considered to be a straightforward and pragmatic assumption for this assessment, and no inference should be drawn for wider UK planning purposes.

Nature of the ILW and Spent Fuel

Hitachi-GE has provided information on the ILW and spent fuel expected to arise from a UK ABWR operating for 60 years, with two different fuel assembly burn-up scenarios: 60 GWd/tU and 50 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 waste and spent fuel are identified:

- Operational ILW: ILW arising from the operation of a reactor.
- Decommissioning ILW: ILW arising from the eventual decommissioning of a reactor.
- Spent fuel: Used nuclear fuel arising from reactor operation.

Based on the stated assumptions, Hitachi-GE has provided information for the following eight types of operational ILW[7]:

- Condensate Filter Facility (CF) Crud.
- Low Conductivity Waste (LCW) Crud.
- Reactor Water Clean-up (CUW) Resin.
- Fuel Pool Cooling Clean-up (FPC) Resin.
- Post-operational Decontamination (DEC) Resin.
- Hafnium (Hf) Control Rods.
- Boron Carbide (B_4C) Control Rods.
- Mixed Metal ILW.

5 Nuclear Decommissioning Authority, *Guidance to the Disposability Assessment Process*, NDA Document WPS/650/03, 2014.

6 Department of Energy and Climate Change, *Implementing Geological Disposal, A Framework for the long-term management of higher activity radioactive waste*, URN 14D/235, 2014.

7 "Operational" ILW includes some of the ILW generated during post-operational decontamination operations, i.e. Post-operational Decontamination Resins, and small quantities of CUW and FPC resins. These wastes have been assessed alongside the operational ILW as they are of a similar chemical and physical form to the ion exchange resins generated during operations.

Hitachi-GE has indicated that the decommissioning ILW should be assumed to comprise the more highly activated steel components that make up two waste streams:

- Reactor pressure vessel internals.
- Reactor pressure vessel.

Information on decommissioning ILW has been assessed based on separation of the wastes into these two waste streams. Decommissioning ILW also comprises a small volume of stainless steel filter housings, which are assessed as additional material in the two decommissioning ILW streams. In practice, decommissioning wastes will comprise a mix of ILW and low-level waste (LLW), but, following discussion with Hitachi-GE, it has been agreed to make the conservative assumption that decommissioning wastes will be managed as ILW. 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.

The fuel used in a UK ABWR is expected to consist of ceramic UO_2 pellets enriched in U-235, encased in Zircaloy-2 cladding to form a fuel rod. It has been stated by Hitachi-GE that fresh natural uranium will be enriched to manufacture the fuel, i.e. recycled uranium is not assumed in the GDA application. The UK ABWR is expected to use the GE14 type of fuel assembly, which comprises a 10x10 array of fuel rods, consisting of 78 full-length fuel rods, 14 part-length rods which span roughly two-thirds of the active core, and two large central water rods occupying the remaining 8 pin locations, all housed in a zirconium alloy channel, which is specifically assumed as Zircaloy-2 in this assessment. As indicated above, information on spent fuel has been supplied by Hitachi-GE based on an assumed fuel assembly burn-up of 60 GWd/tU and 50 GWd/tU. The two radionuclide inventories developed from the information assume that all fuel assemblies would achieve this burn-up. Hitachi-GE has indicated that, in practice, 60 GWd/tU would represent the maximum of a range of burn-up values for individual fuel assemblies.

Proposals for Waste Packaging

Hitachi-GE has put forward proposals for the packaging of operational ILW based on the well-established current practice for similar wastes in the UK. The Disposability Assessment has assumed that crud and resin wastes would be grout cemented into $3m^3$ Drums using in-drum, lost-paddle mixing to ensure a homogeneous wasteform, and that control rods and mixed metal ILW would be grout cemented into $3m^3$ Boxes. The operational crud and resin waste streams would be packaged as they arise. The $3m^3$ Drums and Boxes 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 based on the use of UK standard waste containers consistent with RWM standards and specifications. The reactor vessel ILW is assumed to be grout cemented into 4m Boxes with 200-mm thick concrete walls. Following consideration by RWM of the dose rates from waste packages, it has been concluded that reactor internals ILW would need to be grout cemented into $3m^3$ Boxes.

The GDA Disposability Assessment for the spent fuel from the UK ABWR was based on it being over-packed for disposal. For the purposes of this assessment, disposal using robust disposal containers manufactured from either copper or steel has been considered. It has been concluded that each disposal container would contain twelve fuel assemblies from a UK ABWR. It is further assumed that the spent fuel would be delivered to the disposal facility packaged in the disposal containers, which in turn would be transported in a reusable transport container.

Radionuclide Inventory of ILW and Spent Fuel

The information supplied by Hitachi-GE on the radionuclide inventories of the identified ILW and spent fuel has been used to derive assessment inventories for the various proposed waste packages, including package-specific inventories for the ILW and spent fuel. In all cases, to ensure a full coverage of potentially significant radionuclides, it has been necessary to supplement the information supplied by Hitachi-GE using additional information available to RWM. The assessment inventories are intended to characterise the range of waste package inventories, taking account of the potential variability between packages, and other uncertainties. Typically, an assessment inventory includes a best-estimate (average) and bounding (maximum) inventory for a waste package to encompass such variability and uncertainty.

The uncertainties in the inventories arise from numerous sources, for example the detailed reactor operating regime adopted, including fuel burn-up, and the waste package loadings that would be achieved in practice. The GDA Disposability Assessment has used best endeavours to bound this uncertainty and thereby provide robust, conservative conclusions. It is anticipated that information on the inventories associated with the ILW and spent fuel would be refined as the design of the reactors and their operating regimes are developed further. Such information, together with more refined packaging proposals, would be considered at an appropriate time in the future through the Disposability Assessment process.

Uncertainties that will need to be addressed at later stages of assessment include information on the detailed compositions of the steels used for the control rods and mixed metal ILW, most notably the concentration of cobalt in the steels. Currently, a pessimistic value for cobalt concentration has been used as Hitachi-GE did not provide an accurate concentration.

Reactor water chemistry could have a significant impact on the crud and resin waste streams, and inventory calculations for these waste streams applied Hitachi-GE’s proposed UK ABWR-specific water chemistries, based on data supplied by Hitachi-GE but supplemented by RWM.

For decommissioning ILW, the potential for some of the reactor vessel steels being consigned to LLW management routes could be considered. Hitachi-GE activities of reactor pressure vessel ILW considered activation of the pressure vessel steels. In addition to activation, contamination of the steels could occur, for example, through deposition of corrosion products on the surface of the pressure vessel. Therefore, the impact of contamination was included in the enhanced inventories used to assess the reactor pressure vessels in this Disposability Assessment. When refined inventories are developed for use in more detailed stages of the Disposability Assessment process, the potential for contamination of reactor pressure steels would need to be considered alongside activation inventories.

The spent fuel assembly inventories have been checked by RWM using independent calculations and confirmed to be conservative. The spent fuel inventory supplied by Hitachi-GE is based on ORIGEN v2.2 modelling using Japanese nuclear data libraries. ORIGEN is a state-of-the-art isotope depletion and decay analysis code used internationally for safety analysis and licensing studies of used fuel facilities. There are some differences between the arrangement of fuel assemblies in a UK ABWR and the arrangement modelled in the data libraries used by Hitachi-GE, but the neutron spectra used in the modelling are considered to be representative. The RWM calculations used ORIGEN-ARP, which includes a 2D cross-section of the GE14 fuel assembly and produces a conservative estimate of radionuclide inventories.

RWM has concluded that the inventory data supplied by Hitachi-GE, augmented by supplementary data as required, has provided a robust and conservative data set sufficient to provide confidence in the calculations of the GDA Disposability Assessment.

This Disposability Assessment of the UK ABWR is the first time that disposal of wastes from a boiling water reactor has been considered in the UK; the most similar reactor that has previously been considered is the Pressurised Water Reactor (PWR). Boiling water reactors operate at lower pressures than PWRs, and the reactor pressure vessel is larger in volume. This means that there will be a greater mass of decommissioning wastes produced from the UK ABWR relative to a PWR. However, as the gap between the reactor core and the wall of the pressure vessel is filled with a mixture of stainless steel and water, the neutrons emanating from the core will be more strongly attenuated, resulting in lower activation of the steels in the reactor pressure vessel. The resulting wastes will therefore be less active and, assuming like-for-like packaging, would give lower dose rates. Fuel assemblies used in a UK ABWR are smaller than those used by PWRs. This would result in a larger number of spent fuel assemblies being packaged in each disposal container.

In PWRs, control rods are incorporated as part of the fuel assembly and might be managed alongside the spent fuel. In the UK ABWR, control rods are inserted between every four fuel assemblies, and are expected to be managed separately as ILW. Owing to their proximity to the fuel assemblies, these components see a high neutron flux and will need to be packaged and managed appropriately.

Despite the differences between the UK ABWR and PWRs, both designs are light water reactors, with fuel pellets fabricated from uranium dioxide with similar enrichments of U-235, and with broadly similar energy outputs. Both designs use zirconium-based cladding, and stainless and carbon steel, zirconium-based and Inconel metals in the spent fuel assembly and reactor vessel. Therefore, it is to be expected that the radionuclides listed in the waste and spent fuel inventories and the activities of these radionuclides will be broadly similar. This is borne out by a comparison of radionuclide inventories for the most active ILW stream and for spent fuel from the two reactor types that has been undertaken by RWM as part of the UK ABWR Disposability Assessment.

Assessment of Proposed ILW Packages

The proposals for the packaging of ILW are based on solid wasteforms that provide for the immobilisation of the activity associated with waste. Detailed arguments and supporting evidence on the performance of the proposed packages are currently not available. This is consistent with expectations for the GDA Disposability Assessment. In future, fully-developed proposals would need to be provided for assessment through the Disposability Assessment process.

The proposed use of cementitious 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 RWM standards and specifications, and, therefore, would be compliant with the systems assumed for transport of waste packages to, and disposal of waste packages in, a geological disposal facility, and also compliant with the associated safety cases for the facility. Meeting these standards and specifications might require specific packaging solutions, e.g. use of suitable loading factors and decay storage, for wastes with relatively high activities, for example the hafnium control rods.

The proposal to use RWM standard waste containers provides compliance with many aspects of the existing standards and specifications. Furthermore, the assessment has assumed that transport of the waste packages would be based on transport in a reusable shielded transport overpack to ensure compliance with the dose-rate limits set out in the IAEA Transport Regulations [22].

The resin waste streams are bead and powder mixed bed anionic and cationic cross-linked, polystyrene-based resins. Hitachi-GE has not provided information on the functional groups on the resins. Similar resins have been assessed previously, for example for Sizewell B, and deemed to be disposable. The chemical constituents would need to be further defined by the UK ABWR reactor operator in any subsequent Disposability Assessment submission for the ABWR wastes.

The assessment of the 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 would need to be consistent with meeting the regulatory risk guidance level[8], based on the approach adopted for Disposability 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 historical and currently arising ILW. The additional radionuclide inventory associated with the ILW from a UK ABWR represents only a small fraction of that of the historical and currently arising 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 UK ABWR in a site of the type described would be consistent with meeting the regulatory risk guidance level. The consideration of such a fleet of reactors does not alter this conclusion.

Particular issues that will require further evaluation if and when the plans for the ABWR are taken further include:

- The optimum time for disposal of the ILW. In particular, Hitachi-GE have proposed disposing of the wastes shortly after they arise. For some of the waste streams, this raises concerns in meeting transport limits and operational limits at the GDF. These can be addressed by a period of decay storage for the relevant wastes.
- Hitachi-GE proposed that the RPV decommissioning wastes were packaged in 4m boxes. The evaluations found that a significant period of decay storage would be required before some of the wastes from this waste stream could be transported and placed in the proposed GDF if these containers were used. It was therefore recommended that these wastes should be placed in $3m^3$ boxes and transported in Standard Waste Transport Containers.
- The control rods in the ABWR design differ from those in the previously assessed PWR designs where the potential exists to dispose of with the spent fuel. In the case of the ABWR, the control rods, both hafnium and boron carbide variants, are separate from the fuel assemblies and are proposed to be disposed of as ILW. The nature of these wastes is inherently challenging and they will require a period of decay storage prior to Hitachi-GE's proposal for grout encapsulation in $3m^3$ boxes. While they raise no insurmountable issues precluding disposal, they will need to be subject to further assessment as the disposal plans are further developed.

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 the need to demonstrate the expected performance of the proposed waste packages, these would be the subject of future assessment under the Disposability Assessment process when further details on the packaging proposals have been developed.

The potential impact of the disposal of UK ABWR operational and decommissioning ILW on the size of a geological disposal facility has been assessed. It has been concluded that the 'footprint area' required to dispose of ILW from a UK ABWR corresponds to approximately 45m of vault length for each UK ABWR (178m for a fleet of four reactors) for higher

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

strength rock. For the illustrative fleet of four UK ABWR reactors, this represents no significant change in the overall footprint compared with current assumptions based on the inventory for disposal.

Assessment of Spent Fuel Packages

Hitachi-GE has indicated that the GDA Disposability Assessment for the UK ABWR should assume that the reactor would use fuel elements made from uranium dioxide enriched in U-235, and operated to achieve a maximum fuel assembly average burn-up[9] of 60 GWd/tU and an average burn-up of 50 GWd/tU. These values are regarded as a relatively high burn-up, and are greater than experienced by spent fuel from historical and current LWR operations in the UK. The assessment of spent fuel disposal packages assumed that all of the spent fuel arose at the end of reactor operation; no account was taken for cooling and radioactive decay during interim storage prior to the end of reactor operations. This is a conservative assumption made to simplify the assessment and ensure that the conclusions were robust.

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 is recognised as an important consideration in the assessment of spent fuel from the UK ABWR when compared to the assessment of lower burn-up fuel, for example from reactors that have operated historically and are operating at the present.

Assessment of spent fuel packaging proposals was based on sealing the spent fuel inside durable disposal containers manufactured from suitable materials, which would provide long-term containment for the radionuclide inventory. Although the container material remains to be confirmed, the Disposability Assessment process considers the potential performance of both copper and carbon steel containers. In the copper container case, it is assumed that a cast-iron insert is used to hold and locate the spent fuel assemblies, and to provide mechanical strength. In the carbon steel container case, a carbon steel “tube and plate” basket is used to hold and locate the spent fuel assemblies.

The disposal container provides one component of the multi-barrier system used to ensure safety following closure of a geological disposal facility. In this assessment, the multi-barrier system is assumed to include additional engineered barriers and the geological barrier. The engineered barriers are designed to be compatible with the environment in which a geological disposal facility is constructed. In higher-strength rocks and lower-strength sedimentary rocks, it is assumed that a bentonite buffer will be emplaced around the waste packages, and engineered plugs will form seals to limit groundwater flow at key locations underground. In evaporite rocks, it is assumed that disposal galleries are backfilled with crushed rock salt, and that seals are placed to limit groundwater flow and radionuclide migration along access ways.

The materials used as part of the engineered barrier system, and the characteristics of the host rock, will affect the thermal criteria used to determine the acceptability of the heat output from waste packages consigned for disposal. In the current generic phase of the programme, generic thermal criteria are used to determine approximate cooling times required before disposal of spent fuel. Different thermal criteria are applied in the illustrative disposal concepts for different host rocks. In higher strength rock, the

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9 Burn-up will vary along the length of a fuel assembly and the ‘fuel assembly average burn-up’ is the average of the burn-up along the length of the fuel assembly. Different fuel assemblies will have different ‘fuel assembly average burn-ups’, and it is possible to define the ‘maximum fuel assembly average burn-up’ as the maximum of these. For the purposes of the GDA disposability assessment, RWM assume that this ‘maximum fuel assembly average burn-up’ is achieved by all fuel assemblies, which is a conservative assumption.

temperature criterion requires that the temperature of the inner surface of the bentonite buffer should not exceed 100^{o}C. In lower strength sedimentary rock, the temperature criterion is that the buffer temperature should not exceed 125^{o}C at its mid-point. In evaporites, the temperature criterion is that the temperature of the host rock should not exceed 200^{o}C. These limits are consistent with criteria used in disposal programmes in other countries.

Based on a spent fuel waste package containing twelve UK ABWR fuel assemblies and adopting the spacing used in the illustrative designs for higher strength rock, it would require between 50 and 100 years for the activity, and hence heat output, of the UK ABWR fuel to decay sufficiently to meet the existing temperature criterion. This period allows for both the range of predicted ABWR fuel burn-up (50-60GWd/tU) and the range of rock characteristics that may be encountered for ageological disposal facility at a depth of 650m.

The cooling time required to meet the temperature criteria in the lower strength sedimentary rock illustrative design has a greater range owing to a greater range in the thermal conductivity of the lower strength sedimentary host rocks that could be used to host a geological disposal facility. The cooling time required in lower strength sedimentary rocks is currently estimated to be between 50 and 130 years. This range is for the same burn-ups as the higher strength rock case.

For the illustrative designs in evaporite host rocks, the cooling time required is estimated to be less than 40 years. This is because of the higher temperature criterion on disposal of spent fuel in evaporitic host rocks and the higher thermal conductivity of evaporitic rocks. Therefore, the cooling times are likely to always be the shortest for disposal of spent fuel in evaporite host rocks.

These cooling times are dependent on a number of uncertainties, in particular the conservative assumptions made in developing the inventory for spent fuel, the uncertainty in the thermal conductivity of the host rock, and the details of the underground design (e.g. package spacing). These uncertainties could be reduced by further work, for example, through refinement of the assessment inventory, by taking into account the cooling of the spent fuel being stored prior to the end of the operational period. Ultimately, cooling times can be managed by consideration of alternative container and geological disposal facility designs. RWM continues to look at the options.

RWM 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 UK ABWR spent fuel may influence the arrangements, for example through the need for additional shielding, it is judged that sufficient flexibility exists in the outline designs for transport of spent fuel disposal packages to a geological disposal facility to allow suitable arrangements to be developed.

The GDA Disposability Assessment has considered how spent fuel disposal packages would evolve in the very long term following closure of a geological disposal facility, recognising that radionuclides would be released only subsequent to a breach in a disposal container. Subsequent to any container 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 and the environment.

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 container failure and is influenced by the properties of the spent fuel. The increased irradiation of the higher burn-up UK ABWR fuel could increase the IRF as compared to that for lower burn-up fuel. Available information on the performance of higher burn-up fuel has been used to provide suitably conservative IRFs for the 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 would need to be consistent with meeting the regulatory risk guidance level, this assessment assumed the same site characteristics as assumed for the existing RWM generic assessment. On the basis of the information provided by Hitachi-GE and conservative calculations of spent fuel waste package performance, it was calculated that the spent fuel from a fleet of four UK ABWR reactors would give rise to an estimated risk below the risk guidance level.

The risks calculated for the disposal of spent fuel reflect the assumed performance of the proposed packaging options. Sensitivity analysis has demonstrated that while the calculated risk would be influenced by the container material performance, coupled with the performance of other engineered barriers and the geological barrier, the risk was calculated to be below the regulatory guidance level. This outcome is insensitive to any impact that the high burn-up experienced by the fuel assemblies would have on the IRF.

RWM recognises that the performance of disposal containers will be an important element of a safety case for the disposal of spent fuel. Consequently, it is anticipated that RWM will continue to develop container designs, including the designs of containers for UK ABWR spent fuel, with the intention of substantiating the continued robustness of current assumptions and tailoring the designs to whatever site is ultimately identified.

The potential impact of the disposal of UK ABWR spent fuel on the size of a geological disposal facility has been assessed. The industry ambition of 16GW of nuclear new build has been estimated previously to produce spent fuel containers that will fill approximately 202 disposal tunnels in a geological disposal facility in high strength rock. The assumed operating scenario for a single UK ABWR gives rise to an estimated 800 spent fuel disposal containers, requiring approximately 18 disposal tunnels for disposal in higher strength rock. For the illustrative fleet of four UK ABWR reactors, representing 5.40GW, this would be equivalent to 72 disposal tunnels. This indicates that the required number of disposal tunnels is within the range assumed for a 16GW fleet of new nuclear build.

Conclusions

RWM has undertaken a GDA Disposability Assessment for the higher-activity wastes and spent fuel expected to arise from the operation of a UK ABWR. 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 RWM by Hitachi-GE. This information has been used to assess the implications of the disposal of the proposed waste packages against the waste package standards and specifications developed by RWM, and the supporting safety assessments for a proposed geological disposal facility. The safety of transport operations, handling and emplacement at a geological disposal facility in the UK, 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.

RWM has concluded that sufficient information has been provided by Hitachi-GE to produce valid and justifiable conclusions under the GDA Disposability Assessment. RWM has concluded that ILW and spent fuel from operation and decommissioning of a UK ABWR should be compatible with plans for transport and geological disposal of higher-activity wastes and spent fuel. It is expected that these conclusions would be supported and substantiated by future refinements of the 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, it is expected that more specific and detailed packaging proposals would be assessed, and potentially endorsed, through the established Disposability Assessment process for assessment of waste packaging proposals.

The GDA Disposability Assessment for the UK ABWR has not identified any significant issues that challenge the fundamental disposability of the wastes and spent fuel expected to be generated from operation of such a reactor. This conclusion is supported by the similarity of the wastes to the expected arisings from the existing PWR at Sizewell B. Given a disposal site with suitable characteristics, the wastes and spent fuel from the UK ABWR are expected to be disposable.

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

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. Hitachi-GE Nuclear Energy (Hitachi-GE), a strategic global alliance between Hitachi Limited and General Electric (GE) founded in 2007, is proposing to develop and construct nuclear reactors in the UK based on the United Kingdom Advanced Boiling Water Reactor (UK ABWR) design, and is, therefore, requesting assessment of the design under the GDA process.

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The GDA Disposability Assessment process comprises three main components: a review to confirm the waste and spent fuel properties; an assessment of the compatibility of the proposed waste packages with concepts for geological disposal of higher-activity wastes and spent fuel; and 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 the development of reactor designs and operating regimes, all proposals are necessarily outline in nature. However, this Disposability

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

Assessment has made assumptions to allow the production of a comprehensive and detailed data set describing the intermediate-level waste (ILW) and spent fuel to be generated from operation and decommissioning of a UK ABWR. At a later stage, more specific and detailed proposals will be required for endorsing waste packaging proposals through the existing Disposability Assessment process [5].

The assessment has been undertaken in response to the purchase order from Hitachi-GE dated 15 July 2014 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 Disposability Assessment Framework Contract between Hitachi-GE and RWM, dated 29 April 2014.

1.2 Objectives

The purpose of this GDA Disposability Assessment is to undertake assessment of the disposability of those higher-activity wastes and spent fuel expected to be generated from operation of a UK ABWR. The assessment has been commissioned by Hitachi-GE 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 Hitachi-GE, in the protocol issued by RWM 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 a UK ABWR, 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 (ILW and spent fuel) arising from operation and decommissioning of a UK ABWR;
- comment on initial proposals by Hitachi-GE 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 historical and current programmes. A supplementary objective of the GDA Disposability Assessment is to confirm whether the proposed wastes and spent fuel from a UK ABWR present no technical issues when compared to historical and currently arising 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 historical and currently arising wastes and spent fuel.

In addition, the White Paper flagged the importance of being able to give as much clarity as possible to communities considering hosting a geological disposal facility on the likely volume and the level of radioactivity of the disposal inventory 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 in the geological disposal facility siting process. In fulfilling this objective, RWM has presented additional information for a fleet of UK ABWR reactors noting that the actual impact on the UK's waste inventory from construction and operation of such a fleet of reactors would depend on the size of construction programme and the manner in which the reactors were operated.

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

1.3 Scope

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

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

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

In line with the White Paper [1], it is assumed that spent fuel from a new nuclear power programme will not be reprocessed and will 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 UK ABWR, 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 an assumed operation of four UK ABWRs, which would provide generating capacity of approximately 5.4 GW(e). This is consistent with current Hitachi-GE plans to construct two UK ABWRs at Wylfa and two at Oldbury. On the basis of roughly equivalent generating capacity, these are assumed to replace five AP1000 reactors in the 2013 inventory for disposal, which includes an indicative quantity of waste from new nuclear power stations. This illustrative approach is considered to be a straightforward and pragmatic assumption for this assessment, and no inference should be drawn for wider UK planning purposes.

1.4 Document Structure

This GDA Assessment Report for the UK ABWR 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 Hitachi-GE proposals have been assessed and the assessment methodology applied;
- Section 3 provides an overview of the UK ABWR, the assumptions regarding operation of a UK ABWR used in the GDA Disposability Assessment and summarises the inventory, packaging proposals, disposal package numbers and disposal package characteristics for UK ABWR ILW and spent fuel;
- Section 4 describes the assessment of UK ABWR operational and decommissioning ILW;
- Section 5 describes the assessment of UK ABWR spent fuel;
- Section 6 presents the conclusions;
- Appendix A provides a summary of the Disposability Assessment process;
- Appendix B lists issues identified during the assessment that would need to be addressed by plant operators in future Disposability Assessment interactions.

2 APPROACH TO GDA DISPOSABILITY ASSESSMENT

This section presents the approach used by RWM to the GDA Disposability Assessment, including, in Section 2.1, the background in terms of the disposal concepts assumed during the assessment, and the specifications and standards applied, and, in Section 2.2, an overview of the methodology used to assess the information supplied by Hitachi-GE.

2.1 Assessment Context

2.1.1 Illustrative Geological Disposal Facility Designs

Background

A geological disposal facility will be a highly-engineered facility, located deep underground, where radioactive waste will be isolated and contained within multiple engineered and natural barriers capable of preventing the release of harmful quantities of radioactivity to the surface environment. In order to identify potential sites where a geological disposal facility could be located, the Government is developing a voluntarism approach based on working with communities that are willing to participate in the siting process [7]. Development of the siting process is ongoing and no site has yet been identified for a geological disposal facility.

In order to progress the implementation of geological disposal in the absence of a specific site, illustrative geological disposal facility designs have been developed for three different generic host rocks:

- higher strength rock, for example, granite;
- lower strength sedimentary rock, for example, clay;
- evaporites, for example, halite.

Design and safety studies for a geological disposal facility are currently based on six illustrative geological disposal facility concept examples. These designs incorporate a separate disposal concept in each of these three generic host rocks for ILW, low-level waste (LLW) and depleted, natural and low-enriched uranium, collectively referred to as low-heat-generating waste (LHGW) and HLW, spent fuel, high-enriched uranium and plutonium, collectively referred to as high-heat-generating waste (HHGW) (Table 1) [8]. These illustrative designs have been developed drawing on work done both in the UK and in international programmes in a range of geological environments [9, 10] and aligned with requirements on the disposal system detailed in the Disposal System Specification (DSS) [11, 12].

The illustrative geological disposal facility concepts for higher strength rocks are considered to be bounding, i.e. wastes that are assessed to be disposable in higher strength rocks can be assumed to also be disposable in lower strength sedimentary rocks and evaporites. Therefore, this GDA Disposability Assessment has focused on the illustrative concepts for higher strength rocks. However, where there are potential issues associated specifically with disposal in lower strength sedimentary rocks or evaporites, these have also been explicitly considered in the assessment and are discussed in this report.

As the concept for higher strength rocks is bounding, presentation of the illustrative geological disposal facility designs below focuses on the designs for a geological disposal facility constructed in this generic host rock. Information on the designs for lower strength sedimentary rock and evaporites can be found in the generic disposal facility designs report [8].

Table 1 Sources of illustrative geological disposal concepts for host geological environments and classes of waste

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

Notes

a. Higher strength rocks – the UK LHGW concept and KBS-3V concept for spent fuel were selected due to availability of information on these concepts for the UK context.

b. Lower strength sedimentary rocks – the Opalinus Clay concept for disposal of long-lived ILW, HLW and spent fuel was selected because a recent OECD Nuclear Energy Agency review regarded the Nagra (Switzerland) assessment of the concept as state of the art with respect to the level of knowledge available. However, it should be noted that there is similarly extensive information available for a concept that has been developed for implementation in Callovo-Oxfordian Clay by Andra (France), and which has also been accorded strong endorsement from international peer review. Although we will use the Opalinus Clay concept as the basis of the illustrative example, we will also draw on information from the Andra programme. In addition, we will draw on information from the Belgian super container concept, based on disposal of HHGW in Boom Clay.

c. Evaporites – the concept for the disposal of transuranic wastes (TRU) (long-lived ILW) in a bedded salt host rock at the Waste Isolation Pilot Plant (WIPP) in New Mexico was selected because of the wealth of information available from this facility. The concept for disposal of HHGW in a salt dome host rock developed by DBE Technology (Germany) was selected due to the level of concept information available.

d. For planning purposes the illustrative concept for depleted, natural and low enriched uranium is assumed to be same as for ILW/LLW and for plutonium and highly enriched uranium is assumed to the same as for HLW/SF.

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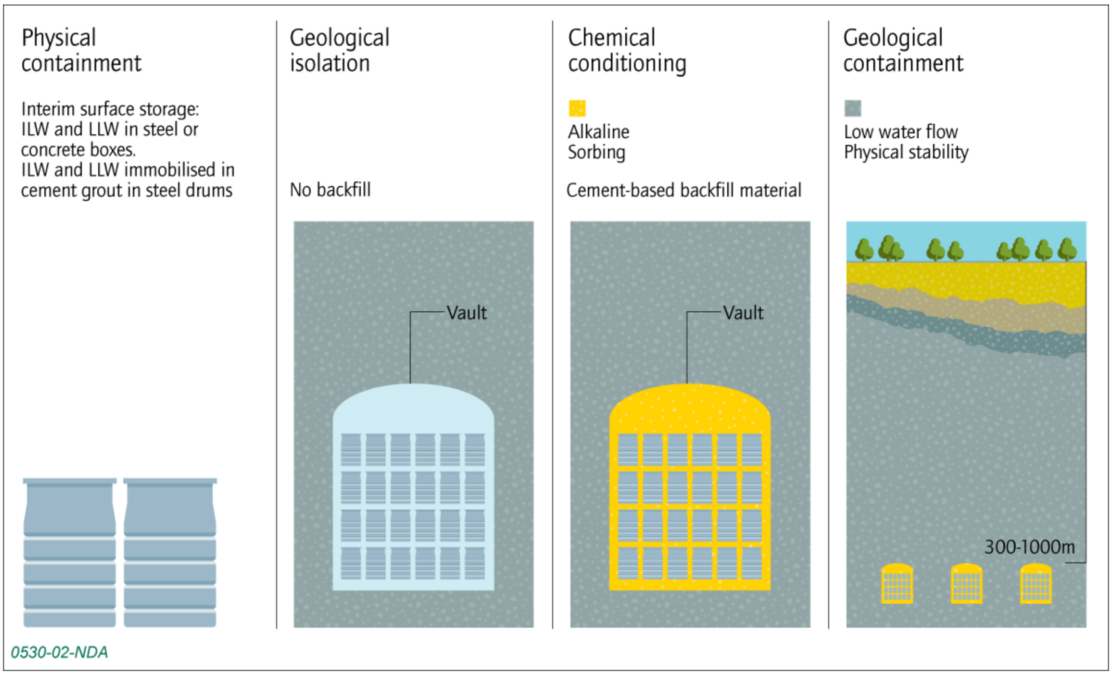
Illustrative geological disposal facility design for ILW in higher strength rock

The illustrative geological disposal facility design for ILW used in the provision of disposability advice envisages conditioning and packaging of ILW in standard, highly-engineered stainless steel or concrete containers (Figure 1). The waste packages would be emplaced in disposal vaults constructed at depth in higher strength rock. When it is time to ultimately close the facility, a cementitious 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.

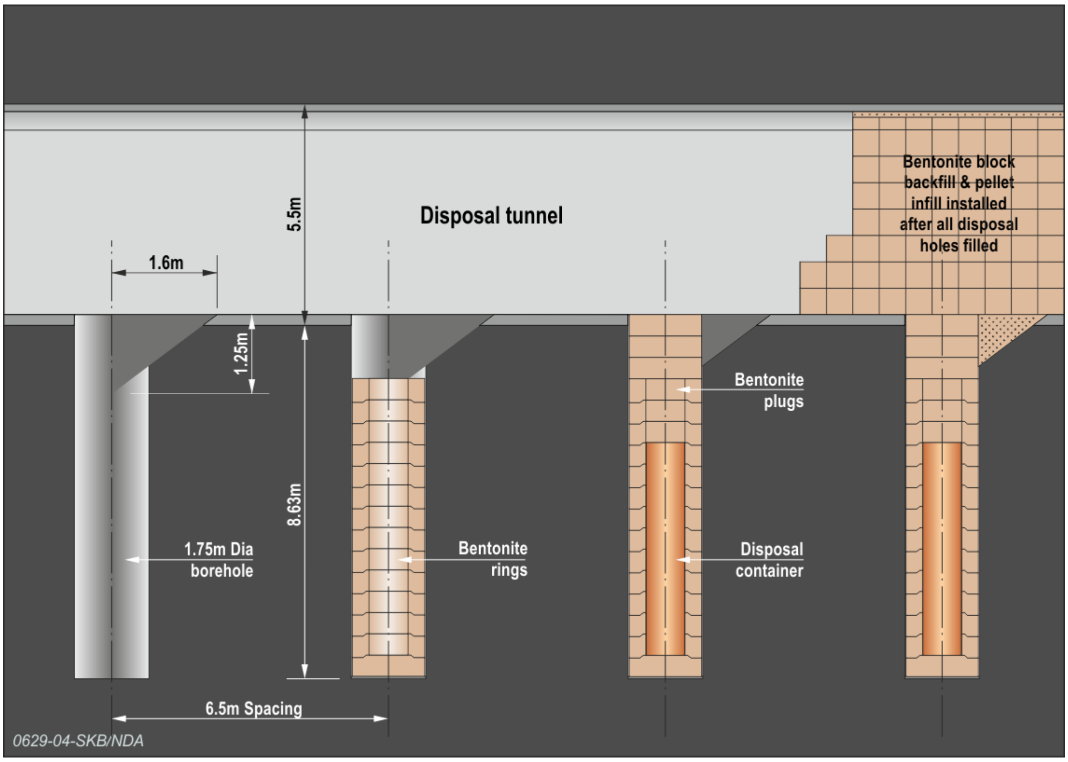
Figure 1 Illustrative concept for the disposal of ILW in higher strength rock



Illustrative concept for spent fuel in higher strength rock

Under the illustrative geological disposal concept for spent fuel in higher strength rock (Figure 2), spent fuel would be over-packed into highly-durable, corrosion-resistant disposal containers manufactured from suitable materials, which would provide long-term containment for the radionuclides contained within the spent fuel. Although the container material remains to be confirmed, the assessment has considered the potential performance of copper and steel containers. In the copper container case, it is assumed that a cast-iron insert is used to hold and locate the spent fuel assemblies, and to provide mechanical strength. In the carbon steel container case, a carbon steel “tube and plate” basket is used to hold and locate the spent fuel assemblies. These containers would be emplaced in disposal holes lined with a buffer made from compacted bentonite, which would swell following contact with water. As shown in Table 1, this illustrative concept is based on the KBS-3V concept developed by SKB for disposal of spent fuel in Sweden [13].

Figure 2 Illustrative concept for the disposal of spent fuel showing the disposal holes and emplacement of disposal containers



2.1.2 Waste Packaging Specifications

As implementer and future operator of a geological disposal facility, RWM will be responsible for the production of Waste Acceptance Criteria (WAC) for the facility. Whilst plans for the construction of a geological disposal facility remain at an early stage, the information necessary to define WAC is not available. In the meantime, and as a precursor to the final WAC, RWM produces generic specifications for packaged waste, which provide a baseline against which the suitability of plans to package waste for disposal can be judged. This assists radioactive waste owners and producers in the development and implementation of such plans by providing confidence that the resulting waste packages would be compatible with the anticipated needs for transport to and disposal in a geological disposal facility.

RWM has developed a hierarchy of waste packaging specifications. At the top level (Level 1), the Generic Waste Package Specification (GWPS) [14] defines high-level requirements for all waste packages destined for geological disposal. At Level 2, Generic Specifications have been developed to apply these high-level requirements to broad categories of waste and materials, including LHGW [15], and HLW and spent fuel [16].

Level 3 Waste Package Specifications apply the Level 1 and 2 requirements to specific designs of waste packages that would result from the use of standard waste containers that have been shown to be compatible with RWM's current plans for geological disposal. The packaging proposals for the ILW and spent fuel expected to arise from operation of the UK ABWR put forward by Hitachi-GE in discussion with RWM as part of the GDA Disposability Assessment have been assessed in relation to their compatibility with RWM's packaging specifications, as described in Section 4 (ILW) and Section 5 (spent fuel). In particular, ILW has been assessed against the Level 2 Generic Specification for LHGW [15], and spent fuel has been assessed against the Specification for HLW and spent fuel [16].

2.2 Assessment Approach and Constraints

2.2.1 Approach followed for GDA Disposability Assessment

The GDA Disposability Assessment of the UK ABWR was managed as a structured project using management procedures controlled under the RWM Management System. These management procedures were based on those applied to assessments undertaken under the existing Disposability Assessment process used by RWM to provide guidance to plant operators on conditioning and packaging of wastes. An overview of the Disposability Assessment Process is provided in Appendix A in order to provide the general context under which the approach to GDA Disposability Assessment is undertaken.

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

Conclusions have been drawn regarding the suitability of Hitachi-GE proposals through comparison of information on UK ABWR ILW and spent fuel with historical and currently arising wastes as follows:

- the key radionuclides and the quantities expected to arise as ILW and spent fuel have been compared to key radionuclides and their quantities in historical and currently arising 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 Disposability Assessment interactions.

Subsequent stages of the assessment considered the proposed waste packages and assessed performance using the approaches, safety assessments and “toolkits” developed for the Disposability Assessment 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, thermal output to estimate the length of the interim storage period, and estimates of risks to humans from migration of radionuclides to the biosphere following closure of a geological disposal facility, with risks considered for the groundwater pathway and gas pathways, human intrusion and criticality, and any environmental impacts from chemotoxic species potentially contained in the waste.

The packaging proposals provided by Hitachi-GE 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 UK ABWR were not available to RWM, and, therefore the assessment inventory has been supplemented by additional information based on assumptions regarding material composition made by RWM. Where this has been the case, RWM has adopted conservative or pessimistic assumptions and made this clear within this report.

2.2.2 GDA Disposability Assessment structure

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

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

Typical Disposability assessments would also consider data recording and quality management system (QMS) issues. However, these were not considered in the GDA Disposability Assessment for the UK ABWR at this stage and would need to be considered in any future Disposability Assessment interactions.

The work undertaken in each stage is described in further detail below.

Stage 1: Nature and Quantity of Waste

This stage comprised a nature and quantity of waste evaluation and a wasteform evaluation. Work under this stage used information supplied by Hitachi-GE, supplemented by existing RWM experience and extensive discussion with Hitachi-GE regarding the dataset used as a basis for this assessment. In particular, the radionuclide inventory for spent fuel was supplemented by undertaking calculations using ORIGEN-ARP [17]. Thermal modelling of the impact of the disposal of spent fuel on near-field temperatures was undertaken to determine the cooling times required before spent fuel could meet the requirements for disposal in a geological disposal facility.

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

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

The nature and quantity of waste 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 objective of the wasteform evaluation was to consider the chemical and physical characteristics of the wasteforms, which required:

- collation of information on proposed conditioning and packaging methods for ILW, including development of techniques as required;
- development of an understanding of organic materials content, potential for gas generation and chemo-toxic content for ILW;
- describing the 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.

Stage 2: Disposal Facility Design Assessment

This stage comprised a waste package performance evaluation and a design impact evaluation.

The waste package performance evaluation considered impact and fire performance of waste packages relevant to possible accident scenarios in transport of waste packages to a geological disposal facility and operations in a geological disposal facility, including estimation of release fractions for a range of standard impact and fire scenarios. 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 of waste and spent fuel generated from the operation of a UK ABWR on the design of a geological disposal facility, including the following:

- the number of disposal tunnels/vaults needed to accommodate the wastes, and the consequent impact on overall geological disposal facility footprint;
- compatibility of waste packaging assumptions with existing design assumptions;
- identification of unique or distinguishing features of the wastes and/or proposed waste packages;
- significance of potential variability in the proposed waste packages;
- consideration of the proposed conditioning or management methods.

The disposal facility design evaluations for ILW and spent fuel are presented in Sections 4.2 and 5.3 respectively.

Stage 3: Safety, Environmental and Security Assessments

This stage comprised a transport safety assessment, an operational safety assessment, a post-closure safety assessment, consideration of environmental issues, and a security evaluation. The safety, environmental and security assessments considered the compatibility of potential operational and decommissioning ILW, and spent fuel from a UK ABWR with existing assessments of RWM reference disposal concepts. The assessments provide the basis for judging the potential disposability of operational and decommissioning ILW, and spent fuel arising from operation of the UK ABWR:

- the transport safety assessment considered the logistics, regulatory compliance and risk of transport operations, with specific consideration of radiation dose, gas generation, containment and heat output under normal and accident conditions - the transport safety assessments for ILW and spent fuel are presented in Sections 4.2 and 5.3 respectively;
- the operational safety assessment considered radiation dose due to accidents, effects of gas generation and criticality safety - 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 longer-term radiological exposure from the groundwater and gas pathways, human intrusion and criticality, and any environmental impacts from chemotoxic species potentially contained in the waste - the post-closure safety assessments for ILW and spent fuel are presented in Sections 4.3 and 5.4 respectively;
- the evaluation of non-radiological environmental issues considered the materials, i.e. resource use, in a geological disposal facility to dispose of the ILW and spent fuel arising from the UK ABWR using the illustrative designs, and commented on proposed waste management strategies and their implications - the environmental

evaluation for ILW and spent fuel are presented in Sections 4.2 and 5.3 respectively;

- the security evaluation considered the likely security categorisation of the proposed waste packages and included commentary on proposals for accountancy and independent verification of the use of nuclear materials - the security evaluations for ILW and spent fuel are presented in Sections 4.2 and 5.3 respectively.

3 UK ABWR OPERATION, WASTES, PACKAGING PROPOSALS AND WASTE PACKAGE CHARACTERISTICS

This section provides a summary of the information used in the GDA Disposability Assessment for the UK ABWR. RWM used the information supplied by Hitachi-GE, supplemented as necessary by information available to RWM and by further calculations, to provide a comprehensive dataset of information covering waste package numbers, inventories and general characteristics of the conditioned and packaged ILW and spent fuel.

This following information is included:

- summary description of a UK ABWR (Section 3.1);
- assumptions regarding the operation of a UK ABWR (Section 3.2);
- description of the ILW streams and spent fuel that will be generated through operation and decommissioning of a UK ABWR (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 UK ABWR wastes in context, the expected ILW and spent fuel from the proposed reactor system are compared to those reported from the Sizewell B nuclear power station. Sizewell B was selected because it represents the sole example of a light water reactor (in this case a pressurised water reactor (PWR) rather than a boiling water reactor (BWR)) operated in the UK, and the ILW and spent fuel from the power station are well understood by RWM.

The waste volumes, package numbers and activities presented in this section comprise the assessment inventory for the UK ABWR Disposability Assessment. The implications of the assessment inventory are discussed in Sections 4 and 5.

3.1 Summary of UK ABWR Design and Operation

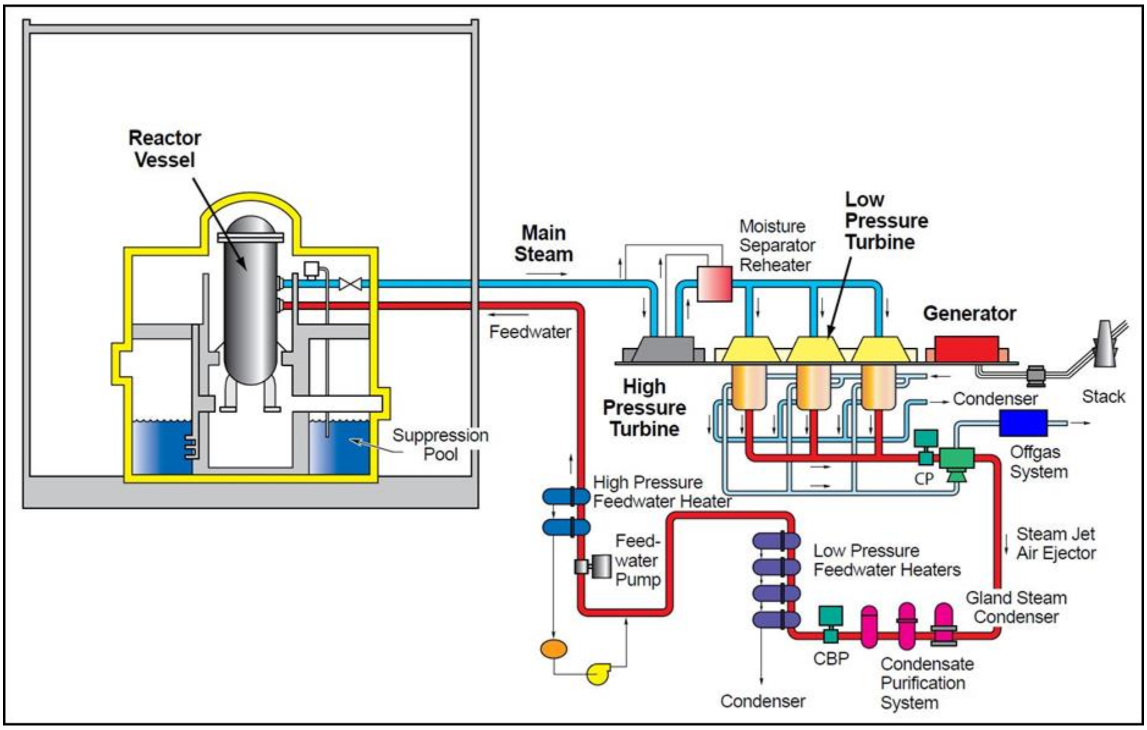
The UK ABWR is an evolutionary BWR design with an electrical power output of 1,350 MW(e).

The ABWR design is based on over 50 years' experience of operating BWRs across the world. In 2007, there were 93 BWRs operating worldwide, with 32 plants operating in Japan and 37 in the United States [18]. The development of the ABWR design was undertaken in the 1980s, with an Establishment Permit, or licence, being issued in Japan in 1991, and a Design Certification approved and signed into law in the United States in 1997. By 2007, four ABWR units had been constructed and were operational in Japan [18].

In BWRs such as the ABWR, ordinary (light) water is utilised to remove the heat produced inside the reactor core by thermal nuclear fission that occurs within the fuel assemblies. This water also slows down (or moderates) neutrons (the constituents of atomic nuclei that are released in the nuclear fission process). Slowing down neutrons is necessary to sustain the nuclear reaction. The heat produced inside the reactor core causes the water to boil, and the resultant steam drives a steam turbine.

The reactor core, which provides the heat source for steam generation in the form of nuclear fuel assemblies, is housed in a reactor vessel (Figure 3). In the ABWR, reactor coolant is forced through the fuel assemblies in the reactor core using an arrangement of ten pumps mounted internally in the reactor vessel. In previous BWR designs, pumps were housed in external recirculation loops; these have been eliminated in the ABWR design. Steam, generated in the reactor, is supplied to the high-pressure turbine and to steam reheaters. Steam leaving the high-pressure turbine passes through a combined moisture separator/reheater, prior to entering a series of low-pressure turbines. Water is collected in drains, combined with exhausted steam, and fed to a condenser and purification system, before circulating back to the reactor through a series of low-pressure and high-pressure pumps.

Figure 3 Principal components of a UK ABWR, reproduced from [18]



3.2 Assumptions

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

- The UK ABWR would be operated for 60 years. During the operation of the reactor, nuclear fuel assemblies would be periodically rotated within the reactor core, and then removed and replaced with other fuel assemblies.
- The date at which operation of power production from a UK ABWR would commence in the UK is uncertain. In the GDA Disposability Assessment for the UK ABWR, estimates of time-dependent properties, e.g. those related to radioactive decay, are assessed from time of generation of the waste. Discussion of the implications for management of radioactive waste assumes that the reactors operate from 2020. This is recognised as being overly optimistic but is the same assumption that was made for GDA Disposability Assessment of the European Pressurised Water Reactor (EPR) and the Advanced Passive Pressurised Water Reactor (AP1000) [19, 20].
- Spent fuel characteristics have been determined on the assumption that the reactor would be operated to achieve a maximum fuel pin burn-up of 65 GWd/tU. This corresponds to a maximum fuel assembly average burn-up of 60 GWd/tU and an average burn-up of 50 GWd/tU [11]. The GDA Disposability Assessment has

[11] Burn-up will vary along the length of a fuel assembly and the 'fuel assembly average burn-up' is the average of the burn-up along the length of the fuel assembly. Different fuel assemblies will have different 'fuel assembly average burn-ups', and it is possible to define the 'maximum fuel assembly

developed inventories based on assumed burn-ups of 60 and 50 GWd/tU. This is a conservative approach and ensures that the conclusions from the assessment are bounding of a wide range of possible operational behaviours.

- The fuel used in the UK ABWR will be manufactured from mined uranium (i.e. not reprocessed uranium) and would not contain any U-236 prior to irradiation.
- It is assumed that ILW and spent fuel from the UK ABWR will arrive at a geological disposal facility in a packaged state, ready for disposal.

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

ILW is defined in the UK in a number of sources (e.g. [7]) as:

> *'Radioactive wastes exceeding the upper activity boundaries for low level waste (LLW) but which do not need heat to be taken into account in the design of storage or disposal facilities.'*

All radioactive waste produces radiogenic heat from the radioactive decay of the radionuclides associated with them. The radiogenic heat output of wastes classed as ILW is generally low in the conventional sense; the average heat output of all of the ILW waste streams recorded in the UK Radioactive Waste Inventory (UKRWI) [21], when conditioned for disposal, being ~1Wm^{-3}. In most contexts such a heat output would be considered low, however, heat output does have to be considered for the design of transport and geological disposal facility systems, even for these low heat outputs, where various regulatory and operational constraints on temperature will apply.

In addition, the average heat outputs of some ILW waste streams are up to two orders of magnitude higher than the overall average value for ILW and there is also significant variation within some waste streams which could result in even higher radiogenic heat outputs for individual waste packages. Radiogenic heat output is therefore not the only discriminator used to define a waste stream as ILW, and the wastes considered as ILW in the UK ABWR are done so on the basis that they will be managed and disposed of using generic concepts for ILW.

3.3.1 Operational ILW Streams and Packaging Assumptions

Hitachi-GE has indicated that eight operational ILW streams[12] would arise from normal operation of a UK ABWR:

- Cruds: Crud is solid material from the backwashing of filters. It is mainly composed of corrosion and erosion products from the reactor internals (primarily derived from steel alloys) and other water circulation systems within the plant. There are two ILW crud waste streams in the UK ABWR inventory:
    - UKABWR01: Condensate Filter Facility (CF) Crud. CF Crud waste arises in the condensers after the steam has passed through the turbines. The material includes small quantities of corrosion product which may have been carried over with the steam from the reactor vessel.

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average burn-up' as the maximum of these. For the purposes of the GDA disposability assessment, RWM assume that this 'maximum fuel assembly average burn-up' is achieved by all fuel assemblies, which is a conservative assumption.

[12] "Operational" ILW includes some of the ILW generated during post-operational decontamination operations, e.g. Post-operational Decontamination Resins. These wastes have been assessed alongside the operational ILW as they are of a similar chemical and physical form to the ion exchange resins generated during operations.

- UKABWR02: Low Conductivity Waste (LCW) Crud. The LCW system collects wastes from various sources including the Reactor Building, Turbine Building and Radwaste Building drains, and processes them via filters, demineraliser and sampling tanks. LCW Crud arises from the backwashing of the filters in the LCW system.

In addition to CF and LCW Crud, the UK ABWR generates other crud waste streams, but these are expected to be classified as LLW. It is uncertain if the Crud waste streams include material derived from the turbines. It can be assumed that the turbines will be built to a high standard, and there would be an efficient filter system in the return line. However, this will need to be checked against more detailed information in future stages of the disposability assessment process. A crud capture system is also incorporated in the spent fuel assembly (the lower tie plate debris filter); this will be disposed of along with the spent fuel.

- Resins: Ion-exchange resins are composed of powder and beads, which are used for removal of dissolved radioactivity from the reactor coolant. There are three resin waste streams in the UK ABWR inventory:
  - UKABWR03: Reactor Water Clean-up (CUW) Resin. CUW Resin is a powder-based polystyrene resin.
  - UKABWR04: Fuel Pool Cooling Clean-up (FPC) Resin. FPC Resin is a powder-based polystyrene resin.
  - UKABWR05: Post-operational Decontamination (DEC) Resin. DEC Resin is a bead-based polystyrene resin.
- Control Rods: The reactor core will contain 205 control rods during operation. The cruciform control rods (Figure 4) contain stainless steel tubes in each wing of the cruciform filled with compacted boron carbide (B_4C) powder. In selected control rods, the boron carbide powder is replaced with hafnium in solid metallic form. The tubes act as pressure vessels to contain the helium gas released by the boron-neutron capture reaction. The tubes are held in cruciform array by a stainless steel sheath extending the full length of the tubes. There are two control rod waste streams in the UK ABWR inventory, one for each type of neutron absorber:
  - UKABWR06: Hafnium Control Rods.
  - UKABWR07: Boron Carbide Control Rods.
- Activated Metals: This waste stream consists of monitoring probes and neutron sources deployed within the reactor core. These include Local Power Range Monitoring (LPRM) assemblies (these contain fission chamber detectors for monitoring reactor pressures and sensors for monitoring the neutron flux – the Automatic Traversing In-core Probe (ATIP)); Neutron Source Units (which contain antimony and beryllium); and Start-up Range Neutron Monitor (SRNM) assemblies (which contain fission chamber sensors). There is a single activated metals operational waste stream in the UK ABWR inventory:
  - UKABWR08: Mixed Metal ILW.

The raw waste volumes of each operational ILW waste stream expected to arise through operation of a UK ABWR, as determined by Hitachi-GE, are provided in Table 2.

To package the cruds and resins, it is assumed that the wastes would be grout cemented into $3m^3$ Drums. To accommodate all of the wastes arising from a 60-year operational lifetime for a single UK ABWR would require 79 off $3m^3$ Drums of Crud and 452 off $3m^3$ Drums of Resin. These package numbers are based on an assumed volume conditioning factor of 2 for the Cruds and 3 for the resins. Based on experience from Sizewell, RWM is of the view that the conditioning factor of 3 that has been applied for the resins may be optimistic; a conditioning factor of 10 may be more appropriate. More information and substantiation of the conditioning factors will be required during further interactions under the Disposability Assessment process.

Figure 4 Illustration of an ABWR control rod, from [18]

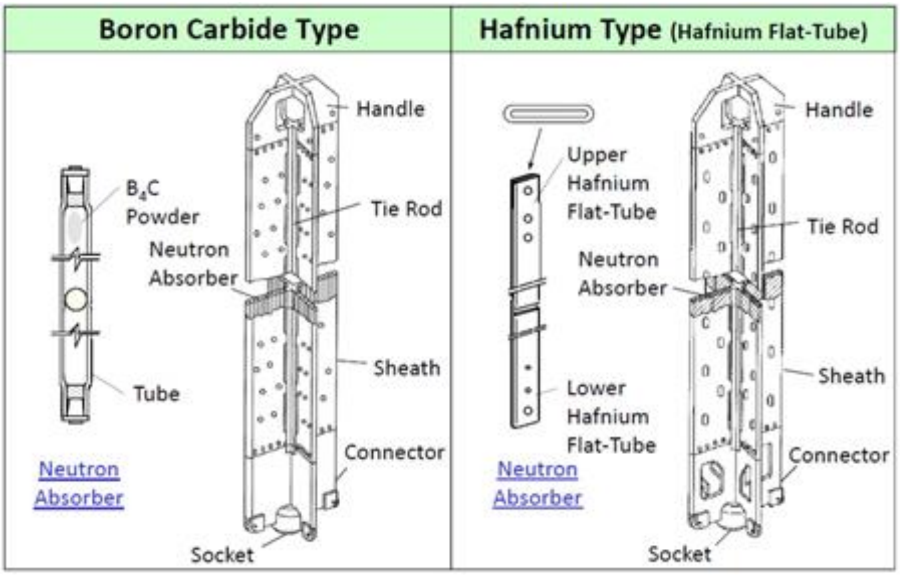


Table 2 Raw waste volumes for operational ILW from a UK ABWR and identifiers used in the GDA Disposability Assessment

Waste Stream	Identifier	Raw Waste Volume/Mass
Cruds: CF Crud	UKABWR01	$72.0m^3$
Cruds: LCW Crud	UKABWR02	$18.0m^3$
Resins: CUW Resin	UKABWR03	$187.6m^3$
Resins: FPC Resin	UKABWR04	$84.5m^3$
Resins: DEC Resin	UKABWR05	$67.0m^3$
Hafnium Control Rods	UKABWR06	31.65t
Boron Carbide Control Rods	UKABWR07	27.95t
Mixed Metal ILW	UKABWR08	33.00t

For transport, the $3m^3$ Drums would be carried inside a Standard Waste Transport Container (SWTC), which is being developed by RWM 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 $3m^3$ Drums containing the CF Crud would need to be transported in an SWTC-70 to meet the 2012 International Atomic Energy Agency (IAEA) Transport Regulation dose rate requirements [22]. The $3m^3$ Drums containing LCW Crud, and CUW, FPC and DEC Resin would need to be transported in an SWTC-285 to meet the 2012 IAEA Transport Regulation dose rate requirements [22].

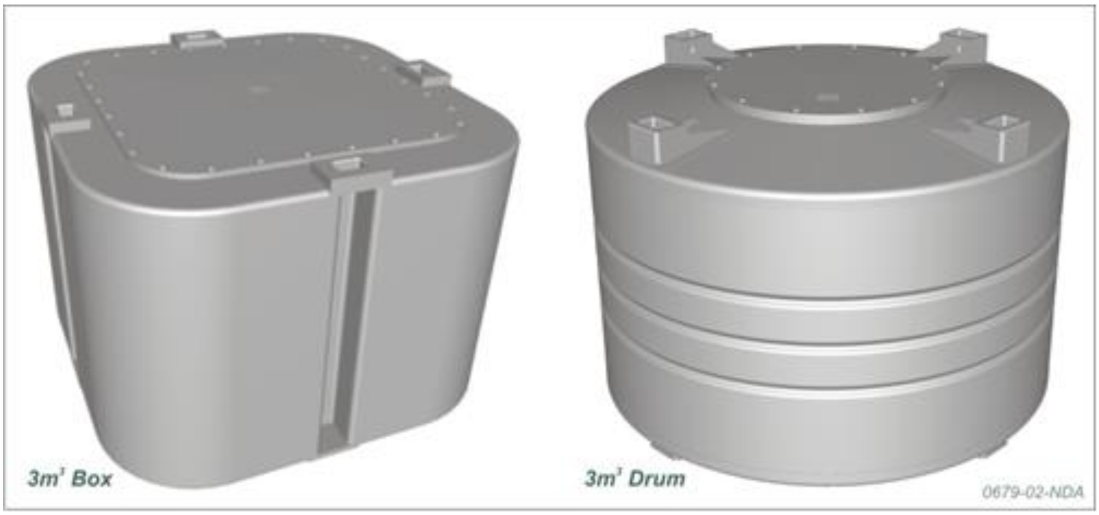
To package the Control Rods and Mixed Metal ILW, it is assumed that the wastes would be grout cemented into $3m^3$ Boxes. To accommodate all of the wastes arising from a 60 year operational lifetime for a single UK ABWR would require 17 off $3m^3$ Boxes of Control Rods, and 4 off $3m^3$ Boxes of Mixed Metal ILW.

The control rods may have to be cut prior to packaging, and future interactions under the Disposability Assessment process will need to consider how control rods are cut without release of the boron carbide powder. To fit into $3m^3$ boxes, Control Rods would have to be cut into at least four pieces. Diagrams of the control rods suggest that the rods are sectioned so that cuts could be made through sections of stainless steel not containing any B_4C powder (Figure 4). However, accidental releases of boron carbide powder could still be possible and the operator of a UK ABWR should have a contingency plan in place in case this should happen and the affected rods cannot be grouted. Evidence that control rods can be cut up is available from PWRs.

As with the Boron Carbide Control Rods, Mixed Metal ILW wastes are long compared to waste containers. A range of methods for packaging these wastes can be envisaged, including cutting, disassembly and folding. The method of packaging needs to be established in more detailed interactions under the Disposability Assessment process.

The package numbers assume that approximately 40 control rods would be packaged in each $3m^3$ Box. This packing density is considered to be high (a value closer to 12-15 control rods per package may be more realistic), and should be reconsidered at later stages of assessment in the future through the Disposability Assessment process. The number of Mixed Metal ILW packages is based on a packing density of 7.58t of raw waste per package. This packing density is considered unlikely to be feasible on volume grounds and should be reassessed in future interactions under the Disposability Assessment process. The $3m^3$ Boxes would need to be transported in an SWTC-285 to meet the 2012 IAEA Transport Regulation dose rate requirements [22]. Both the $3m^3$ Drum and $3m^3$ Box are standard RWM waste containers and are illustrated in Figure 5.

Figure 5 Illustration of a $3m^3$ Box (left) and a $3m^3$ Drum (right) as proposed for packaging of operational and decommissioning ILW from a UK ABWR



Based on experience from previous GDA assessments, the wastestreams identified for the UK ABWR are considered to be relatively comprehensive. There are a number of resin and crud wastestreams that are, based on current Hitachi-GE assumptions, likely to be ILW upon discharge from the reactor but will be classified as LLW once packaged, these are therefore not included in this assessment. Given these wastestreams will be similar in nature to the resin and crud ILW streams identified above and considered by this assessment, the latter should be bounding.

RWM has discussed with Hitachi-GE the possibility of additional ILW, such as contaminated clothing, boots and other such materials arising as a result of reactor operations. Hitachi-GE is of the view that these materials will be classified as LLW and, for example, would only be ILW in non-standard operation scenarios. In any case, RWM is of the opinion that such material in limited quantities would not present any significant disposability issues and that methods exist to successfully condition and treat such wastes.

Hitachi-GE has stated that there will be no miscellaneous contaminated items that would be classified as ILW. This needs to be stated in the design and operation protocols if it remains to be an assumption for disposability assessments, as some reactors do produce ILW contaminated metal items during operation. For example, there are examples of component replacement (such as pressurisation system components for PWRs) during outages. Hitachi-GE need to be aware that this is a potential route for generation of operational ILW and to report the consignment route for such material.

There is an option to dry store the UK ABWR fuel in a purpose built building. The radiation levels from the discharged spent fuel are such that some neutron activation of the steel and concrete store structure would be likely to occur if dry storage was adopted. In the case of the materials closest to the fuel this may lead to an additional ILW stream that has not been assessed in the GDA Disposability Assessment, as the store was only one option put forward at the time of the submission. As the store building is likely to be fabricated using standard construction industry materials (e.g. carbon steel, stainless steel and concrete) it is considered that while this will result in a small number of additional waste packages, they will not present any particular issues for the transport, operational and post-closure safety cases.

3.3.2 Decommissioning ILW Streams and Packaging Assumptions

Decommissioning ILW has been assigned to two broad waste streams: UKABWR09: Reactor Pressure Vessel Internals; and UKABWR10: Reactor Pressure Vessel.

UKABWR09 (Reactor Pressure Vessel Internals) consists of in-vessel stainless steel structures that support the reactor core and its safety systems, and manage the flow of coolant into and out of the core. Because the neutron flux falls rapidly with distance from the core, this waste has been subject to a wide range of neutron irradiation levels.

Hitachi-GE's original proposals for packaging this waste stream were to utilise 4-metre Boxes. However, preliminary analysis of this packaging proposal by RWM demonstrated that it would not be possible to meet the 1m dose rate limit contained in the 2012 IAEA Transport Regulations [22] for any credible cooling period following reactor shutdown. Therefore, following discussion between RWM and Hitachi-GE, the proposed packaging assumption for Reactor Pressure Vessel Internals was modified to be for grout encapsulation in $3m^3$ Boxes transported in an SWTC-285. Such an arrangement provides a much enhanced level of gamma shielding compared to packaging in a 4-metre Box and transporting as an IP-2 package.

Stainless steel filter housings added to the reactor following operation when the reactor would be flooded for decommissioning purposes and with relatively high activity, would also be packaged in $3m^3$ Boxes are therefore also included in this waste stream[13].

The raw waste mass for this waste stream is approximately 374t, with 126 waste packages required to accommodate the whole waste stream, assuming a waste loading of ~3t of waste per $3m^3$ Box. The waste loading assumption is consistent with previous GDA Disposability Assessments undertaken by RWM (e.g. [23]).

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13 Filter housings are generated during post-operational decontamination operations. These wastes have been assessed as part of the Reactor Pressure Vessel Internals waste stream as they are of a similar chemical and physical form, and would be packaged in a similar manner.

UKABWR10 (Reactor Pressure Vessel) consists of the carbon steel reactor vessel and the stainless steel liner on the inside of the vessel. Hitachi-GE proposals are for cement encapsulation of this waste in 4-metre Boxes with 200-mm concrete walls for shielding. Some stainless steel filter housings with relatively low activity would be packaged in 4-metre Boxes, and have therefore also been included in this waste stream. Thirty-nine waste packages would be required to accommodate the whole waste stream, assuming a waste loading of ~16.7t of waste per 4-metre Box. This assumption is derived from consideration of the total mass of the waste stream and the total number of 4-metre Boxes Hitachi-GE original proposed to package decommissioning ILW. The 4-metre Boxes would be transported as IP-2 packages.

The raw waste masses of the decommissioning ILW waste streams expected to arise through operation of a UK ABWR, as determined by Hitachi-GE, are provided in Table 3. The 4-metre Box is a standard RWM waste container and is illustrated in Figure 6.

Table 3 Raw waste masses for decommissioning ILW from a UK ABWR and identifiers used

Waste Stream	Identifier	Raw Waste Mass
Reactor Pressure Vessel Internals	UKABWR09	374t
Reactor Pressure Vessel	UKABWR10	646t

Figure 6 Illustration of a 4-metre Box as proposed for packaging some decommissioning ILW from a UK ABWR

3.3.3 ILW Package Numbers and Characteristics

Overview

The information supplied by Hitachi-GE on the radionuclide inventories of the identified wastes and spent fuel 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 Hitachi-GE with information available to RWM. The assessment inventories are intended to characterise the range of waste package inventories, taking account of uncertainties and variability between packages.

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

- A best-estimate (average) waste package inventory. This inventory, when taken with the number of waste packages, defines the total inventory associated with the waste stream. This inventory is applied during the post-closure assessment and some aspects of operational safety assessment.
- A bounding (maximum) inventory for the waste package. This is used for transport safety assessment and certain aspects of the operational safety assessment where individual waste packages are considered.

The UK ABWR ILW waste package radionuclide-related parameters and waste quantities (package numbers and total packaged volume) are given in Table 4. Radionuclide related parameters (e.g. dose rate) are calculated at the time of arising (i.e. zero-decayed for cruds and resins ILW; reactor closure plus 6 years for control rods; reactor closure plus one year for reactor control housings and 40-year-decayed for decommissioning ILW).

For operational ILW, information on the raw waste volumes, package types, package numbers and radionuclide contents were derived from consideration of operation of existing BWRs in Japan, Europe and the US. Hitachi-GE provided radionuclide inventories containing the concentration of most of the key radionuclides. These datasets were "enhanced" by estimating the concentration of all of the 112 radionuclides considered by RWM to be potentially significant for radioactive waste management. The enhancement method was specific to each type of waste and is described below.

Different enhancement approaches were used for each type of material (cruds, resins, control rods, activated metals and reactor pressure vessel steels) as described below.

Table 4 UK ABWR Waste Stream Data: ILW (1) (2)

Waste Stream (3)	Package Type	Number of Packages	Average Package Alpha Activity (TBq)	Average Package Beta/ Gamma Activity (TBq)	Average Package A_2 Content	Average Package Heat Output (Watts)	Average Transport Package Dose Rate at 1m from Package (mSv/hr)	Additional Time Required for Average Package to Decay to the Transport Dose Rate Limit of 0.1 mSv/hr at 1m (years)
The average package data for the 5 wastestreams below are at reactor discharge. These are packaged cruds and ion-exchange resins. See Table 2.								
UKABWR01	$3m^3$ Drum	79	1.73E-05	2.29E-02	1.56E-02	7.67E-04	6.70E-07	0
UKABWR02	$3m^3$ Drum		2.38E-04	2.73E-01	2.02E-01	9.55E-03	8.53E-06	0
UKABWR03	$3m^3$ Drum	452	7.11E-04	9.17E+00	9.91E+00	1.28E+00	1.52E-03	0
UKABWR04	$3m^3$ Drum		2.80E-04	3.56E+00	3.86E+00	5.00E-01	5.94E-04	0
UKABWR05	$3m^3$ Drum		3.06E-02	1.31E+02	1.97E+02	2.00E+01	8.93E-03	0
The average package data for the 2 waste streams below are at closure plus 6 years. These are packaged hafnium and boron carbide control rods. See Table 2.								
UKABWR06	$3m^3$ Box	17	7.68E-03	2.72E+03	2.97E+04	7.17E+02	5.12E-01	13
UKABWR07	$3m^3$ Box		2.45E-04	1.33E+03	1.09E+03	1.78E+02	1.84E-01	5
The average package data for the wastestream below is at reactor closure plus one year. These are packaged reactor control instrument housings, etc. See Table 2.								
UKABWR08	$3m^3$ Box	4	9.00E-03	1.42E+04	2.00E+04	3.30E+03	2.02E+00	23
The average package data for the 2 decommissioning wastestreams below are at reactor closure plus 40 years. These are packaged RPV stainless steel structures and the RPV and its stainless steel lining. See Table 3.								
UKABWR09	$3m^3$ Box	126	1.58E+00	1.12E+06	1.83E+02	2.04E+01	1.17E-02	0
UKABWR10	4-metre Box	39	5.53E-05	1.09E+01	8.40E-03	8.80E-04	7.91E-04	0
TOTALS		717						

Notes:

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

(2) Dose rate refers to that 1m outside an SWTC-285 for all waste streams except UKABWR10. For UKABWR10, the dose rates are 1m outside of a 4-metre Box with 200-mm concrete shielding.

(3) See Section 3.3.1 for a description of UKABWR01 to UKABWR08 waste streams, and Section 3.3.2 for a description of UKABWR09 and UKABWR10 waste streams.

Cruds and Resins

Crud and resin assessment inventories were based on zero-cooled data, i.e. no account was taken for interim storage in the transport and operational safety assessments that were based on the assessment inventories.

For cruds and resins, Hitachi-GE provided an inventory containing activity data for 31 potentially significant radionuclides. An original data set provided by Hitachi-GE was derived by using measured radionuclide concentrations in reactor coolant in operating ABWRs in Japan and undertaking activity balance calculations to determine the exchange of contaminants between the coolant and solid phases (e.g. ion-exchange resins in the condensate purification system and in the reactor water clean-up system).

Hitachi-GE has proposed the use of a new coolant chemistry regime in the UK ABWR, designed to reduce the release of radionuclides to the coolant water. This includes modifications to the concentration of zinc ions in the coolant (zinc would replace some of the radioactive cobalt potentially deposited on stainless steel surfaces, and thereby reduce the dose rate from these materials) and the use of hydrogen-dosed water (to reduce the potential for radiolysis and its impact on stress corrosion cracking corrosion). This has not previously been used in Japan. Therefore, a revised data set was developed for the UK ABWR GDA Disposability Assessment. The revised dataset used the coolant chemistries proposed for use in the UK to estimate the dissolution rate of activated metals in the reactor, benchmarked against operational experience with the ABWR.

In order to enhance the Hitachi-GE supplied data set (i.e. to extend the activity data from the 31 potentially significant radionuclides to the full suite of 112 potentially significant radionuclides), average package concentrations for other potentially significant radionuclides were estimated based on a scaling of the supplied data, using ratios between the additional radionuclide and a representative radionuclide. These ratios were available to RWM based on previous inventory enhancement work.

For cruds, the representative radionuclide used was Co-60 for radionuclides sourced from activated metal components; Cs-137 for soluble or gaseous fission products and Ce-144 for insoluble fission products (or Cs-137 if Ce-144 was not available for the comparator waste stream). For resins, the representative radionuclide was Ni-63.

The activity of the other significant radionuclides were estimated by multiplying the concentration of the representative radionuclide by the estimated ratio of the other radionuclide and the representative radionuclide. The estimated ratio was the maximum of ratios for similar waste streams from existing datasets for Sizewell B, EPR and AP1000 reactor wastes. In order to develop a conservative assessment inventory, appropriate for this stage of assessment, the maximum of the submitted specific activity and the scaled activities from the comparator waste streams was used in the assessment inventory.

The use of this scaling approach has significant consequences for the transport, operational and post-closure safety assessments that have applied the assessment inventories based on this scaling approach. For example, and most significantly, the values for I-129 have applied a scaling factor of approximately 1E-03, which is the maximum ratio between Ni-63 and I-129 for the three datasets listed above. There are significant differences in the ratios between the three datasets (the minimum ratio is approximately 1E-06), and the ratio in data supplied by Hitachi-GE was of the order 1E-09 to 1E-10. Therefore, the conclusions from the transport, operational and post-closure safety assessments have taken this uncertainty into account, and where necessary have undertaken additional calculations to determine the significance of other radionuclides. The use of waste-type specific operational experience and data to reduce this uncertainty would be considered at an appropriate time in the future through the Disposability Assessment process.

The maximum package activities for cruds and resins was estimated by multiplying the average package activities by 12. This is consistent with the ratio between average and maximum package activities for cruds and resins used in previous assessments. It also represents the approximate uncertainty in the scaling factors applied to estimate the average package activities. In future interactions under the Disposability Assessment process a method for calculation of the maximum package inventories will need to be proposed by the operator.

For resins, the dose is dominated by Cs-137. Radioactive caesium exchanged by the resins could cause self-irradiation of the resin material during the period that the resin is held in a storage tank. In addition, it is necessary to consider mechanical attrition and degradation during storage, and the possibility that retrieval could encounter degraded material. These issues should be addressed in future interactions under the Disposability Assessment process.

Control Rods and Activated Metals

Control rod assessment inventories were based on six-year cooled data; it is currently assumed that decommissioning will start six years after reactor shutdown. Activated metal assessment inventories were based on one year of cooling following reactor shutdown.

For control rods and activated metals, Hitachi-GE provided an inventory based on ORIGEN modelling of the control rod composition as a homogeneous body and a description of the control rod materials, including elemental compositions of the various steels used in the rods. Hafnium contained in the Hafnium Control Rods is assumed to be in metal form, in a structure referred to as the “flat tube”. The flat tubes are housed in a rod structure composed of GXM1 and SUS316L steel, with the latter alloy being dominant. GXM1 steel has a high composition by mass of manganese (4-6 wt%), which is thought to harden the steel surface. The Boron Carbide Control Rods contain boron carbide in powder form crimped inside neutron absorber tubes made of TP304L steel.

Although the control rods would be exposed to a variable neutron flux, the calculations assumed that the entire length of the rods was exposed to the maximum core flux. The supplied inventory was therefore considered to be conservative. No cobalt was included in the steel compositions provided by Hitachi-GE. Co-60 can be a significant contributor to dose during transport and operations, and, therefore, RWM enhanced the inventory by assuming that the control rod metals contained 0.26% cobalt. This is the typical concentration of cobalt in Type 304 stainless steel, for which RWM hold detailed precursor composition data. Extension of the supplied radionuclide inventory to the full list of relevant radionuclides was undertaken by scaling inventories using the declared Ni-63 activities.

Hf-178n is not modelled by ORIGEN. The maximum package activities for control rods and activated metals were based on wastes arising at the end of reactor operation with an additional scaling factor added for uncertainty. In the extended radionuclide inventory, Hf-178n is a significant contributor to dose at short timescales (the half-life of Hf-178n is approximately 31 years), and, therefore, should be included in future inventories for the UK ABWR supplied by Hitachi-GE.

The assumption that control rod metals contain 0.26% cobalt, leads to relatively high activities for Co-60 in the waste package inventories. The estimated activities can, in certain cases, challenge the limits on transport included in the 2012 IAEA Transport Regulations [22], and the assumptions in RWM’s operational safety case.

For some of the steels, Hitachi-GE had provided good information on the steel compositions, but for some steels no compositions had been provided. For the assessment, the composition of Type 304 steel had been used to fill gaps and to ensure that the inventory was pessimistic. The steel used in the Activated Metals is unlikely to be Type 304, and a more corrosion-resistant metal is likely to be used by reactor operator.

The neutron sources included within the Activated Metals waste stream does not include antimony and beryllium. These are common elements in modern neutron sources, but different sources may be used by the developer of an ABWR in the UK. Similarly, the monitoring probes may include fission chambers containing uranium and thermocouples used to monitor reactor temperature might use silver or indium. Although it is unlikely that the monitoring probes will contain significant quantities of uranium, any uranium within the probes should be reported in more detailed disposability assessments.

The Disposability Assessment has not considered removal of metal items from storage baskets, but it is feasible that the storage baskets will be packaged with Activated Metals. This would reduce the dose. The management of Activated Metals prior to packaging and the consequent impact on package inventories should be considered in more detailed interactions under the Disposability Assessment process.

In future, RWM would expect to work with Hitachi-GE to reduce pessimisms in the inventories for Control Rods and Activated Metals. This might include consideration of the steel alloys used in the UK ABWR, for example, consideration of low-cobalt steel for the hafnium control rods.

Decommissioning ILW

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.

For decommissioning ILW, Hitachi-GE provided an activation product inventory for 57 radionuclides. This inventory was based on ORIGEN activation calculations for a Japanese ABWR irradiated for 40 years at a 75% load factor, i.e. 30 Equivalent Full Power Years (EFPY). Analytical formulae were used to extend this inventory for the UK ABWR based on a conservative 60 EFPY irradiation assumption.

In order to check the Hitachi-GE activation data for Reactor Pressure Vessel Internals, independent activation calculations were undertaken by RWM using EASY2010, consisting of FISPACT2007 [24], equipped with the EAF2010 decay [25] and cross-section libraries [26]. These calculations required the development of detailed neutron flux data, as these data had not been supplied by Hitachi-GE. The detailed neutron flux data were generated using a two-step process:

- First, an approximate effective energy dependent flux was derived for three broad energy groups by calculating the neutron fluxes for each neutron energy range required to reproduce the total activity of certain key activation products. The broad energy ranges used and the associated nuclear reactions used to estimate neutron flux in each energy group are shown in Table 5. The effective fluxes are shown in Table 6. The effective fluxes were checked for credibility against previously calculated flux/fluence data for a BWR shroud [27] [28]; this comparison is also shown in Table 6.
- Second, the within group energy dependence for the three broad energy groups were specified based on information from a 74-group flux spectrum available to RWM from a project undertaken to identify the potentially relevant radionuclides from the perspective of geological disposal [29].

Table 5 Broad energy groups and associated nuclear reactions used to estimate neutron flux in each energy group

Energy Group	Energy Range	Activation Product	Key Production Reaction
Thermal	1E-05eV – 0.5eV	C14	N14(n, p)C14
		Fe55	Fe54(n, g)Fe55
		Ni59	Ni58(n, g)Ni59
		Ni63	Ni62(n, g)Ni63
Epithermal	0.5eV – 820keV	Tc99	Mo98(n, g)Mo99(beta+)Tc99*
Fast	820keV – 20MeV	Mn54	Fe54(n, p)Mn54

Table 6 Broad energy group flux data unfolded from Hitachi-GE activity data and similar data reported in the international literature

Energy Group	Flux Derived from Activity Unfolding Process ($n/cm^2/s$)	Flux Data Reported or Derived from data in the International Literature for a BWR Shroud ($n/cm^2/s$)	Basis of International Literature Value
Thermal	1.2E+13	2.5E+13	Core Mid Plane computed value reported in Table G-9 of [27]
Epithermal	3.2E+12	3.0E+12	Core Mid Plane computed value reported in Table G-9 of [27]
		5.6E+12	Derived from data in Table 4 of IAEA TECDOC-1471 [28] i.e. fluence in Energy range 0.1 to 1.0MeV = 0.9×10^{25} n/m^2Assuming a standard 1/E energy dependence for the flux from 0.5eV to 1MeV
Fast	2.0E+12	1.0E+12	Derived from data in Table 4 of IAEA TECDOC-1471 [28] i.e. fluence in Energy range >1.0MeV = 1.0×10^{25} n/m^2

The two inventories were in good agreement. For the radionuclides making a dominant contribution to total activity at 6-years cooling (Ni-63, Fe-55, Co-60 and Ni-59), agreement between Hitachi-GE data and the cross-check by RWM was better than a factor of two. Good agreement was also found for the important long-lived radionuclide C-14.

The final activation product inventory applied in the GDA Disposability Assessment for Reactor Pressure Vessel Internals used the Hitachi-GE radionuclide activities when these exceeded 50% of the RWM activities, and the RWM activities when this was not the case.

In addition to the activation product inventory, the GDA Disposability Assessment has also considered the inventory associated with contamination of the reactor pressure vessel internals. This inventory is trivial compared to the activation inventory (typically 3-5 orders of magnitude lower), and, therefore, no independent check of the contamination inventory was made.

Independent inventory calculations were not undertaken for the reactor pressure vessel materials, because:

- the check on the reactor pressure vessel internals demonstrated a reliable prediction of the dominant radionuclides;
- the steel precursors used by Hitachi-GE were judged to be appropriate; and
- the specific activities of the RPV steels are 5-6 orders of magnitude lower than the reactor pressure vessel internals.

However, it was necessary to enhance the inventory with activities for U-234, U-235, U-238 and Th-232, as uranium and thorium are present as trace element concentrations in steel, and activities for U-234, U-235, U-238 and Th-232 had not been declared in Hitachi-GE's activation inventory. As little of this material would have been subject to activation owing to the low neutron fluxes experienced by the pressure vessel, enhancement was undertaken by calculating the masses for each radionuclide and converting these to specific activities.

For the Reactor Pressure Vessel, a contamination inventory was added to the activation inventory, as no contamination inventory was included in the submission. The Reactor Pressure Vessel Internals contamination inventory data were used as a basis for this enhancement. This Reactor Pressure Vessel contamination inventory focused on Fe-55, Ni-63 and Co-60, which comprise approximately 97% of the Reactor Pressure Vessel Internals contamination inventory at reactor shutdown. For these radionuclides, Hitachi-GE provided surface contamination data for the Reactor Pressure Vessel Internals in terms of Bq/cm^2. These data were used, together with an estimate of the surface area of the liner to derive the contamination inventory for Fe-55, Ni-63 and Co-60 for the Reactor Pressure Vessel. Fe-55, Ni-63 and Co-60 constitute approximately 97% of the total contamination at reactor shutdown. In addition, an inventory for Ni-59 was estimated based on a ratio of Ni-59 to Ni-63 of 0.01, which is the recognised ratio of these radionuclides in response to activation of elemental nickel.

The UK ABWR inventory of C-14 is dominated by the reactor internals wastestream, with 599TBq. C-14 is important in the waste disposal inventory as it has a relatively long half-life (circa 5,730 years) and has the potential to migrate to the biosphere as a gas phase. RWM is currently studying the wastestream sources of C-14 to better understand its behaviour, chemical forms and speciation. The main source of C-14 in the steels is from nitrogen. RWM may investigate whether new-build designers could specify low nitrogen steels for those components that do not require the nitrogen for specific required properties.

The C-14 decommissioning wastes total activity for UK ABWR at 599TBq is of the same order as the figures derived in previous GDA Disposability Assessments, which were in the range 190TBq to 950TBq.

3.3.4 Comparison of UK ABWR ILW with Sizewell B ILW

In order to place the information on ILW from a UK ABWR in context, a comparison has been made with ILW from Sizewell B, which is a light water reactor (in this case a pressurised water reactor rather than a boiling water reactor) operated in the UK by Electricité de France (EdF), (Table 7).

The comparison was made for the most active ILW stream in the UK ABWR inventory, as low-activity waste streams are likely to have less influence on the overall conclusions on disposability from the assessment. The waste streams compared were:

- UK ABWR09: RPV stainless steel internals for the UK ABWR.
- 3S306: decommissioning stainless steel ILW for Sizewell B.

The total activity of these streams were compared against the total activity of the other waste streams for these reactors to ensure that the highest activity streams have been chosen. In the case of the Sizewell B waste streams, 3S306 has an activity that is at least an order of magnitude greater than any other Sizewell B ILW waste stream at 2075 (40 years after reactor shutdown). For the UK ABWR, the stainless steel internals are several orders of magnitude more active than the RPV after 40 years of cooling, and at least an order of magnitude more active than any of the operational ILW waste streams.

The activity of UK ABWR RPV stainless steel internals (stream UKABWR09) is compared with the activity of 3S306 in Table 7. The basis for Table 7 is as follows:

- radionuclide activities have been estimated for 40 years after reactor shutdown;

- the activity data have been normalised to the total thermal power output of the two reactors (Sizewell B – 3478 MW (thermal) for 40 years, UK ABWR 3926 MW (thermal) for 60 years);
- the radionuclides considered in Table 7 are the top 10 most active in the UK ABWR wastes for which estimates were also available for the Sizewell B PWR wastes; and
- the cell colouration displayed in the sixth column of Table 7 is used to indicate the closeness of the agreement that presents the ratio of UK ABWR to Sizewell B normalised activities as follows: green <10, orange >10.

As can be seen in Table 7, with the exception of Mo-93 and Tc-99, the activities of all of the radionuclides are similar and within a factor of 10. The presence of Mo-93 and Tc-99 in the UK ABWR RPV stainless steel internals will be the result of activation of molybdenum present in the steel alloys, and, therefore, the difference in the UK ABWR and Sizewell B activities for these two radionuclides is thought to be the result of differences in the assumed trace concentrations of molybdenum and the differences in the neutron fluxes used to determine the activation products. RWM has applied conservative upper bound trace element concentrations in the inventory enhancement work.

In addition, EdF has quoted a factor of 1,000 uncertainty on Decommissioning Stainless Steel ILW for Sizewell B in their submission for the 2013 UK RWI [30]. Therefore, the factor of approximately 40 difference between the estimated activities for Mo-93 and Tc-99 in UK ABWR and Sizewell B wastes is considered insignificant and the agreement between the radionuclide inventories is considered to be good.

Table 7 Comparison of radionuclide activities for Reactor Pressure Vessel Internals from a UK ABWR with Equivalent ILW stream from Sizewell B PWR (3S306)

Nuclide	UK ABWR (UKABWR09) (TBq)	Sizewell B (3S306) (TBq)	UK ABWR (UKABWR09) (TBq per MW (thermal).yr)	Sizewell B (3S306) (TBq per MW (thermal).yr)	(UKABWR09) / (3S306)
Ni63	3.66E+05	3.71E+04	1.55E+03	2.67E+02	5.83E+00
Ni59	3.57E+03	3.23E+02	1.52E+01	2.32E+00	6.53E+00
Co60	3.05E+03	8.06E+02	1.29E+01	5.79E+00	2.24E+00
C14	5.29E+02	3.59E+01	2.25E+00	2.58E-01	8.71E+00
H3	9.11E+01	8.77E+01	3.87E-01	6.30E-01	6.14E-01
Fe55	8.95E+01	1.79E+02	3.80E-01	1.29E+00	2.95E-01
Mo93	8.32E+01	1.21E+00	3.53E-01	8.66E-03	4.08E+01
Nb93m	7.39E+01	3.77E+02	3.14E-01	2.71E+00	1.16E-01
Tc99	7.95E+00	1.21E-01	3.37E-02	8.72E-04	3.87E+01
Nb94	2.19E+00	4.04E+00	9.28E-03	2.90E-02	3.20E-01

The practices used in operating a UK ABWR 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 a UK ABWR 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 UK ABWR 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 UK ABWR waste streams is comparable with that for Sizewell B.

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

3.4.1 Description of Spent Fuel

The reactor core of a UK ABWR is comprised of fuel assemblies, control rods and nuclear instrumentation. Control rods and nuclear instrumentation will be managed as ILW, and are discussed above in Section 3.3. There are 872 fuel assemblies in the reactor core during operation. The fuel assembly consists of a fuel bundle and an interactive fuel channel (Figure 7). The fuel bundle contains the fuel rods and the hardware necessary to support and maintain the proper spacing between the fuel rods. The channel is a Zircaloy-2 box, which surrounds the fuel, and is used to direct the vertical core coolant flow through the bundle. It also provides a surface to guide the control rods as they are inserted.

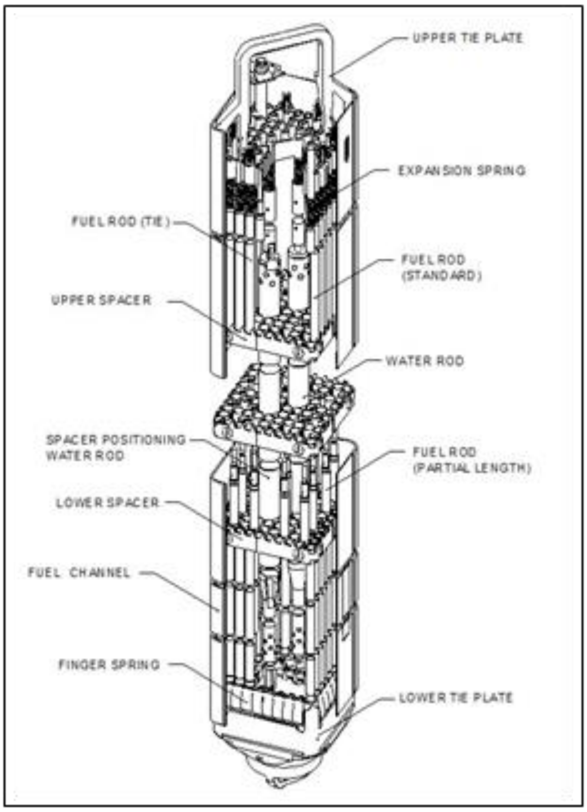
The UK ABWR is expected to use the GE14 type of fuel assembly. This type of fuel assembly is already in use in BWRs in many countries, including Sweden and Finland. In GE14, each fuel assembly is formed by a 10x10 array of 78 full-length fuel rods, 14 part-length rods which span roughly two-thirds of the active core, and two large central water rods. The fuel bundle assembly is held together by eight of the full-length rods located around the periphery; these are referred to as tie rods. The assembly is referred to as the '10x10-8' assembly because the water rods replace eight of the central fuel rods in the array.

The fuel rods consist of seal-welded Zircaloy-2 cladding tubes and end plugs, containing either UO_2 or $(U, Gd)O_2$ pellets. A United States Nuclear Regulatory Commission (USNRC) document describing the GE14 fuel assembly notes that the fuel pin is filled with helium at 44psi to improve heat transfer [31]. There is also a ~0.25m-long free volume known as the plenum region in the top of each pin. This region is designed to collect and retain volatile fission products that escape from the fuel pellets. The plenum region contains a plenum spring to axially compress the stack of fuel pellets so that they are firmly seated in the fuel rod.

As shown in Figure 7, the height of the fuel pellet stack, i.e. the active height of the fuel assembly, in the full-length and part-length rods is 3.81m and 2.13m respectively. Other dimensional information is provided in Table 8 and mass information is provided in Table 9. The fuel pellets are enriched relative to the concentration of the fissile isotope, U-235, in natural uranium (0.72 wt%). The average bundle enrichments and batch sizes used in each fuel cycle are a function of the desired cycle length.

The initial ABWR core has an average enrichment ranging from approximately 1.7wt% U-235 to approximately 3.2wt% U-235 for cycle lengths ranging from one to two years. For ABWR reload cores using GE14 fuel, the average bundle enrichment is roughly 4.2wt% U-235 with a reload batch fraction of 35% for a two-year cycle (i.e. 35% of the fuel is reloaded into the core after each cycle) [32].

Figure 7 Illustration of a GE14 fuel assembly, the fuel assembly expected to be used in a UK ABWR; the diagram on the left shows the components of the fuel assembly and the diagram on the right shows the full-length and part-length fuel rods



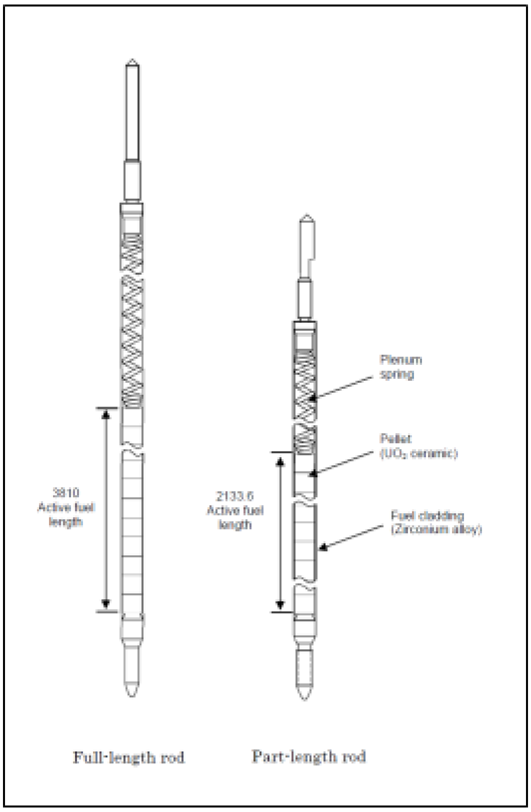


Table 8 Dimensional information for UK ABWR fuel assemblies and rods

Fuel Assembly	
External maximum section (mm x mm)	140.16 x 140.16
Maximum length (mm)	4468
Active length (mm)	3810
Overall mass (kg)	298
Uranium oxide mass (kg)	204
Fuel Rod	
Number of fuel rods	92
Fuel rod outer diameter (mm)	10.26
Cladding thickness (mm)	0.660
Pin pitch (mm)	12.95

Table 9 Estimates of component mass for a UK ABWR fuel assembly

Component of fuel assembly	Material	Mass per assembly (kg)
UO_2	UO_2	204
Gd neutron poison material *	Gd	1.3
Channel box	Zircaloy-2	85.0
Fuel rod cladding	Zircaloy-2	
Water rods	Zircaloy-2	
Spacers	Zircaloy-2	
Zircaloy minor components	Zircaloy-2	
Lower tieplate	Stainless steel	6.7
Upper tieplate	Stainless steel	
Stainless steel minor components	Stainless steel	
Expansion springs	Inconel X-750	0.4
Spacer springs	Inconel X-750	
Inconel X-750 minor components	Inconel X-750	
Additional unspecified material	Unknown	0.5
Total		**298**

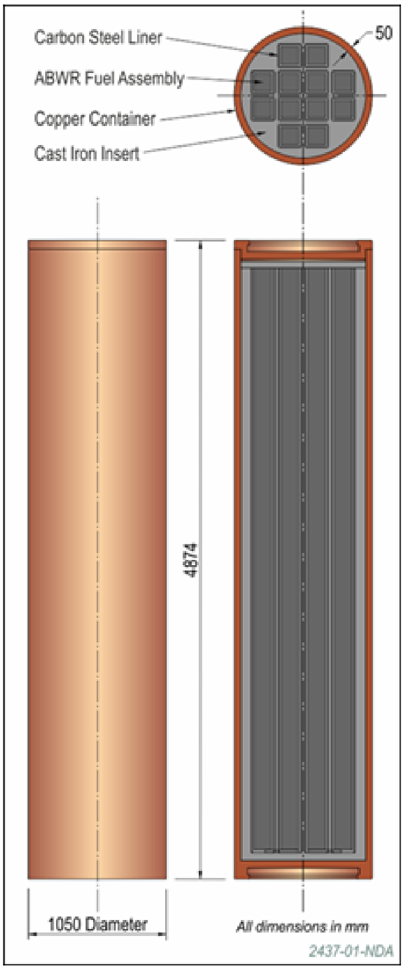
* A small fraction of UO_2 pellets contain the burnable neutron poison, Gd_2O_3

3.4.2 Spent Fuel Packaging Assumptions

As discussed in Section 2, the packaging assumptions for UK ABWR spent fuel are based on concepts developed by RWM to date [8]. Under these concepts, spent fuel would be over-packed into durable, corrosion-resistant disposal containers manufactured from suitable materials, which would provide long-term containment for the radionuclides contained within the spent fuel (Figure 8). Although the container material remains to be confirmed, the Disposability Assessment process considers the potential performance of both copper and carbon steel containers. In the copper container case, it is assumed that a cast-iron insert is used to hold and locate the spent fuel assemblies, and to provide mechanical strength. In the carbon steel container case, a carbon steel “tube and plate” basket is used to hold and locate the spent fuel assemblies.

Based on plans for packaging BWR spent fuel in Finland and Sweden [33, 34], this GDA Disposability Assessment has assumed that twelve UK ABWR spent fuel assemblies would be packaged in each disposal container. The disposal container would have a length of 4.874m and a diameter of 1.050m; the diameter of the containers would be identical to the diameter of containers used to dispose of other types of spent fuel in a UK geological disposal facility [31]. The displacement volume of the container is $4.13m^3$. As discussed in Section 2, for the higher strength rock illustrative design, which is considered to be the bounding case, these containers would be emplaced in deposition holes lined with a buffer made from compacted bentonite, which swells following contact with water (Figure 2).

Figure 8 Illustration of a UK ABWR spent fuel disposal container, assumed for the higher strength rock illustrative design



It is assumed that transport of packaged spent fuel would be undertaken using a preliminary RWM design for a Disposal Container Transport Container (DCTC) which provides two layers of shielding material:

- Immediately adjacent to the container is a stainless steel gamma shield with thicknesses of 140mm in the radial direction and 50mm at the ends of the container.
- Surrounding the stainless steel gamma shield is a 50-mm-thick neutron shield made of high neutron capture material.

Although the quantitative analyses conducted in the GDA Disposability Assessment for the UK ABWR are based on certain disposal concept assumptions, the implications of alternative disposal concepts have also been considered, as discussed in Section 5.

3.4.3 Spent Fuel Package Numbers and Characteristics

Hitachi-GE has estimated that a UK ABWR would use 9,600 off 50GWd/tU spent fuel assemblies over its 60-year operational life. This number of fuel assemblies was confirmed by RWM by consideration of the design thermal output of the UK ABWR.

The design thermal output of a UK ABWR is 3.962GW. If such an output was sustained for the whole 60 years of reactor operation without any shutdowns the total energy generated would be 8.60E+04GWdays. However, few if any modern light-water reactors achieve load factors greater than 90%, so the realistic maximum thermal energy generation from a UK ABWR is 7.74E+04GWdays.

For a burn-up of 50GWd/tU and 9,600 fuel assemblies, the total thermal energy generated would be 8.63E+04GWdays. This is in excess of the theoretical maximum based on the design thermal output, and, therefore, RWM's checking confirmed that the number of fuel assemblies proposed by Hitachi-GE was conservative.

For a packing assumption of 12 spent fuel assemblies per disposal container, 9,600 fuel assemblies corresponds to 800 disposal containers. Assuming that the lifetime heat energy coming from the reactor would be the same for the 60 GWd/tU case, 8,000 spent fuel assemblies would be discharged, which would require 667 disposal containers. Given the fuel assembly dimensions shown in Table 8, the raw waste volume associated with these numbers of fuel assemblies is 841m^3 for a burn-up of 50GWd/tU and 701m^3 for a burn-up of 60GWd/tU. Packaged volumes would be 3,304m^3 and 2,755m^3 respectively.

Hitachi-GE provided one-year-cooled ABWR fuel assemblies for the 50 and 60 GWd/tU burn-up cases. These calculations were made with ORIGEN v2.2, using nuclear data libraries used for licensing in Japan. The reactor physics model used to generate the nuclear data libraries was based on a 9x9 fuel rod assembly. This assembly is considered to be a reasonable approximation to the GE14 '10x10-8' assembly, because the moderator to fuel ratio, and hence the neutron spectrum, should be similar in the two types of assembly.

These inventories were based on the fuel assembly compositions provided in Table 10. These fuel assembly compositions assume chlorine impurities in fuel based on data provided in [35], and chlorine concentrations in Zircaloy and stainless steel based on the Nirex Cl-36 Project [36].

To check that the inventories supplied by Hitachi-GE were conservative, RWM generated independent inventories using the ORIGEN-ARP inventory calculation tool and used these inventories to calculate heat output from the fuel assemblies. The independent calculations used an available cross-section library for the GE14 fuel assembly. The calculations in ORIGEN use 2D neutron transport calculations, and, therefore, the 3D nature of the UK ABWR fuel assembly (e.g. the presence of the part-length fuel rods and the difference in the physical form of the coolant which is converted to steam as it traverses the fuel rod) was accounted for in a representative manner.

In RWM's calculations, a value for the fuel assembly steel mass of 10% of the true assembly steel mass has been used. Such an approach simulates the fact that all the steel is located in an area where the neutron flux is only around 10% of that experienced by the UO_2. RWM's calculations included U-234, whose concentration was obtained from ORIGEN-ARPs automatic fuel composition routine. The inclusion of U-234 makes the calculations more realistic.

The composition was based on the data in the ASTM standard for UO_2 [37] supplemented by the data from the Oak Ridge Laboratory [38]. The Oak Ridge data was also used to derive the nuclear data library used by Hitachi-GE.

Table 10 Starting composition data for the spent fuel inventory calculations

Element/Isotope	Mass (g/tU)
[redacted]	[redacted]
[redacted]	[redacted]

Element/Isotope	Mass (g/tU)
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
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[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]
[redacted]	[redacted]

The independent calculations conducted by RWM demonstrated that the Hitachi-GE inventories were conservative. For the 50 GWd/tU case, the heat output derived from the Hitachi-GE inventory exceeds that from the ORIGEN-ARP calculations by no more than 5.6% in the cooling time range 1-100 years. At longer cooling times (up to 260 years), the difference in the heat output between the Hitachi-GE and ORIGEN-ARP calculations increases, but still does not exceed 10.3%. The comparison for the 60 GWd/tU case is similar, although the differences in the heat output are slightly larger. At 100 years cooling, temperatures generated from the Hitachi-GE inventory are 6.2% higher than the ORIGEN-ARP inventory data, with the difference rising to 11.1% at 260 years cooling.

3.4.4 Comparison of UK ABWR Spent Fuel with Sizewell B PWR Spent Fuel

In order to place the information on spent fuel from a UK ABWR in context, RWM has assembled comparisons of the radionuclide inventories for the most significant post-closure radionuclides in spent fuel from a UK ABWR with radionuclide inventories for spent fuel from PWR (Sizewell B). Two comparisons are made:

- the comparison in Table 11 is based on the inventory of radionuclides estimated to be present in per tonne of uranium present in fresh fuel; and
- the comparison in Table 12 is based on the inventory of radionuclides estimated to be present in a single disposal container.

The two comparisons provide different perspectives on the radionuclide inventories of UK ABWR spent fuel compared to Sizewell B spent fuel. The comparison of radionuclide inventories highlights any fundamental differences that may arise owing to, for example, the type of reactor or operating regime. The comparison of container inventories illustrates any differences that result in different packaging approaches, i.e. the quantity of spent fuel packaged in each disposal container.

The comparisons are based on two different burn-ups for each type of reactor:

- ‘Average Burn-up’ Activities: Average burn-up activities are based on the 50 GWd/tU UK ABWR inventory. The Sizewell B data are for stocks, which are assumed to have experienced a burn-up of 45 GWd/tU and an initial enrichment of 4.2%. The Sizewell B average burn-up data are modelled with 8 years cooling.
- High Burn-up Activities: High burn-up activities are based on the 60 GWd/tU UK ABWR inventory. The Sizewell B data are for planned fuel arisings, which are assumed to have experienced a burn-up of 55 GWd/tU and an initial enrichment of 4.4%. The Sizewell B high burn-up data are modelled with 1 years cooling.

There is assumed to be 0.1798 tU per ABWR fuel assembly and, assuming that 12 are incorporated in a single disposal container, this corresponds to 2.158 tU per disposal container. In contrast, there is assumed to be 1.834 tU in a Sizewell B disposal container, based on 4 PWR fuel assemblies.

These comparisons demonstrate that the radionuclide inventories for the UK ABWR and Sizewell B are very similar. For the comparison of total activities per disposal container, only two radionuclides, Cl-36 and Sn-126, have activities in UK ABWR spent fuel greater than twice the activities in Sizewell B spent fuel, and only two radionuclides, Ni-59 and U-233 have activities in UK ABWR less than half of the activities in Sizewell B spent fuel. These radionuclides are highlighted in Tables 11 and 12 in orange (more than twice) and green (less than half) respectively.

The activities of Cl-36 are approximately three times higher in the estimated inventory for the UK ABWR used in this Disposability Assessment compared to the inventory for Sizewell B spent fuel in the 2013 Derived Inventory [39]. This is due to the differences in the assumed contamination of the fresh spent fuel with chlorine; the UK ABWR inventory was based on 25.5g/tU of chlorine per fuel assembly, whereas the Sizewell B inventory was based on 6.12g/tU of chlorine per fuel assembly.

The activities of Sn-126 are approximately three times higher in the estimated inventory for the UK ABWR used in this Disposability Assessment compared to the inventory for Sizewell B spent fuel in the 2013 Derived Inventory [39]. This is due to the differences in the assumed precursor concentration of tin in the fresh spent fuel (including cladding); the UK ABWR inventory was based on 8,030g/tU of tin per fuel assembly, whereas the Sizewell B inventory was based on 4,440g/tU.

The activities of Ni-59 are approximately five times lower in the estimated inventory for the UK ABWR used in this Disposability Assessment compared to the inventory for Sizewell B spent fuel in the 2013 Derived Inventory [39]. This is due to the lower quantities of Inconel, and hence nickel precursor, assumed to be used in a UK ABWR fuel assembly when compared with a Sizewell B fuel assembly; the UK ABWR inventory was based on 2,790g/tU of nickel per fuel assembly, whereas the Sizewell B inventory was based on 9,210g/tU.

The activities of U-233 are approximately thirty times lower in the estimated inventory for the UK ABWR used in this Disposability Assessment when compared with the inventory for Sizewell B spent fuel in the 2013 Derived Inventory [39]. This is predominantly owing to the differences in the assumed precursor concentration of thorium in the fresh spent fuel; the UK ABWR inventory assumes a concentration of 10ppm of thorium in Zircaloy-2 resulting in 10g/tU of thorium per fuel assembly, whereas the Sizewell B inventory was based on 282g/tU, derived from the Earth's crustal abundance method, which assumes a minimum concentration of 1,000ppm.

Table 11 Comparison of radionuclide activities for spent fuel from an UK ABWR with spent fuel from Sizewell B; total activities are presented per tonne of uranium in the fresh fuel (key radionuclides only)

	Average Burn-up SF Activity (TBq)			High Burn-up SF Activity (TBq)		
Nuclide	**ABWR**	**SZB**	**Ratio**	**ABWR**	**SZB**	**Ratio**
C-14	1.17E-01	2.14E-01	0.55	1.48E-01	2.13E-01	0.69
Cl-36	3.15E-03	9.94E-04	3.17	3.89E-03	9.94E-04	3.92
Ni-59	5.96E-02	2.91E-01	0.21	7.13E-02	2.91E-01	0.25
Se-79	1.93E-03	3.17E-03	0.61	2.20E-03	3.80E-03	0.58
Sr-90	9.10E+02	8.55E+02	1.06	6.75E+02	6.52E+02	1.04
Tc-99	7.11E-01	6.60E-01	1.08	8.14E-01	7.78E-01	1.05
Sn-126	3.82E-02	1.16E-02	3.30	4.83E-02	1.46E-02	3.30
I-129	1.78E-03	1.35E-03	1.31	2.15E-03	1.68E-03	1.28
Cs-135	2.86E-02	2.66E-02	1.08	3.39E-02	3.19E-02	1.06
Cs-137	1.43E+03	1.30E+03	1.10	1.15E+03	1.06E+03	1.09
U-233	1.43E-04	4.39E-03	0.03	1.49E-04	4.40E-03	0.03
U-234	6.74E-02	7.19E-02	0.94	8.05E-02	8.75E-02	0.92
U-235	4.47E-04	6.32E-04	0.71	2.52E-04	4.55E-04	0.55
U-236	1.30E-02	1.33E-02	0.98	1.33E-02	1.45E-02	0.92
U-238	1.15E-02	1.15E-02	1.00	1.14E-02	1.14E-02	1.00
Np-237	1.94E-02	1.97E-02	0.99	2.38E-02	2.51E-02	0.95
Pu-238	1.24E+02	1.18E+02	1.05	1.55E+02	1.47E+02	1.05
Pu-239	1.17E+01	1.37E+01	0.85	1.14E+01	1.41E+01	0.81
Pu-240	2.40E+01	2.20E+01	1.10	2.71E+01	2.53E+01	1.07
Pu-241	3.03E+02	3.20E+02	0.95	1.44E+02	1.57E+02	0.92
Pu-242	1.28E-01	1.10E-01	1.16	1.84E-01	1.55E-01	1.19
Am-241	1.76E+02	1.83E+02	0.96	1.90E+02	2.10E+02	0.90
Am-242m	2.80E-01	3.32E-01	0.85	2.82E-01	3.71E-01	0.76
Am-243	1.47E+00	1.55E+00	0.95	2.45E+00	2.45E+00	1.00

Table 12 Comparison of radionuclide activities for spent fuel from an UK ABWR with spent fuel from Sizewell B; total activities are presented per disposal container (key radionuclides only)

	Average Burn-up SF Activity (TBq)			High Burn-up SF Activity (TBq)		
Nuclide	**ABWR**	**SZB**	**Ratio**	**ABWR**	**SZB**	**Ratio**
C-14	2.53E-01	3.92E-01	0.64	3.19E-01	3.91E-01	0.81
Cl-36	6.79E-03	1.82E-03	3.73	8.40E-03	1.82E-03	4.61
Ni-59	1.29E-01	5.33E-01	0.24	1.54E-01	5.33E-01	0.29
Se-79	4.16E-03	5.82E-03	0.72	4.75E-03	6.96E-03	0.68
Sr-90	1.96E+03	1.57E+03	1.25	1.46E+03	1.20E+03	1.22
Tc-99	1.53E+00	1.21E+00	1.27	1.76E+00	1.43E+00	1.23
Sn-126	8.24E-02	2.12E-02	3.89	1.04E-01	2.68E-02	3.89
I-129	3.84E-03	2.48E-03	1.55	4.65E-03	3.08E-03	1.51
Cs-135	6.18E-02	4.88E-02	1.27	7.31E-02	5.85E-02	1.25
Cs-137	3.09E+03	2.39E+03	1.29	2.48E+03	1.94E+03	1.28
U-233	3.08E-04	8.06E-03	0.04	3.22E-04	8.06E-03	0.04
U-234	1.45E-01	1.32E-01	1.10	1.74E-01	1.61E-01	1.08
U-235	9.65E-04	1.16E-03	0.83	5.44E-04	8.35E-04	0.65
U-236	2.81E-02	2.44E-02	1.15	2.87E-02	2.66E-02	1.08
U-238	2.49E-02	2.11E-02	1.18	2.46E-02	2.09E-02	1.18
Np-237	4.19E-02	3.61E-02	1.16	5.14E-02	4.61E-02	1.12
Pu-238	2.68E+02	2.17E+02	1.24	3.35E+02	2.70E+02	1.24
Pu-239	2.52E+01	2.52E+01	1.00	2.46E+01	2.58E+01	0.95
Pu-240	5.19E+01	4.03E+01	1.29	5.85E+01	4.64E+01	1.26
Pu-241	6.55E+02	5.88E+02	1.11	3.10E+02	2.88E+02	1.08
Pu-242	2.77E-01	2.03E-01	1.37	3.97E-01	2.84E-01	1.40
Am-241	3.79E+02	3.36E+02	1.13	4.09E+02	3.85E+02	1.06
Am-242m	6.05E-01	6.08E-01	0.99	6.09E-01	6.80E-01	0.89
Am-243	3.17E+00	2.84E+00	1.12	5.30E+00	4.49E+00	1.18

4 ASSESSMENT OF UK ABWR OPERATIONAL AND DECOMMISSIONING ILW

In this section, we discuss the assessment of Hitachi-GE's packaging proposals for ILW against RWM's waste package specification [15] and disposal system specification [12], which were summarised in Section 2.1. The approach used follows that described in Section 2.2.2. The assessment is reported in four sections:

- Section 4.1 describes the assessment of the packages proposed by Hitachi-GE, including consideration of the characteristics of the proposed waste containers (Section 4.1.1) and wasteforms (Section 4.1.2), and a description of the methodology used to determine waste package performance (Section 4.1.3);
- Section 4.2 describes consideration of the impact of Hitachi-GE'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 a geological disposal facility – transport safety (Section 4.2.2), safety during the receipt, handling and emplacement of waste in a geological disposal facility – 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 Hitachi-GE's waste packaging proposals on long-term safety following closure of a geological disposal facility;
- Section 4.4 provides a statement regarding the overall disposability of ILW from a UK ABWR and identifies the basis for this statement.

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

4.1 Container and Wasteform Characteristics and Waste Package Accident Performance

The Level 2 Generic Specification for LHGW [15] is the primary means by which RWM defines the required characteristics and key features of ILW packages, taking into account the needs for the safe and efficient transport of waste packages and their disposal in a geological disposal facility. The requirements are, in general, defined for the complete waste package and linked to the requirements of transport, operational and post-closure safety, but, in practice, the manner in which they are achieved will depend on a number of factors including:

- the nature of the waste container;
- the physical, chemical and radiological properties of the waste; and
- the means by which the waste is conditioned for disposal.

Accordingly, in the L2 GS for LHGW, the requirements are grouped to reflect those which are most directly related to the waste container, the wasteform, and the waste package as a whole. Requirements on the waste package are considered as part of the accident performance, and transport, operational and post-closure safety assessments and are therefore addressed in Sections 4.2 and 4.3. In this section, the extent to which the proposals submitted by Hitachi-GE meet the requirements on containers and wasteforms are considered. In addition, the approach taken to determining the waste package accident performance is described in Section 4.1.3.

4.1.1 Waste Container Characteristics

Context

The Level 2 Generic Specification for LHGW specifies the following characteristics for waste containers:

- *External dimensions*:
    - The external dimensions of the waste package shall be compatible with the transport and geological disposal facility handling systems.
    - The overall dimensions of a transport package should not exceed 6.058m x 2.438m plan x 2.591m high.
    - The dimensions of a transport package carried by rail shall not exceed 2.67m wide or 2.40m high.
- *Handling feature*:
    - The waste package shall enable safe handling by way of the transport and geological disposal facility handling systems.
    - The waste package shall incorporate handling features to enable lifting under a load equivalent to twice the maximum specified gross mass without any effect that would render it non-compliant with any of the requirements defined in the L2 GS for ILW.
    - Where tie down within a conveyance is necessary for their safe transport, waste packages which are transport packages in their own right shall incorporate tie-down features suitable for their maximum specified gross mass.
    - The design of the waste package should enable remote handling.
- *Stackability*:
    - Where required by the transport or disposal system, the waste package shall enable safe stacking.
    - Waste packages which rely on their design to withstand stacking loads should be capable of being stacked to a height of 11m with other waste packages of the same design, each with their maximum specified gross mass. This loading shall not result in any effect that could render the waste package non-compliant with any of the requirements defined in the Level 2 Generic Specification for LHGW.
    - Waste packages which are transport packages in their own right shall comply with the stacking requirements defined by the 2012 IAEA Transport Regulations [22].
- *Identification*:
    - The waste package shall enable unique identification until the end of the geological disposal facility operational period.
    - The waste package shall be marked at multiple defined locations with a unique alpha-numeric identifier.
    - The waste package shall remain identifiable by automated systems for a minimum period of 150 years following manufacture.
- *Durability of waste container integrity*:
    - The waste package shall enable safe handling by way of its handling feature until the end of the geological disposal facility operational period.

- The waste container shall maintain containment for as long as is required by the geological disposal facility safety case.
- The integrity of the waste container should be maintained for a period of 500 years following manufacture of the waste package.

Results and Implications

The criteria on waste containers specified in the L2 GS for ILW have been used as a check-list for the review of waste containers proposed in the Hitachi-GE submission. The results of the evaluation are provided in Table 13, and the most significant points discussed below.

The $3m^3$ Box, $3m^3$ Drum and 4-metre Box packaging options proposed by Hitachi-GE in consultation with RWM for UK ABWR ILW are standard containers. The case for compliance with the waste container criteria defined in the Level 2 Generic Specification for LHGW should be readily made and are unlikely to raise any waste container compatibility issues. RWM will, however, wish to assess specific designs in future Disposability Assessment stages to confirm that the container criteria will be met.

Table 13 Check-list criteria for the different waste containers proposed by Hitachi-GE for the packaging of ILW

Waste Container	Operational Waste		Decommissioning Waste	
	$3m^3$ Box	**$3m^3$ Drum**	**$3m^3$ Box**	**4-metre Box**
External dimensions	Dimensions are compatible, as defined in Waste Package Specification (WPS) 310 or WPS 315	Dimensions are compatible, as defined in Waste Package Specification WPS 320	Dimensions are compatible, as defined in Waste Package Specification (WPS) 310 or WPS 315	Dimensions are compatible, as defined in Waste Package Specification WPS 330
Handling feature	The $3m^3$ Box provides handling using twistlock fittings on the top face of the container	The $3m^3$ Drum provides handling using twistlock fittings on the top face of the container	The $3m^3$ Box provides handling using twistlock fittings on the top face of the container	The 4-metre Box provides handling using twistlock fittings on the top face of the container
Stackability	The design of the $3m^3$ Box includes integral stacking posts, designed for 7-high stacking with similar packages, each at 12 tonnes gross mass (72 tonne compressive load)	The $3m^3$ Drum is designed for 7-high stacking with similar packages, each at 8 tonne gross mass (48 tonne compressive stack load)	The design of the $3m^3$ Box includes integral stacking posts, designed for 7-high stacking with similar packages, each at 12 tonnes gross mass (72 tonne compressive load)	The 4-metre Box is designed for 6-high stacking with similar packages, each at 65 tonne gross mass (325 tonne compressive stack load)
Identification	The design of the $3m^3$ Box includes an alpha-numeric identifier in machine readable format in four positions on the box body	The design of the $3m^3$ Drum includes an alpha-numeric identifier in machine readable format in four positions on the box body	The design of the $3m^3$ Box includes an alpha-numeric identifier in machine readable format in four positions on the box body	The design of the 4-metre Box includes an alpha-numeric identifier in machine readable format in four positions on the box body

Waste Container	Operational Waste		Decommissioning Waste	
	$3m^3$ Box	$3m^3$ Drum	$3m^3$ Box	4-metre Box
Durability of waste container integrity	The design of the $3m^3$ Box assumes that the container is comprised of stainless steel, which, provided suitable conditions during storage, would be designed to last for the appropriate period	The design of the $3m^3$ Drum assumes that the container is comprised of stainless steel, which, provided suitable conditions during storage, would be designed to last for the appropriate period	The design of the $3m^3$ Box assumes that the container is comprised of stainless steel, which, provided suitable conditions during storage, would be designed to last for the appropriate period	The design of the 4-metre Box assumes that the container is comprised of stainless steel, which, provided suitable conditions during storage, would be designed to last for the appropriate period

4.1.2 Wasteform Characteristics

Context

The Level 2 Generic Specification for LHGW requires the properties of the wasteform to be such that, in conjunction with those of the waste container, it satisfies all requirements for the waste package. In addition, the properties of the wasteform shall comply with the requirements for containment within a geological disposal concept, as defined by a geological disposal facility safety case. The physical, chemical, biological and radiological properties of the wasteform shall make an appropriate contribution to the performance of the waste package (including having no deleterious effect on the performance of the waste container).

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 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, and chemical reactivity should be minimised through conditioning.
- *Consideration of, and, if necessary, controls on, the presence of*:
  - free liquids;
  - activity or hazardous materials in particulate form;
  - voidage;
  - in-homogeneity;
  - reactive materials;

- other hazardous materials, e.g. flammable, explosive, pyrophoric, chemotoxic and oxidising materials, sealed containers and objects containing stored energy; and
- materials that could have a deleterious effect on the other barriers that make up a geological disposal system.

Further requirements concern the evolution of the wasteform. The Level 2 Generic Specification for LHGW states that evolution of the wasteform shall ensure maintenance of the waste package properties that are necessary for safe transport and operations at a geological disposal facility, and the required safety functions for post-closure performance as set out in the Environmental Safety Case (ESC). The DSTS defines a single post-closure safety function for wasteforms, requiring them to "*provide a stable, low-solubility matrix that limits the rate of release of the majority of radionuclides by dissolution in groundwater that comes into contact with the wasteform"*.

Results and Implications

The criteria on wasteforms described above have been used as a check-list for the review of the proposed wasteforms. The results of the evaluation are provided in Table 14 (operational ILW) and Table 15 (decommissioning ILW). Any features of the wasteforms proposed by Hitachi-GE that could present a potential issue for further consideration in any future Disposability Assessment interactions are identified by shading of the relevant cell in orange. These issues are discussed further below.

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 Hitachi-GE, consistent with expectations for this stage of the GDA Disposability Assessment. In future, RWM would expect to work with potential reactor operators to achieve fully-developed proposals through the Disposability Assessment process.

The proposed use of grout cement for waste conditioning conforms to existing practices for similar wastes in the UK and would be expected to produce wasteforms that could meet existing RWM specifications. However, Hitachi-GE did not identify candidate grouts. Instead, Hitachi-GE has referred to accepted practice used in the UK to package wastes with similar physical and chemical characteristics. Details of specific grouts, their properties and formulation development will be required in future Disposability Assessment submissions.

In particular, resins will also contain some particulates (e.g. corrosion products); these are not expected to be problematic. Resins may also take up solutes added to the coolant (such as zinc, which is added to the coolant for corrosion control). Zinc could retard the setting of the cement grout, but it is expected to be at sufficiently low concentrations (on the order of 1ppb) that this is of little concern, and can be accounted for in defining the formulation envelope for the concrete. However, the possibility of zinc being present as a potential cement retardant would need to be addressed in more detail in future interactions under the Disposability Assessment process.

Crud consists largely of corrosion products of steel and zirconium, and is usually referred to as sludge. At this stage in the Disposability Assessment process no information has been assessed on how such products of steel and zirconium are concentrated into the Crud wastes. Other trace contaminants are expected to be present, and they would have to be accounted for in the grout formulation during the wasteform development process. Zinc could be incorporated into crud waste streams (e.g. if it plates out on steel surfaces) instead of, or as well as, being taken up by ion exchange resins. The zinc could potentially act as a cement retardant in crud wasteforms if present in sufficient quantities. Therefore, future interactions under the Disposability Assessment process should evaluate this potential route for zinc contamination.

Conditioning of control rods will require size reduction, as the control rods are approximately 4.3m in length and the maximum dimension of the $3m^3$ Box is 1.72m. Size reduction will need to be undertaken carefully to prevent release of the B_4C powder. Size reduction will also be required for Activated Metals, as the materials in this waste stream have lengths between 4 and 4.2m.

In general, the wastes, as defined by Hitachi-GE, contain no materials (e.g. complexing agents, acids, or high-silica-content materials) that are likely to affect chemical containment within the near field. However, information on the types of resins present in the wastes, a discussion of the expected degradation products and their impact on wasteform properties and radionuclide behaviour would be required as part of future submissions.

In general, Decommissioning ILW are expected to be successfully immobilised through encapsulation with grout. Details will need to be provided on the use of in-box furniture and residual void space in packages at more detailed stages of the Disposability Assessment process.

Table 14 Check-list criteria for wasteform characteristics resulting from Hitachi-GE proposals: Operational ILW

Waste Stream Group	**Cruds (UKABWR01 & UKABWR02)**	**Resins (UKABWR03, UKABWR04 & UKABWR05)**	**Control Rods (UKABWR06 & UKABWR07)**	**Activated Metals (UKABWR08)**
Conditioning proposal	The waste would be grout-cemented into $3m^3$ Drums using an in-drum, lost-paddle mixing to ensure a homogeneous wasteform	The waste would be grout-cemented into $3m^3$ Drums using an in-drum, lost-paddle mixing to ensure a homogeneous wasteform; polymer encapsulants would be used for any challenging resins that cannot be solidified using grout cement	The waste would be grout-cemented into $3m^3$ Boxes after size reduction	The waste would be grout-cemented cemented into $3m^3$ Boxes

Waste Stream Group	Cruds (UKABWR01 & UKABWR02)	Resins (UKABWR03, UKABWR04 & UKABWR05)	Control Rods (UKABWR06 & UKABWR07)	Activated Metals (UKABWR08)
Physical immobilisation	Significant UK experience of sludge encapsulation in grout cement gives high confidence of successful immobilisation; work needed to develop suitable formulation envelope	Possibility of segregation during in-drum mixing (resins are lower density than typical grouts); capping grout likely to be required; use of powder resins may require further wasteform development; particulates and solutes (e.g. Zn) unlikely to be of concern	Components expected to be unreactive and capable of successful encapsulation; in-box furniture may be needed to aid infiltration; boron carbide powders unlikely to be infiltrated by grout – possible implications for accident performance	Components expected to be unreactive and capable of successful encapsulation.
Mechanical / physical properties	Suitable compressive strength likely to be achieved, but should be confirmed through measurement; no data provided on mass transport properties, but expected to be acceptable; for mass and heat output, see Sections 4.2 (Tables 23 and 25) and 4.3			
Chemical containment	May contain minor amounts of organic species as contaminants that could degrade to form complexants, but concentrations not expected to be sufficient to affect chemical containment	Degradation products (e.g. amines, sulphates) not expected to be significant complexants compared to those from cellulose, but more information would be required in future submissions	Contain no materials likely to affect chemical containment in the near-field	Contain no materials likely to affect chemical containment in the near-field
Hazardous materials and other problematic components	No data provided on presence of toxic / hazardous materials, but not expected to be significant	No data provided on presence of toxic / hazardous materials, but not expected to be significant; radiolysis and thermal degradation may result in volatile amines, which may be significant during transport and operations	No data provided on presence of toxic / hazardous materials, but not expected to be significant	No data provided on presence of toxic / hazardous materials, but not expected to be significant

Waste Stream Group	Cruds (UKABWR01 & UKABWR02)	Resins (UKABWR03, UKABWR04 & UKABWR05)	Control Rods (UKABWR06 & UKABWR07)	Activated Metals (UKABWR08)
Wasteform evolution	Long-term stability of wasteform should be demonstrated through measurements of dimensional change and compressive strength as part of a formulation development programme, but no significant issues expected	Long-term stability of wasteform should be demonstrated through measurements of dimensional change and compressive strength as part of a formulation development programme, but no significant issues expected	No significant issues expected	No significant issues expected

Note: Orange Cells indicate potential issue for further consideration in any future Disposability Assessment interactions

Table 15 Check-list criteria for waste packages resulting from Hitachi-GE proposals: Decommissioning ILW

Waste Stream Group	RPV Internals (UKABWR09)	RPV (UKABWR10)
Conditioning proposal	Grout cemented into $3m^3$ Boxes	Grout cemented into 4m Boxes with 200-mm thick concrete walls
Physical immobilisation	Expected to be successfully encapsulated in the grout cement with immobilisation of the radionuclide content; in-box furniture may be necessary to aid infiltration; filters containing steel cuttings may need additional penetrations to immobilise steel cuttings and minimise voidage	
Mechanical / physical properties	Suitable compressive strength likely to be achieved, but should be confirmed through measurement; no data provided on mass transport properties, but expected to be acceptable; for mass and heat output, see Sections 4.2 (Tables 24 and 26) and 4.3	
Chemical containment	Contain no materials likely to affect chemical containment in the near-field	
Hazardous materials and other problematic components	Decommissioning RPV stainless steels contain elements such as lead, nickel, chromium and antimony, but these will also be present in the stainless steel packaging used throughout a geological disposal facility and hence would be expected to be an insignificant addition	
Wasteform evolution	No significant issues expected, although it will be necessary to confirm that the rate of carbon steel corrosion in RPV ILW is insufficient to result in significant wasteform expansion over the required period	

4.1.3 Waste Package Accident Performance

Context

The objective of the waste package accident performance evaluation is to assess the performance of waste packages under impact accidents and fire accidents, and to derive release fractions for each waste package under these conditions. The accident conditions evaluated include:

- impact accidents, for Type B packages:
  - a 9-m drop onto a flat unyielding target during transport;
  - an 11-m drop onto a flat unyielding target during geological disposal facility operations;
  - a 10-m drop onto an aggressive target during geological disposal facility operations;
- impact accidents, for Industrial Packages consequences are minimised by stringent requirements on the allowable contents, but RWM assess the following operational fault:
  - a 10-m drop onto a flat unyielding target;
- fire accidents:
  - a 30-minute, 800°C, fire during transport;
  - a 30-minute, 1,000°C, fire during geological disposal facility operations for unshielded ILW packages; and
  - a 1-hour, 1,000°C, fire during geological disposal facility operations for shielded ILW packages and HHGW packages.

In addition, the waste accident performance evaluation has considered the requirements of the L2 GS for LHGW where these are not explicitly addressed in the subsequent safety assessments. The requirements are:

- under all credible accident scenarios the release of radionuclides and other hazardous materials from the waste package shall be low and predictable; and
- the waste package should exhibit progressive release behaviour within the range of all credible accident scenarios.

Impact assessment methodology

To assess the impact performance of waste packages, finite element simulation of the package is used to predict energy absorption of the wasteform. This, together with data from break-up tests of wasteforms, is used to predict the break-up of the wasteform and estimate the particulate generated and particulate release fraction. The methodology applied in the waste package assessment performance for the UK ABWR is illustrated in Figure 9. The methodology is described in detail in [40]. Key steps include:

- calculation of the normalised impact energy (J/kg);
- prediction of the proportion of impact energy absorbed by the wasteform;
- use of existing break-up test charts for the most relevant wasteform to estimate the <100µm particulate generated by the impact as a proportion of the wasteform mass;
- conversion of the estimate of <100µm particulate generated to a release fraction through estimation of factors for the size of opening created in the container by the impact, the effect of internal container barriers (e.g. capping grout or anti-flotation plate) and the distance between the location of particulate generation and the container opening.

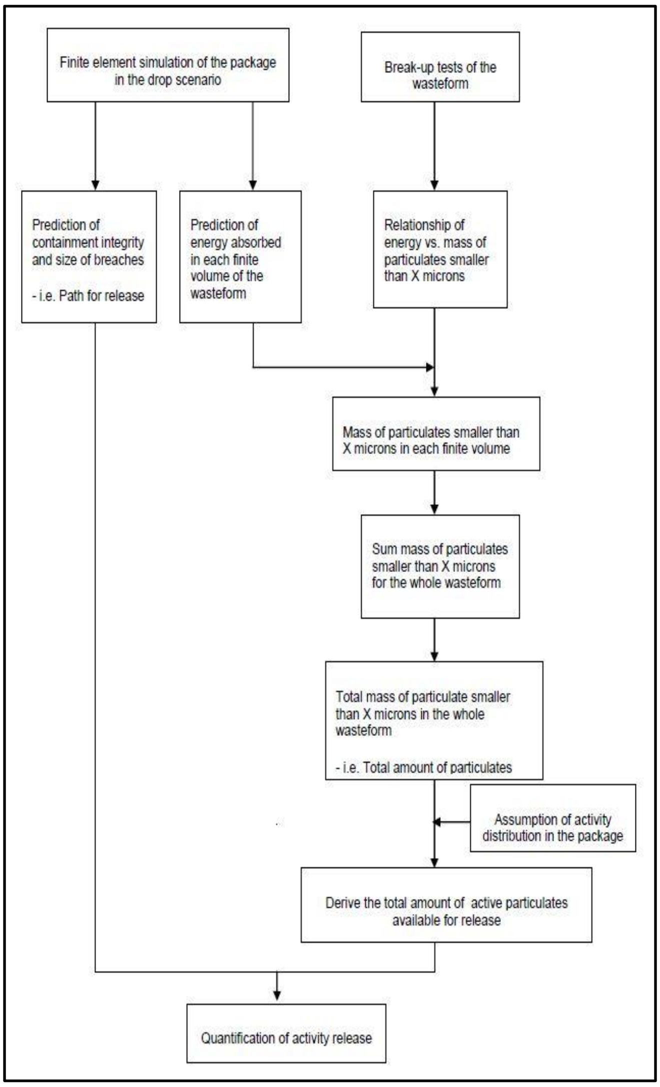
Release fractions for waste packages proposed by Hitachi-GE in discussion with RWM for the UK ABWR were based on existing work conducted by RWM, mainly finite element analyses. Because no grout formulation was supplied by Hitachi-GE, results are reported for the resulting wasteform that produces the greatest particulate generation for the given energy absorbed.

Fire assessment methodology

RWM has developed a methodology for assessing the quantity of radionuclides released from a waste package exposed to a fire, which has been successfully applied to a range of different waste packages [41]. First, the maximum temperature experienced by the wasteform is determined using finite element models of heat transfer (thermal modelling). Small-scale tests have been performed to determine the release fractions of several significant radionuclides as a function of temperature, from various wasteforms. Recommended release fractions for other radionuclides have been determined through similarity to one of the tested radionuclides (based on volatility). Thus, the expected release of radionuclides from the wasteform at the expected temperature can be determined.

Typical fire release fractions for a range of waste packages [42] were used in this GDA Disposability Assessment, because the submission from Hitachi-GE did not provide detailed descriptions of waste package contents and grout formulations.

Figure 9 Methodology used to estimate impact assessment release fractions [43]



Results and implications

Impact Assessments

For the $3m^3$ Box, previous work discussed in [43] has been used to develop the following assumptions to underpin estimation of release fractions for impact accidents:

- for transport accidents, the SWTC provides containment; in addition, finite element modelling predicts that there is no breach of the $3m^3$ Box releasing activity into the SWTC;
- based on finite element analyses, it is conservatively assumed that the wasteform absorbs 65% of the total impact energy, and that 100% of generated particulates would be released following an 11-m drop onto an unyielding target; and
- for drops onto aggressive targets, a drop at the worst attitude (usually the lid edge) is assessed, in which case the impact would affect a weak part of a container and almost all of the impact energy might be absorbed by the wasteform; therefore, it is assumed that the wasteform absorbs 100% of the total impact energy and that 100% of the particulate is released.

For the 3m^3 Drum, previous work discussed in [43] has been used to develop the following assumptions to underpin estimation of release fractions for impact accidents:

- for transport accidents, the SWTC provides containment; in addition, consistent with the approach for the 3m^3 Box, it is assumed that there is no release from the drum;
- based on finite element analyses, it is conservatively assumed that the wasteform absorbs 75% of the total impact energy, and that 10% of generated particulates would be released following an 11-m drop onto an unyielding target; this is lower than the 3m^3 Box owing to the rounded nature of the 3m^3 Drum; and
- for drops onto aggressive targets, consistent with the approach for the 3m^3 Box, it is assumed that the wasteform absorbs 100% of the total impact energy and that 100% of the particulate is released.

For the 4-metre Box with 200-mm thick concrete shielding, previous finite element modelling discussed in [43] has estimated that the energy absorbed by the wasteform under a 15-m drop to be 12.6-20% of the total impact energy. For the GDA Disposability Assessment it is conservatively assumed that for a 10-m drop onto a flat, unyielding surface, the wasteform absorbs 20% of the total impact energy and 100% of the particulate is released.

Impact release fractions are presented in Tables 16 and 17.

Table 16 Impact release fractions for operational ILW

Waste Type	Package	Predicted Release Fraction for <100μm Particulate		
		2012 IAEA Transport Regulations – 9-m drop onto a flat, unyielding target	**ILW Operational Safety Case – 11-m drop onto a flat, unyielding target**	**ILW Operational Safety Case – 10-m drop onto an aggressive target**
Cruds (UKABWR01 & UKABWR02)	3m^3 Drum	0	5.30E-05	6.42E-04
Resins (UKABWR03, UKABWR04, & UKABWR05)	3m^3 Drum	0	5.30E-05	6.42E-04
Hafnium Control Rods (UKABWR06)	3m^3 Box	0	4.33E-05	6.06E-04
Boron Carbide Control Rods (UKABWR07)	3m^3 Box	0	4.39E-05	6.14E-04
Activated Metals (UKABWR08)	3m^3 Box	0	4.23E-05	5.92E-04

Table 17 Impact release fractions for decommissioning ILW

Waste Type	Package	Predicted Release Fraction for <100μm Particulate		
		2012 IAEA Transport Regulations – 9-m drop onto a flat, unyielding target	ILW Operational Safety Case – drop onto a flat, unyielding target	ILW Operational Safety Case – 10-m drop onto an aggressive target
RPV Internals (UKABWR09)	$3m^3$ Box	0	4.36E-05 (11-m drop)	6.10E-04
RPV (UKABWR10)	4-metre Box	Not Applicable	1.75E-04 (10-m drop)	Not Applicable

Fire Assessments

For the $3m^3$ Box and $3m^3$ Drum, previous modelling of a 500-litre Drum contained in an SWTC-285 has been used to assume that the wasteforms experience a temperature between 85°C and 143°C. For the 4-metre Box, release fractions used in the GDA Disposability Assessment for the UK ABWR are the same as those estimated in RWM's generic release fractions report [44].

Fire release fractions are presented in Tables 18-21. For the assessment of release fractions, RWM groups all the elements relevant in disposal assessment into six Volatility Groups which decrease in volatility from I to VI. Group I includes C, Cl, H and I; Group II Cd, Cs, Sn and Tc through to Group VI which contains the least volatile elements that include Gd and Hf.

Table 18 Release fractions for Operational ILW for a transport fire

Volatility Group	Cruds (UKABWR01 & UKABWR02)	Resins (UKABWR03, UKABWR04 & UKABWR05)	Control Rods (UKABWR06 & UKABWR07)	Activated Metals (UKABWR08)
I	1E+00	1E+00	1E+00	1E+00
II	3E-03	8E-06	7E-04	7E-04
III	3E-03	8E-06	7E-04	7E-04
IV	3E-04	1E-06	6E-05	6E-05
V	9E-05	1E-06	3E-05	3E-05
VI	9E-05	1E-06	3E-05	3E-05

Table 19 Release fractions for Operational ILW for a fire at the GDF

Volatility Group	Cruds (UKABWR01 & UKABWR02)	Resins (UKABWR03, UKABWR04 & UKABWR05)	Control Rods (UKABWR06 & UKABWR07)	Activated Metals (UKABWR08)
I	1E+00	1E+00	1E+00	1E+00

Volatility Group	Cruds (UKABWR01 & UKABWR02)	Resins (UKABWR03, UKABWR04 & UKABWR05)	Control Rods (UKABWR06 & UKABWR07)	Activated Metals (UKABWR08)
II	1E-02	8E-06	1E-03	1E-03
III	3E-03	8E-06	1E-03	1E-03
IV	6E-04	1E-06	6E-05	6E-05
V	1E-04	1E-06	3E-05	3E-05
VI	1E-04	1E-06	3E-05	3E-05

Table 20 Release fractions for Decommissioning ILW for a transport fire

Volatility Group	RPV Internals (UKABWR09)	RPV (UKABWR10)
I	1E+00	1E+00
II	7E-04	7E-04
III	7E-04	7E-04
IV	6E-05	6E-05
V	3E-05	3E-05
VI	3E-05	3E-05

Table 21 Release fractions for Decommissioning ILW for a fire at the GDF

Volatility Group	RPV Internals (UKABWR09)	RPV (UKABWR10)
I	1E+00	1E+00
II	1E-03	7E-04
III	1E-03	7E-04
IV	6E-05	6E-05
V	3E-05	3E-05
VI	3E-05	3E-05

As noted above, the current assessment has developed conservative inventories. As an example, the conservatisms in the control rod inventories are summarised below:

- The control rods are assumed to be activated to the maximum level along their entire lengths. In reality they will only activated to the highest level at their tips that spend the greatest time in the reactor flux.
- The control rods are assumed to be packaged 40 in a box. In reality, the control rods will require locating furniture and the packing per box will then fall to around 15 control rods per package. The activity per package will therefore be reduced to around a third of it currently assumed value.
- The RWM TOpCAT model assumes the entire mass of the radioactive element present is released into the transport container. In reality, the H-3 and C-14 are

present spread evenly through the volume of the metal. The vast majority is unavailable for release into the transport container.

- The control rods are planned to be grouted into a $3m^3$ box. In the fire accident conditions defined (1000^oC for 1 hour or 1000^oC for 30 minutes) the vast majority of control rod material will not experience significantly elevated temperatures.

The criteria on wasteforms described above have been used as a check-list for the review of the proposed wasteforms. The results of the evaluation are provided in Table 22 (operational ILW) and Table 23 (decommissioning ILW). Based on the information supplied to date, the wasteforms proposed by Hitachi-GE do not present any issue for further Disposability Assessment interactions.

Table 22 Check-list criteria for wasteform characteristics resulting from Hitachi-GE proposals: Operational ILW

Waste Stream Group	Cruds (UKABWR01 & UKABWR02)	Resins (UKABWR03, UKABWR04 & UKABWR05)	Control Rods (UKABWR06 & UKABWR07)	Activated Metals (UKABWR08)
Releases Low and Predictable	The release fractions estimated using a conservative approach for the UK ABWR Cruds are based on impact modelling and drop testing of a generic $3m^3$ Drum; and demonstrate that the releases are low and releases are predictable	The release fractions estimated using a conservative approach for the UK ABWR Resins are based on impact modelling and drop testing of a generic $3m^3$ Drum; and demonstrate that the releases are low and releases are predictable	The release fractions estimated using a conservative approach for the UK ABWR Control Rods are based on impact modelling and drop testing of a generic $3m^3$ Box; and demonstrate that the releases are low and releases are predictable	The release fractions estimated using a conservative approach for the UK ABWR Activated Metals are based on impact modelling and drop testing of a generic $3m^3$ Box; and demonstrate that the releases are low and releases are predictable
Progressive Release Behaviour	Impact modelling and drop tests of a generic $3m^3$ Drum have demonstrated progressive release behaviour	Impact modelling and drop tests of a generic $3m^3$ Drum have demonstrated progressive release behaviour	Impact modelling and drop tests of a generic $3m^3$ Box have demonstrated progressive release behaviour	Impact modelling and drop tests of a generic $3m^3$ Box have demonstrated progressive release behaviour

Table 23 Check-list criteria for waste packages resulting from Hitachi-GE proposals: Decommissioning ILW

Waste Stream Group	RPV Internals (UKABWR09)	RPV (UKABWR10)
Releases Low and Predictable	The release fractions estimated using a conservative approach for the RPV Internals are based on impact modelling and drop testing of a generic $3m^3$ Box; and demonstrate that the releases are low and releases are predictable	The release fractions estimated using a conservative approach for the RPV are based on impact modelling of a generic 4-metre Box; and demonstrate that the releases are low and releases are predictable

Waste Stream Group	RPV Internals (UKABWR09)	RPV (UKABWR10)
Progressive Release Behaviour	Impact modelling and drop tests of a generic $3m^3$ Box have demonstrated progressive release behaviour	Impact modelling of a generic 4-metre Box have demonstrated progressive release behaviour

4.2 Disposal System Issues

4.2.1 Impact on Disposal Facility Design

Context

The GDA Disposability Assessment for the UK ABWR has considered implications for geological disposal facility design of disposing of ILW from a UK ABWR. This has included estimating the impact of the additional ILW from operation and decommissioning of a UK ABWR on the projection of a geological disposal facility area on the land surface (the “footprint”). This analysis is based on the illustrative designs discussed in Section 2.1.1.

The design of the transport system currently assumes the use of a four-axle rail wagon, as this would allow transport of waste on a large proportion of the UK rail network. The maximum permitted gross mass of such a wagon is 90t. Assuming that the mass of an unladen rail wagon is ~26t and the mass of an empty SWTC-285 is 52t, the maximum gross mass of a waste package transported by an SWTC-285 would be 12t. This is applied as the gross mass limit for $3m^3$ Boxes. For $3m^3$ Drums, a lower gross mass limit of 8t is applied. For 4-metre Boxes, which are IP packages the transport limit of 64t is applied.

Results and implications

The evaluation of design impact [45] assumed that operational and decommissioning ILW packaged in $3m^3$ Boxes and $3m^3$ Drums would be emplaced in unshielded ILW (UILW) vaults and decommissioning ILW packaged in 4m Boxes would be emplaced in shielded ILW (SILW) vaults. The $3m^3$ Boxes, $3m^3$ Drums and 4m Boxes that are proposed for packaging of UK ABWR operational and decommissioning ILW are UK standard packages, and, therefore, would not present any new issues for handling, stacking, lifting and identification. The expected waste package masses are within the limits required for transport given the current transport system design assumptions, and also for stacking in vaults given the current assumptions regarding stack height (see Tables 24 and 25).

The potential impact of the disposal of UK ABWR operational and decommissioning ILW on the size of a geological disposal facility has been assessed. It has been concluded that the ‘footprint area’ required to dispose of ILW from a UK ABWR corresponds to approximately 45m of vault length for each UK ABWR (178m for a fleet of four reactors) for higher strength rock. For the illustrative fleet of four UK ABWR reactors, this represents no significant change in the overall footprint compared with current assumptions based on the inventory for disposal.

Table 24 Check-list criteria for waste package gross mass resulting from Hitachi-GE proposals: Operational ILW

Waste Stream Group	Cruds (UKABWR01 & UKABWR02)	Resins (UKABWR03, UKABWR04 & UKABWR05)	Control Rods (UKABWR06 & UKABWR07)	Activated Metals (UKABWR08)
Gross mass	Package mass is 4.9t, so within 8t limit for $3m^3$ Drum	Package mass is 4.9t, so within 8t limit for $3m^3$ Drum	Package mass is 8.7t (Hf control rods) or 7.7t (B_4C control rods), so within 12t limit for $3m^3$ Box	Package mass is 11.4t, so within 12t limit for $3m^3$ Box

Table 25 Check-list criteria for waste package gross mass resulting from Hitachi-GE proposals: Decommissioning ILW

Waste Stream Group	RPV Internals (UKABWR09)	RPV (UKABWR10)
Gross mass	Package mass is 8.18t, so within 12t limit for $3m^3$ Box	Package mass is 62.23t, so within 64t limit for 4m Box

4.2.2 Transport Safety Assessment

Context

RWM's remit includes development of the transport system by which ILW will be delivered from sites of arising to a geological disposal facility. This includes development of transport container concepts which will enable packaged wastes to be transported to a geological disposal facility in full compliance with the 2012 IAEA Transport Regulations [22] as incorporated into UK transport legislation. For demonstration of compliance, RWM assumes that transport packages are assessed under regulations related to exclusive use. RWM has produced a generic Transport System Safety Assessment [46] and generic Transport Safety Case [47]. These documents are routinely used within the Disposability Assessment process to check that proposed waste packages are compliant with transport plans and do not compromise the generic safety case.

The aim of the ILW Transport Safety Assessment undertaken as part of the GDA Disposability Assessment was to examine the information supplied by Hitachi-GE, including any enhancement undertaken during earlier stages of the Disposability Assessment, to assess whether transport of ILW from the reactor under consideration to a geological disposal facility site will be possible, given the knowledge existing at the time of the assessment. The assessment was undertaken in three parts:

- an assessment of compliance with appropriate parts of the 2012 IAEA Transport Regulations [22], in particular:
    - radionuclide (activity) content: (i) for waste packages transported as part of a Type B transport package, or as Type B transport packages in their own right the total activity content of the transport package should not exceed 10^5A_2, and (ii) the contents of waste packages transported as part of a Type IP transport package, or as Type IP transport packages in their own right, shall be capable of being categorised as low specific activity (LSA) by

having a specific activity of less than 1E-04A_2g^{-1} for LSA-II or 2E-03 A_2g^{-1} for LSA-III; or as surface contaminated objects (SCO);

- dose rates: for Type B packages (i.e. the 3m^3 Box and 3m^3 Drum in an SWTC) the external dose rate of the waste package should be compatible with the dose rate at 1m from any external surface of a transport package, not exceeding 0.1$mSvh^{-1}$; for IP packages (i.e. the 4m Box) Para 517 of the 2012 IAEA Transport Regulations [22] requires that "*the external radiation level at 3m from the unshielded material or object or collection of objects does not exceed 10mSv/h*.".
- pressurisation: gases generated by waste packages transported as part of a Type B transport package, or as Type B transport packages in their own right, shall not cause the internal pressure of the transport package to exceed a gauge pressure of 700kPa under normal conditions of transport;
- containment: release of activity from a Type B transport package must be less than $10^{-6}A_2h^{-1}$; only a negligibly small fraction of particulates would be released from an SWTC under normal conditions of transport, and, therefore, compliance with this requirement is only tested for gas release; RWM apply this limit to Industrial Packages as well as to Type B packages;
- containment following loss of shielding: the release of radioactive material from a Type B package is required to be less than 1A_2 in the week following loss of shielding; for IP packages, RWM assesses performance against the 2012 IAEA Transport Regulations [22] criteria of 10mSv/hr at 3m from the unshielded contents of the waste package.
- criticality; the 2012 IAEA Transport Regulations [22] state that a waste package is fissile excepted if it contains less than 45g of fissile material; this criterion has been used to judge the acceptability of the proposed waste packages for transport;

- an assessment of compliance with the L2 GS for LHGW [15] (not including those elements already checked against the requirements of the 2012 IAEA Transport Regulations [22]), in terms of:
    - heat output: thermal modelling work undertaken by RWM has shown that current designs of SWTC could be used to transport waste packages with heat outputs of up to ~400W without exceeding the regulatory temperature or heat flux limits [48], and that the corresponding value for a 4-metre Box is ~200W [49].
- Effect on the generic Transport System Safety Assessment [46], in terms of whether Hitachi-GE's packaging proposals result in any significant change to operator doses calculated for the transport of the Derived Inventory of UK radioactive waste to a geological disposal facility.

Only components of the 2012 IAEA Transport Regulations [22], Level 2 Generic Specification for LHGW and the generic Transport System Safety Assessment that are appropriate to consider at this stage in assessment of Hitachi-GE's proposals have been assessed. This is because certain requirements relate to detailed specifications for the wasteforms and transport packages that are not available at this stage in the process. These include assessment of other dangerous properties (e.g. explosiveness, flammability, pyrophoricity, chemical toxicity and corrosiveness) surface contamination, and flammable and toxic gas generation. Consideration of these issues will be included in later stages of assessment under the Disposability Assessment process.

The assessment was carried out on the following transport packages:

- $3m^3$ Drum in SWTC-070 (Type B Transport Package):
    - CF Crud.
- $3m^3$ Drum in SWTC-285 (Type B Transport Package):
    - LCW Crud;
    - CUW Resin;
    - FPC Resin; and
    - DEC Resin.
- $3m^3$ Box in SWTC-285 (Type B Transport Package)
    - Hafnium Control Rods;
    - Boron Carbide Control Rods;
    - Mixed Metal; and
    - RPV Internals.
- 4-metre Box with 200mm shielding; no overpack (IP-2 Transport Package):
    - RPV.

Results and implications

The criteria on transport packages discussed above have been used as a check-list for the review of transport packages assumed in the UK ABWR GDA Disposability Assessment. The results of the evaluation are provided in Table 26 (operational ILW) and Table 27 (decommissioning ILW).

The transport safety assessment undertaken as part of the GDA Disposability Assessment has provided confidence that transport packages used to transport ILW from a UK ABWR will be transportable according to the 2012 IAEA Transport Regulations [22], RWM waste package specifications and the RWM generic Transport System Safety Assessment.

A range of issues have been identified through the transport assessment [50]. These have been highlighted by shading of the relevant cells in Tables 26 and 27 in orange and are discussed further below. These are principally related to the assumptions regarding the maximum package inventories and management of these inventories during packaging, and RWM expect that these issues would be considered in a future Disposability Assessment interaction with the operators.

The average package inventory for cruds and resins assumes that the material arises on a yearly basis for sixty years. The material could be moved off-site as it arises, and, if so, short-lived radionuclides may increase the package inventories. Therefore, a longer list of radionuclides might need to be reported. Reporting a longer list of radionuclides should be considered in future interactions under the Disposability Assessment process.

Activity content limits and heat output limits are exceeded for the maximum packages for Hafnium Control Rods and Activated Metals. For Hafnium Control Rods, this could be resolved by adjusting the assumed packing density, which is considered to be high, as noted in Section 3.3.1. Future operators of a UK ABWR should implement a process that ensures that the tips of control rods are not packaged together. The activities of the Activated Metal waste streams are also likely to revised downwards once conservatisms in the inventories are addressed.

The proposal to use RWM 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 2012 IAEA Transport Regulations [22]. In certain cases, as described in Table 26, dose rates are exceeded by the transport packages. However, as described in Section 3.3, there are significant pessimisms in the assessment inventories. These include the neutron fluxes experienced by the control rods and the expected date of shipment. More realistic assessment inventories, as would be expected at later stages of the Disposability Assessment process, will reduce the dose rates.

Similarly, pessimistic assumptions were made regarding gas generation from the waste packages, as no gas generation rates were available. If the entire inventory of volatility group I radionuclides were in a gaseous form, limits would be exceeded for some waste streams, and, therefore, there is a need to undertake a gas generation assessment at a later stage of the Disposability Assessment process. In particular, there is a need to establish the physical form of H-3 and C-14, which dominate the activity during transport (H-3 and C-14 are assumed to be 100% gaseous in the assessment). The current assessment has identified that this is particularly an issue for the B_4C control rods. A future assessment could consider, for example, the mechanisms of H-3 production through activation and the mechanisms of gas release (e.g. the extent to which diffusion through the metallic wastes and the extent to which isotopic exchange between escaping H-3 and water contained in the grout encapsulants would reduce the rate of gas release from the waste packages).

For DEC Resins, the assessment inventory includes a maximum package inventory of 78.7g of fissile nuclides per waste package. Pessimisms in this inventory and/or proposed packing densities will need to be re-evaluated during future interactions under the Disposability Assessment process to determine if this will reduce the maximum package mass of fissile material.

The heat output for the maximum inventories at time of transport, Waste stream heat output for maximum package inventories are lower than the limits of 200W per 4m box and 400W per SWTC except for Control Rod and Activated Metals waste streams which are significantly higher and so present a challenge. If the pessimisms in the inventory noted in Section 3.3.3 are removed and revised packing densities used, these heat outputs will be reduced. In addition, the heat output is dominated by Co-60 and Hf-178n which have a half-life of 5.27 and 31 years respectively. Therefore, decay storage could reduce the heat outputs to acceptable levels.

Table 26 Check-list criteria for waste package transport safety resulting from Hitachi-GE proposals: Operational ILW

Waste Stream Group	Cruds (UKABWR01 & UKABWR02)	Resins (UKABWR03, UKABWR04 & UKABWR05)	Control Rods (UKABWR06 & UKABWR07)	Activated Metals (UKABWR08)
Activity content	The maximum A_2 multiples for cruds is 2.42E+00	The maximum A_2 multiples for resins is 2.36E+03	The maximum A_2 multiples for Control Rods is 1.20E+05 for the Hafnium Control Rods	The maximum A_2 multiples for Activated Metals is 1.43E+05

Waste Stream Group	Cruds (UKABWR01 & UKABWR02)	Resins (UKABWR03, UKABWR04 & UKABWR05)	Control Rods (UKABWR06 & UKABWR07)	Activated Metals (UKABWR08)
External dose rate	Crud waste streams have dose rates for the maximum package at 1m of at or below 4.73E-02mSvh^{-1}	With the exception of Decommissioning Resin, resin waste streams have dose rates for the maximum package at 1m of at or below 1.83E-02mSvh^{-1}; Decommissioning Resin has dose rates for the maximum package at 1m of 1.07E-01mSvh^{-1}	Control Rod waste stream dose rates exceed the limits in the 2012 IAEA Transport Regulations; with dose rates for the maximum package at 1m up to 4.99E+00mSvh^{-1}	The Activated Metal waste stream dose rate exceeds the limits in the 2012 IAEA Transport Regulations; with dose rates for the maximum package at 1m of 1.44E+01mSvh^{-1}
Pressurisation	The transport assessment concluded that gas generation rates will be low enough to ensure satisfactory transport container leakage rates and that pressurisation of the transport container will be avoided; this should be confirmed in a future Disposability Assessment			
Containment: Gas Generation	At this pre-conceptual stage of assessment, no gas generation rates were available; if the entire inventory of volatility Group I radionuclides were in a gaseous form, limits would be exceeded for some waste streams, and, therefore, there is a need to undertake a gas generation assessment at a later stage of the Disposability Assessment process; in particular, there is a need to establish the physical form of H-3 and C-14, which dominate the activity during transport (H-3 and C-14 are assumed to be 100% gaseous in the assessment); Kr-85 and Ar-39 are also significant for some waste streams			
Containment Following Loss of Shielding	Crud waste streams are assessed to release less than 0.1% of an A_2 multiple in the week following an accident	Resin waste streams are assessed to release 0.5% of an A_2 multiple or less in the week following an accident, apart from Decommissioning Resin, which is assessed to release approximately 23% of an A_2 in the week following an accident	Control Rod and Activated Metal waste streams exceed the accident limits, but, as for containment during normal conditions, this is the result of pessimistic assumptions regarding H-3 and C-14	

Waste Stream Group	Cruds (UKABWR01 & UKABWR02)	Resins (UKABWR03, UKABWR04 & UKABWR05)	Control Rods (UKABWR06 & UKABWR07)	Activated Metals (UKABWR08)
Criticality Safety	Crud waste streams contain a maximum of 3.03E-02g of fissile nuclides per package and are therefore fissile excepted	CUW and FPC Resin waste streams contain a maximum of 1.83E+00g of fissile nuclides per package and are therefore fissile excepted; DEC Resins contain a maximum of 7.87E+01g of fissile nuclides per package and must therefore be considered further	Control Rod waste streams contain a maximum of 1.53E+00g of fissile nuclides per package and are therefore fissile excepted	Activated Metals contain a maximum of 3.47E+00g of fissile nuclides per package and are therefore fissile excepted
Heat Output	Heat output for Crud waste streams is a maximum of 9.55E-03W at discharge	Heat output for Resin waste streams is a maximum of 2.40E+02W at discharge	Heat output for Control Rod waste streams is a maximum of 5.94E+03W at discharge	Heat output for Activated Metals is a maximum of 2.35E+04W at discharge
Operator Doses	The impact on operator dose from transport of operational ILW from a UK ABWR due to the waste packaging proposal as calculated by TranSAT are not significant (the combined operational and decommissioning ILW inventory leads to an increase in the maximum operator dose of 1.96%) and do not have any implications on the safety arguments presented in the Transport System Safety Assessment			

Note: Orange Cells indicate potential issue for further consideration in any future Disposability Assessment interactions.

Table 27 Check-list criteria for waste package transport safety resulting from Hitachi-GE proposals: Decommissioning ILW

Waste Stream Group	RPV Internals (UKABWR09)	RPV (UKABWR10)
Activity content	The maximum package A_2 multiples for RPV Internals is 9.14E+02	The maximum package specific activity for RPV ILW is 6.53E-10A_2g^{-1}. Note: this meets the LSA-II criterion.
External dose rate	RPV Internals have dose rates for the maximum package at 1m of at or below 5.84E-02$mSvh^{-1}$	RPV ILW has dose rates for the maximum package at 1m of at or below 1.31E-02$mSvh^{-1}$
Pressurisation	The transport assessment concluded that gas generation rates will be low enough to ensure satisfactory transport container leakage rates and that pressurisation of the transport container will be avoided; this should be confirmed in a future Disposability Assessment	
Containment: Gas	At this pre-conceptual stage of assessment, no gas generation rates were available; if the entire inventory of	Gas would be released during packaging and interim storage; therefore, at this stage in the

Waste Stream Group	RPV Internals (UKABWR09)	RPV (UKABWR10)
Generation	volatility Group I radionuclides were in a gaseous form, limits would be exceeded for some waste streams, and, therefore, there is a need to undertake a gas generation assessment at a later stage of the Disposability Assessment process; in particular, there is a need to establish the physical form of H-3 and C-14, which dominate the activity during transport (H-3 and C-14 are assumed to be 100% gaseous in the assessment); Kr-85 and Ar-39 are also significant for some waste streams	assessment, it is reasonable to assume that a release rate of less than $10^{-6}A_2h^{-1}$ is achieved during transport; a gas generation assessment is required to confirm this
Containment Following Loss of Shielding	RPV Internals exceed the accident limits, but, as for containment during normal conditions, this is the result of pessimistic assumptions regarding H-3 and C-14	Maximum dose rate is 5.96E-02mSv/hr
Criticality Safety	RPV Internals contain a maximum of 3.16E-01g of fissile nuclides per package and are therefore fissile excepted	RPV wastes contain a maximum of 2.57E-05g of fissile nuclides per package and are therefore fissile excepted
Heat Output	Heat output for RPV Internals is a maximum of 1.02E+02W at discharge	Heat output for RPV wastes is a maximum of 2.61E-03W at discharge
Operator Doses	The impact on operator dose from transport of decommissioning ILW from a UK ABWR due to the waste packaging proposal as calculated by TranSAT are not significant (the combined operational and decommissioning ILW inventory leads an increase in the maximum operator dose of 1.96%) and do not have any implications on the safety arguments presented in the Transport System Safety Assessment	

Note: Orange Cells indicate potential issue for further consideration in any future Disposability Assessment interactions.

4.2.3 Operational Safety Assessment

Context

The Disposability Assessment process uses RWM's generic Operational Safety Case [51] and a series of generic Operational Safety Assessments [52, 53, 54, 55] to test proposed waste packages and to check compliance with assumed performance and accident consequence criteria. A similar approach has been adopted for the UK ABWR GDA Disposability Assessment.

The aim of the operational safety assessment is to examine the information supplied by Hitachi-GE, including any enhancement undertaken during earlier stages of the Disposability Assessment, to assess whether disposal of waste from the reactor under consideration does not undermine assumptions within the generic safety assessments.

When ILW packages arrive at a geological disposal facility site they are assumed to be subject to acceptance checks and dispatched underground using the onsite transportation system. Packages arriving in an 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. Industrial Packages, such as the

4-metre Box would be placed in an underground temporary storage area prior to emplacement in a vault.

The Operational Safety Assessments are supported by a fault and hazard schedule which is routinely used within the Disposability Assessment 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 UK ABWR ILW, 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 [56]. The estimated doses were then compared to targets for design basis fault sequence mitigated doses currently being considered by RWM. These targets are reproduced in Table 28, and related to Basic Safety Limits (BSLs) and the Basic Safety Objective (BSO) as defined in Paragraphs 698-701 of the Office for Nuclear Regulation (ONR)'s Safety Assessment Principles (SAPs) [57]. The criteria for acceptability are listed in RWM's Radiological Protection Policy Manual (RPPM) [58] and are based on the regulatory expectations set out in the SAPs.

In the GDA Disposability Assessment for the UK ABWR, operational safety assessment of Cruds and Resin wastes considered specific bounding waste streams; i.e. LCW Crud, and CUW and DEC Resins [59].

Table 28 Targets for design fault sequence mitigated doses used in the UK ABWR Operational Safety Assessment

Criterion		On-site	Off-site
BSL	For initiating fault frequencies >1E-03 per annum	20mSv	1mSv
	For initiating fault frequencies between 1E-03 and 10-4 per annum	200mSv	10mSv
	For initiating fault frequencies <1E-04 per annum	500mSv	100mSv
BSO		0.1mSv	0.01mSv

The following general criteria are also considered in the GDA Disposability Operational safety assessment:

- limits on radiological dose to workers under normal conditions of operation: doses to workers during operation are evaluated on both a per-package basis and per-waste stream basis; the per-package limits are the same as those applied for transport, and, therefore, are not re-evaluated here; RWM's RPPM [58] establishes a design target of 1mSv for an employee working with ionising radiation, and these have been used to assess the acceptability of dose to workers during operations; for Type B packages, the assessment assumes a 10-minute handling and inspection period while the waste package is contained inside the transport container (SWTC); for Industrial Packages the assessment assumes an exposure time of 10 minutes; both of these assessments assume that the full inventory is disposed of in one year;
- limits on radioactive gas release;
- limits on toxic chemical release; and
- compliance with operational criticality safety limits: the general criticality safety assessment (GCSA) for wastes containing low levels of fissile material and

packaged in standard containers [60] supported a waste package screening level of 50g Pu-239 (or its equivalent in terms of other fissile isotopes). Application of the GCSA limit also requires that:

- the waste package contains no more than 1kg graphite, 100g beryllium, and 100g deuterated material;
- the waste package contains no more than trace quantities of exotic fissile materials (i.e. fissile materials other than U-233, U-235, Pu-239 and Pu-241) or their precursors; this constraint is satisfied for waste streams that arise from normal operations associated with the nuclear fuel cycle;
- the waste stream does not contain moderating materials that are more efficient than full-density polyethylene homogeneously distributed throughout the waste; and
- the wastes do not include favourable sites for sorption of fissile material relative to other repository materials, such as backfill, that could potentially lead to the accumulation of fissile material from many waste packages.

Results and implications

The operational safety criteria discussed above have been used as a check-list for the review of operational safety for the UK ABWR GDA Disposability Assessment. The results of the evaluation are provided in Table 29 (operational ILW) and Table 30 (decommissioning ILW).

The operational safety assessment undertaken as part of the GDA Disposability Assessment has provided confidence that waste packages can be disposed of in a manner that is consistent with the assumptions of the RWM Operational Safety Case [51] and generic Operational Safety Assessments [52, 53, 54, 55].

A range of issues have been identified through the operational safety assessment [59]. These have been highlighted by shading of the relevant cells in Tables 29 and 30 in orange and are discussed further below. These are principally related to the assumptions regarding the maximum package inventories; management of these inventories during packaging, assumptions in the development of the assessment inventories, and the expected schedule for disposal, and RWM expect that these issues would be considered in a future Disposability Assessment interaction with the operators.

Dose Assessments

There are several instances when the predicted doses are above the BSO/BSL (Tables 27 and 28). However, these can be attributed to known conservatisms in the assessment, which can be addressed at further stages of the Disposability Assessment process:

- the most significant contribution to the dose calculations for both impact and fire faults are from Co-60, and, as has been discussed in Section 3.3.3, a pessimistic assumption regarding the precursor concentration of Co-60 in metals; in addition, the assessment has considered transport and disposal of ILW at the time of arising, as the half-life of Co-60 is approximately 5 years, a significant reduction in the impact of this radionuclide can be gained through a more realistic treatment of the schedule of arisings in the assessment inventory and through consideration of decay storage;
- for fire faults, there is a significant contribution to estimated doses from radionuclides that are assigned to Fire Element Group 1 (FEG1), including H-3, C-14, Cl-36 and Se-79; the contribution for these radionuclides is pessimistic:
    - the inventory does not account for loss of activity during waste conditioning;

  - the assessment assumes that all of the inventory is in gaseous form and has a release fraction of unity; and
  - the assessment assumes that no capture occurs on the ventilation exhaust filters for these radionuclides.
- in the impact assessment, the activated materials are assumed to be in particulate form, this is a pessimistic assumption made by the ROSA Toolkit as it is highly unlikely that the activated metals could be rendered into a finely divided and releasable particulate in reality, so it tends to overestimate the doses in the impact accidents; and
- there are a number of conservatisms involved in the derivation of the impact release fractions; for example, for the most severe impact fault, which involves multiple packages being affected by a stack collapse, the same impact release fraction is applied to all damaged packages, although some will not experience the same drop height.

Toxic Chemical Release

These wastes are not expected to contain significant quantities of hazardous materials. With the exception of the resins, the wastes are predominantly metals. There is no indication of any significant chemical content that could be released in a volatile form (apart from H_2) although this will be re-considered for a future stage submission when further details on the chemical composition of the wastes is provided. It is noted from the wasteform evaluation (see Table 13) that the radiolysis and thermal degradation of anion-exchange resins may be expected to result in the formation of volatile amines, which may be of significance during transport and operations. This may require consideration of a requirement to abate and manage any arisings. However, currently there are no details of the resins to be used and it is therefore recommended that further information on the resins and information on the chemical content of the waste be sought in future stage submissions.

Boron carbide powder contained in Control Rods would not produce radionuclides that could lead to significant doses if a waste package containing ungrouted control rods were to be involved in an accident. However, the dust could present a chemotoxic hazard. No route has been identified as yet for treating any fine particulates that might be generated from size reduction during packaging (e.g. from grinding and drilling); any material <100 μm has the potential to be released as suspendible particles.

Antimony and beryllium, which may be present in the Activated Metals wastes, may present chemotoxic hazards. Such materials are unlikely to be important in terms of number of boxes, but might lead to specific packages requiring special consideration. The possibility of such materials being present in the waste will need to be considered in future interactions under the Disposability Assessment process.

Operational Criticality Safety

Fissile material contained within the ILW packages is, with the exception of DEC Resins, significantly less than the generic screening level for fissile inventory of 50g Pu-239 equivalent per package, and it is therefore assumed that these waste packages can be covered by RWM's generic criticality safety assessment. However criticality assurance documentation will be required for future stage submissions to confirm the applicability of the generic assessment to DEC Resins.

Table 29 Check-list criteria for waste package operational safety resulting from Hitachi-GE proposals: Operational ILW

<table>
<tr><th>Waste Stream Group</th><th>Cruds
(UKABWR01 & UKABWR02)</th><th>Resins
(UKABWR03, UKABWR04 & UKABWR05)</th><th>Control Rods
(UKABWR06 & UKABWR07)</th><th>Activated Metals
(UKABWR08)</th></tr>
<tr><td>Design Basis Impact Faults</td><td>Both on-site (worker) and off-site (public) protected doses were below the BSO</td><td>Some on-site (worker) doses were above the BSO, but the dose calculation included known conservatisms (e.g. applying the same release fraction to all waste packages affected by stack collapse incidents); on-site (worker) protected doses were below the BSL and off-site (public) protected doses were below the BSO</td><td colspan="2">Some on-site (worker) doses were above the BSL, with the doses being almost entirely dominated by Co-60, which has a pessimistic high activity in the assessment inventory; in addition, the ROSA toolkit assumes that all activated materials are in particulate form, thereby adding an additional over-estimation of the doses; off-site (public) protected doses were below the BSO, with the exception of one case, which was below the BSL 1mSv</td></tr>
<tr><td>Design Basis Fire Faults</td><td>Both on-site (worker) and off-site (public) protected doses were below the BSO</td><td>Both on-site (worker) and off-site (public) protected doses were below the BSO, with the exception of one fault for which the doses were just above the BSO owing to conservative assumptions regarding the release of elements assigned to FEG1 (Se-79 and Cl-36), these radionuclides would most likely be released during the packaging process</td><td colspan="2">The worst case results for fire scenarios gave on-site (worker) and off-site (public) protected doses above the BSO, and, in some cases above the BSL; these doses are the consequence of pessimistic activities of Co-60, an assumption that H-3, C-14, Cl-36 and Se-79 are all in gaseous form and have a release fraction of 1, that activated materials are in particulate form, and that no capture occurs on the ventilation exhaust filters for these radionuclides; for modified ROSA calculations without these radionuclides, the off-site (public) protected doses are reduced to below or close to the BSO</td></tr>
</table>

Waste Stream Group	Cruds (UKABWR01 & UKABWR02)	Resins (UKABWR03, UKABWR04 & UKABWR05)	Control Rods (UKABWR06 & UKABWR07)	Activated Metals (UKABWR08)
Operational Safety under Normal Conditions	The estimated dose was 1.12E-04mSv, which is below the RPPM design target	The maximum estimated dose was 0.13mSv, which is below the RPPM design target	The maximum estimated dose was 0.68mSv, which is below the RPPM design target	The estimated dose was 1.35mSv, which is above the RPPM design target
Gas Release	Significant gas generation not expected	Gas could be generated from radiolysis of ion-exchange resins and grout porewater. Over half Hitachi-GE-provided inventory of C-14 for operational ILW from one reactor (total of 79.6 TBq) is associated with DEC resin	Gas could be generated from corrosion of metals, but corrosion rates of carbon steel and stainless steel in high-pH environments are low and amounts of gas generated are not expected to be significant	Gas could be generated from corrosion of metals, but corrosion rates of carbon steel and stainless steel in high-pH environments are low and amounts of gas generated are not expected to be significant
Criticality Safety	Crud waste streams contain a maximum of 2.60E-02g of Pu-239$_{eq}$ and therefore meet the waste package screening level	CUW and FPC Resin waste streams contain a maximum of 1.21E+00g of Pu-239$_{eq}$ and therefore meet the waste package screening level; DEC Resins contain a maximum of 5.22E+01g of Pu-239$_{eq}$ and must therefore be considered further	Control Rod waste streams contain a maximum of 1.38E+00g of Pu-239$_{eq}$ and therefore meet the waste package screening level	Crud waste streams contain a maximum of 3.32E+00g of Pu-239$_{eq}$ and therefore meet the waste package screening level

Table 30 Check-list criteria for waste package operational safety resulting from Hitachi-GE proposals: Decommissioning ILW

Waste Stream Group	RPV Internals (UKABWR09)	RPV (UKABWR10)
Design Basis Impact Faults	On-site (worker) protected doses were above the BSO of 0.1mSv but below the most restrictive BSL of 20mSv for two of the three worst case impact faults; these faults are dominated by Ni-63 and Co-60 the ROSA Toolkit assumes these radionuclides are in particulate form which is an pessimistic assumption and overestimates the doses in the impact accidents; all the faults gave off-site (public) protected doses below the BSO of 0.01mSv	Both on-site (worker) and off-site (public) protected doses were below the BSO
Design Basis Fire Faults	On-site (worker) protected doses are above the BSO of 0.1mSv but below the BSL of 20mSv; the main contributors to dose are C-14 and Ni-63, the assessment assumed that all C-14 is released in a fire, which is pessimistic; off-site (public) protected doses are significantly above the BSL, but the assessment assumes that key radionuclides are all in gaseous form and have a release fraction of 1, that activated materials are in particulate form, and that no capture occurs on the ventilation exhaust filters for these radionuclides	On-site (worker) protected doses were below the BSO; off-site (public) protected doses were slightly above the BSO, but it is expected that removal of conservatisms associated with the temperature profile in the waste packages during a fire would reduce the doses to below the BSO
Operational Safety under Normal Conditions	The estimated dose was 0.25mSv, which is below the RPPM design target	The estimated dose was 5.14E-03mSv, which is below the RPPM design target
Gas Release	Gas could be generated from corrosion of metals, but corrosion rates of carbon steel and stainless steel in high-pH environments are low and amounts of gas generated are not expected to be significant	
Criticality Safety	RPV Internals contain a maximum of 3.12E-01g of Pu 239eq and therefore meet the waste package screening level	RPV wastes contain a maximum of 2.56E-05g of Pu 239eq and therefore meet the waste package screening level

Note: Orange Cells indicate potential issue for further consideration in any future Disposability Assessment interactions.

4.2.4 Environmental Evaluation

Context

The environmental evaluation 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 a geological disposal facility.

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 (EIA) requirements.

It is assumed that the environmental implications of waste treatment, packaging and storage prior to disposal will be addressed by the waste producers – including by discharging their obligations under the relevant UK EIA Regulations.

The environmental implications of disposing of UK ABWR wastes will, to a large extent, be determined by the design changes (to both a geological disposal facility and its associated transport system) required to accommodate the wastes. The environmental evaluation therefore draws on the Design Impact Assessment for UK ABWR wastes (Section 4.2.1).

Results and implications

Based on the submitted proposals and in terms of generic (non-radiological) environmental effects, the environmental evaluation concluded that the wastes from a single UK ABWR reactor should be considered as disposable. Current assumptions for treatment and packaging of the wastes are consistent with those made in generic design and assessment work. The proposals present no novel treatment and packing options that are likely to give rise to environmental (or socio-economic) effects that go beyond the scope of RWM's current generic assessment work.

As documented in Section 4.2.1, the disposal of UK ABWR wastes will result in a change in the underground footprint of a geological disposal facility to accommodate the additional wastes and an associated increase in excavated rock spoil, although this may be insignificant. This could result in additional environmental and socio-economic effects (both positive and negative) – largely associated with the transport of additional or reduced rock spoil off-site for disposal and with an extended operational period for the facility.

Should any additional wastes exceed the capacity of a geological disposal facility then a second facility may be needed, in which case the associated environmental effects would be significant.

The environmental evaluation provided the following recommendations with respect to the non-radiological environmental evaluation of wastes from a UK ABWR:

- further development of the waste management strategy for UK ABWRs should demonstrate application of the waste hierarchy;
- further development and selection of waste treatment and packaging options should demonstrate optimisation through use of a Best Available Technique (BAT) approach, which includes explicit consideration of environmental and socio-economic issues;
- BAT studies should consider the whole life cycle of wastes, from initial production through treatment, packaging, storage, transport to and disposal in a geological disposal facility; and
- BAT studies should be undertaken within the framework of an Integrated Waste Strategy to ensure that waste management and associated environmental effects are optimised across a UK ABWR site, and not just for one waste stream.
- It should be noted that in all cases, optimisation of radiation dose is an overriding requirement.

4.2.5 Physical Protection and Safeguards Evaluation

Context

The objectives of the physical protection evaluation were to determine whether the physical protection requirements for the wastes that could arise from the operation and decommissioning of an ABWR would be consistent with regulatory requirements and with plans for the transport of waste packages to a geological disposal facility.

Nuclear Material is defined as *"(a) any fissile material in the form of uranium metal, alloy or chemical compound, or of plutonium metal, alloy or chemical compound; or (b) any other fissile material which may be prescribed by regulations made by the Secretary of State"* [61].

The quantity of Nuclear Material contained within waste packages is required to be controlled such that they can be transported subject to standards of physical protection no higher than those defined for the transport system.

The Nuclear Industries Security Regulations (NISR) [62]) lay down the requirements for security of nuclear premises, security of transport of nuclear material and security of sensitive nuclear information. The Office for Nuclear Regulation (ONR) has issued National Objectives, Requirements and Model Standards (NORMS) for the protective security of civil licensed nuclear sites, other nuclear premises and nuclear material in transit [63] to support implementation of the NISR.

The security standards in NORMS are offered as a benchmark (i.e. Model Standard) to reflect internationally agreed recommendations on the physical protection of Nuclear Material published by the IAEA [64]. These standards also reflect the United Kingdom's obligations under the *Convention on the Physical Protection of Nuclear Materials* and its commitments under the Nuclear Suppliers Group Guidelines and Plutonium Management Guidelines.

The NORMS specifies mass limits for the quantities of Nuclear Material and ILW/LLW containing Nuclear Material that can be transported with four 'categories' of physical protection (Categories I to IV, Category I being the most restrictive). The generic Disposal System Technical Specification states the assumption that a geological disposal facility for LLW, ILW, HLW and spent fuel will be a Category II facility as a minimum under the current system [12].

The categorisation of Nuclear Material depends on whether or not the Nuclear Material can satisfy the NORMS definition of 'waste' which requires the material to be:

> *'…Nuclear Material arising from operations which have been or are to be discarded as Intermediate or Low Level Waste…provided that:*
>
> *a. the waste is in solid form including sludges without free liquid;*
>
> *b. the Nuclear Material is well dispersed and is not readily separable or recoverable;*
>
> *c. the mass of the Nuclear Material content is less than 1% of the total mass of the waste;*
>
> *d. the waste is stored or transported within the UK.'*

Furthermore, the waste must be transported in '*Concreted waste disposal containers'* which '*include containers where the waste is immobilised in a cementitious grout'*.

With respect to safeguards, all Nuclear Material is subject to safeguards, unless the safeguards status can be terminated. Termination can be achieved, following agreement between the site operator and Euratom, on the grounds of low Nuclear Materials concentration (e.g. 0.1%w/w for DU, or 4ppm for Pu), or if the Nuclear Material is in a form unsuitable for further use (e.g. finely dispersed in a cement matrix, or as widely spread surface contamination).

Results and implications

The physical protection evaluation concluded that all of the projected ILW, if packaged as suggested, could be transported with standards of physical protection no higher than those defined by the NORMS as Category III.

It is recommended that ONR(CNS) are contacted to discuss and agree the physical protection requirements for transport of the waste packages to a geological disposal facility.

For ILW from a UK ABWR 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. As such, it is RWM's view that termination could be achieved for most, if not all of this ILW.

4.3 Post-Closure Safety

Following emplacement of ILW and the decision to seal and close a geological disposal facility, the void space around ILW packages will be backfilled with suitable material. The current disposal concept adopts a cementitious backfill material, 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, a geological disposal facility will be expected to re-saturate 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 ESC which is required to demonstrate to regulators the expected behaviour of the disposal system in the long term. At this stage of geological disposal facility development, the post-closure safety component of the ESC exists for a generic geological disposal facility design and geological setting and is published as the generic Post-closure Safety Assessment (PCSA) [65], which supports the generic ESC [66]. The PCSA provides quantitative assessments for groundwater and human intrusion pathways for historical and currently arising wastes, and 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. The post-closure safety assessment considered the impact of disposing of ILW from a UK ABWR on these assessments.

In the case of UK ABWR 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 generic PCSA. This comparison included consideration of the following:

- groundwater pathway radiological assessment, including:
  - in-package inventory comparison;
  - groundwater pathway assessment;
  - Impact on solubility and sorption;
  - impact of voidage; and
  - consideration of disposal in alternative geological environments.
- gas pathway assessment;
- human intrusion assessment;
- limits on heat output;

- chemotoxic assessment;
- criticality assessment.

The generic PCSA does not provide quantitative assessments for chemotoxic materials or the gas pathway. To provide consistency with Disposability Assessment post-closure assessments, chemotoxic hazards from UK ABWR ILW is assessed against a screening assessment undertaken in 2007 [67], and hazards from the gas pathway are assessed against the results of an update to the 2003 Generic Post-closure Performance Assessment (GPA03) gas assessment [68].

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 a geological disposal facility [69]. 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 a UK ABWR 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.

The total packaged waste volume of UK ABWR operational ILW is 2,784 m^3, which is small (~0.8%) compared to the total conditioned waste volume of 364,000 m^3 of ILW assessed in the generic Post-closure Safety Assessment. Similarly, the number of packages, 717, is also small (0.3%) compared to the 208,350 ILW packages assessed in the generic Post-closure Safety Assessment[14]. The radionuclide activities of Reactor Pressure Vessel Internals, which is the most active ILW stream, were compared with an equivalent ILW stream from Sizewell B in Section 3.3.4, and shown to be similar. Given that the 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.

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 UK ABWR in a site of the type described would be consistent with meeting regulatory targets. The consideration of a fleet of four reactors would not alter this conclusion.

In-package inventories and risk from the groundwater pathway are consistent with the generic Post-closure Safety Assessment. The ILW from a UK ABWR has a much lower inventory of actinides than the ILW considered in the generic PCSA, and, therefore, risks in the long term are significantly lower. However, in-package inventory comparison and calculation of risks from the groundwater pathway have highlighted the impact of I-129 on post-closure performance. Peak risk from the UK ABWR ILW is almost entirely due to the inventory of I-129, and nearly all of this occurs in the DEC Resins. This conclusion results from the pessimistic approach to development of the assessment inventory for I-129, which, as explained in Section 3.3.3, was based on the maximum ratio to Ni-63 in existing datasets held by RWM. This is considered to be an artefact of the method used to estimate the activity of I-129 for the UK ABWR, based on assumptions relating to other reactor system performance, and may not accurately reflect the risks from a UK ABWR.

The post-closure safety assessment [69] has identified one issue regarding the organic content of the operational waste which would need to be considered in a future

[14] These numbers do not include LLW or DNLEU.

Disposability Assessment interaction with the operators. Assessment of the characteristics of the operational ILW waste packages noted that the mass loading of cellulose in the UK ABWR resin waste streams is about a factor of four higher than the average cellulose loading in the generic Post-closure Safety Assessment. Furthermore, the Hitachi-GE packaging proposals result in a higher organic loading per waste package for resin wastes than the generic Post-closure Safety Assessment average (2.2% compared to an average of 0.53% in historical and currently arising wastes). The total quantity of organic material in the UK ABWR ILW is not a major issue, however, the concentration in waste packages is at a high level (factor of 4 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 U-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 a geological disposal facility waste emplacement strategy. This issue would require further consideration by operators and RWM under a future Disposability Assessment interaction.

In addition, Reactor Pressure Vessel Internals stainless steel waste has high specific activity for a range of radionuclides, in particular C-14. This waste stream contains 595TBq of C-14. This value is similar to those calculated for the equivalent waste stream (RPV Internals) for other reactor systems. Further consideration/examination should be given to the assumed inventory of C-14 and its release rate from the steel matrix. In particular, it will be necessary to determine the fraction of the C-14 that would be released as carbon dioxide (and would react with the cementitious backfill) and the fraction that would be released as methane (and which could migrate to the biosphere). RWM currently assumes that all C-14 forms radioactive methane. The form in which C-14 is released from ILW is a matter of ongoing generic research within RWM.

Human Intrusion Pathway

The siting process adopted by Government [70] has identified geological environments that should be avoided due to the presence of natural resources and which are, therefore, areas where human intrusion is more likely to occur. Addressing the environment agencies' Guidance on Requirements for Authorisation (GRA) requirements [71] for human intrusion requires that any practical measures to reduce the risk from human intrusion are implemented in a geological disposal facility and that potential risks from human intrusion are optimised. These requirements do not relate, therefore, to the fundamental disposability of ILW.

Heat Outputs

The Level 2 Generic Specification for LHGW states that the temperature targets for a geological disposal facility would not be threatened by average heat output of $6Wm^{-3}$ of conditioned waste and that individual waste packages with heat outputs of up to ~$100Wm^{-3}$ of conditioned waste would not cause the temperature targets to be exceeded. The waste stream that most significantly challenges the heat output criteria is the hafnium control rods, which are estimated to have an average package heat output at 2150 (the assumed date of backfilling) of approximately $15Wm^{-3}$, and a maximum package heat output at 2150 of approximately $49Wm^{-3}$.

It is recognised that there are only eight hafnium control rod waste packages in the disposal inventory for a UK ABWR, and, therefore, it is appropriate to assess the wastes against the $100Wm^{-3}$ limit. In addition, RWM are of the view that it is unlikely that 40 control rods can be sectioned and placed in a $3m^3$ box. RWM believe it is more likely that box furniture will be required to locate the sectioned control rods and that the equivalent of 12-15 full control rods will be packaged in each $3m^3$ box, increasing the number of product boxes pro rata. Furthermore, it is noted that the control rod assessment inventory pessimistically assumes that the control rods will experience the maximum neutron flux across their entire length, although in reality only the tips will see the maximum neutron flux.

Nonetheless, the during future Disposability Assessment interactions the hafnium control rod waste package heat outputs should be re-evaluated to check their suitability for disposal at the assumed cooling times.

Chemotoxic Assessment

Consideration of the potential impact of chemotoxic elements contained in the UK ABWR ILW was undertaken by comparing the mass of chemotoxic species in the RPV Internals and the RPV waste streams with the masses considered in the 2007 screening assessment [67]. In all cases, the ratio of the masses of chemotoxic species in the ABWR ILW to the totals in the 2007 screening assessment were small (approximately 1:1,000 to 1:10,000). The UK ABWR ILW is therefore unlikely to significantly increase the chemotoxic hazard presented by a geological disposal facility.

However, further confirmation should be sought during future interactions under the Disposability Assessment process that all chemotoxic species have been identified in the UK ABWR ILW, and also that the 2007 screening assessment remains valid following updates to the inventory for disposal.

Criticality

The package inventories of fissile radionuclides (in particular U235 and Pu239) for UK ABWR ILW at 2150 are generally very small by comparison with those considered in the generic Post-closure Safety Assessment. It is therefore concluded that the addition of the UK ABWR wastes to a geological disposal facility would not increase the likelihood of a criticality event in a geological disposal facility.

Summary

The operational and decommissioning ILW from a UK ABWR 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 to be acceptable. Some issues have been noted which would be taken forward in future interactions with operators through the Disposability Assessment process, including the organic content of operational waste streams and impact on the long-term safety case, and the C-14 content of steels and its impact on risk from the gas pathway.

4.4 Summary of the Disposability of UK ABWR ILW

4.4.1 General

Taking into consideration the analysis of the wastes covered in Section 3.3, the waste container and wasteform properties discussed in Section 4.1, the performance of the waste packages during transport to and emplacement in a geological disposal facility discussed in Section 4.2 and the performance of the packages following sealing and closure of a geological disposal facility 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 Disposability Assessment process. These issues have been discussed in the results and implications sub-sections throughout Section 4, and are listed separately 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 ILW that would arise from operation of a UK ABWR. 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).

The total packaged waste volume (2,783 m^3) is small compared to legacy wastes (364,000 m^3). For the most active waste stream, Reactor Pressure Vessel Internals, the total activities of eight of the ten most active radionuclides are similar (within a factor of ten) to the equivalent waste streams from Sizewell B (Section 3.3.4, Table 7). Differences in the inventory comparison are within the bounds of uncertainty recognised for the Sizewell B waste stream. 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 I-129 content of Resins and the Co-60 content of Control Rods and Activated Metals are high. For the I-129, this was the result of assumptions regarding the failure of fuel pins applied in datasets for other reactor systems used to produce enhanced assessment inventories. Revised scaling factors might be applied in future consideration of the disposability of ILW from a UK ABWR. For Co-60, the pre-cursor concentration was derived from available conservative steel compositions, and could be addressed through the more representative actual steel compositions and use of low-cobalt steels in (e.g.) the UK ABWR control rods, monitoring probes and the pressure vessel.

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, RWM would expect to work with potential reactor operators and provide assessment of fully-developed proposals through the Disposability Assessment process.

The proposed operational ILW packages use standard RWM 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 cementitious 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 RWM standards and specifications.

4.4.4 Impact on Design

The potential impact of the disposal of UK ABWR operational and decommissioning ILW on the size of a geological disposal facility has been assessed. It has been concluded that the 'footprint area' required to dispose of ILW from a UK ABWR corresponds to approximately 45m of vault length for each UK ABWR (178m for a fleet of four reactors) for higher strength rock. For the illustrative fleet of four UK ABWR reactors, this represents no significant change in the overall footprint compared with current assumptions based on the inventory for disposal.

4.4.5 Transport Safety

The proposal to use RWM 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 as compliant with the dose-rate limits set out in the 2012 IAEA Transport Regulations [22].

The transport safety assessment has identified that further work is required to establish the physical form of H-3 and C-14, which dominate the gaseous activity during transport in the current assessment. The assessment notes that the prominent contributor to radiological risk for the ILW wastes in the groundwater pathway is the I-129 present in the Decommissioning Resin waste stream, however ass noted above, this is an artefact of the inventory enhancement process producing an overly conservative I-129 value. Separate evaluations demonstrated that the Hitachi-GE supplied I-129 data resulted in a groundwater pathway risk that was comparable to legacy wastes.

4.4.6 Operational Safety

The operational safety assessment for ILW from a UK ABWR 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 RWM has judged that this issue may be addressed through future refinement of the assessment inventory, especially the inventory of Co-60, and refinement of the assessment methodology, including a more detailed understanding of the release of radionuclides in gaseous form during fire accidents. Issues associated with Co-60, could be reduced through adoption of low-cobalt steels, as mentioned above, and may also be addressed by incorporating the effects of interim storage on dose rates during accident and normal operating conditions.

4.4.7 Environmental Considerations

No environmental issues that challenge the viability of the disposal of ILW from a UK ABWR have been identified.

4.4.8 Security and Safeguards

The ILW from operation of a UK ABWR is expected not to present any significant security or safeguards issues.

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 [71], based on the approach adopted for Disposability 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 a UK ABWR 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 mass loading of cellulose in the UK ABWR resin waste streams is about a factor of four higher than the average cellulose loading in the generic Post-closure Safety Assessment. This organic material could interact with neighbouring packages and result in deleterious effects on the solubility and sorption of key radionuclides such as U-238 which may be present in neighbouring waste packages. This issue would require further consideration by operators and RWM under a future Disposability Assessment interaction.

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

5 ASSESSMENT OF SPENT FUEL

This section presents the assessment of Hitachi-GE packaging proposals for spent fuel (described earlier in Section 3.4) against RWM's preliminary waste package specification [16] and disposal system specification [12]. 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, to allow for decay cooling of 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 a geological disposal facility, environmental issues, and security implications.
- Section 5.4 describes the assessment of the impact of spent fuel disposal packages on long-term safety following closure of a geological disposal facility;
- Section 5.5 provides a statement regarding the overall disposability of spent fuel from a UK ABWR.

For each component of the assessment, the report addresses the context (i.e. the required performance), and the results and the implications of the assessment. Issues identified through GDA Disposability Assessment for each component of the evaluation are listed in Appendix B and would be expected to be addressed by future operators through the Disposability Assessment process.

5.1 Interim Storage Period for Spent Fuel

Spent fuel contains a wide range of radionuclides with widely varying half-lives, 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 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. No specific assumptions have been made in this GDA Disposability Assessment on the approach to interim storage of spent fuel, since this is yet to be defined by Hitachi-GE.

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 to ensure compliance with heat constraints.

As described in Section 2.1, current disposal concept work is focused on a bounding concept for higher strength rock which 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 [16] currently specifies an upper limit on disposal package thermal output determined by a temperature constraint on the "near-field" of a geological disposal facility[15]. The current thermal constraints of 100°C, 125°C and 200°C that RWM applies to the illustrative disposal concepts in higher strength rock, lower strength sedimentary rock and evaporite respectively are based on international precedent, for example [72]. Therefore, heat transfer calculations conducted by RWM in support of the GDA Disposability Assessment have applied 100°C as a limit to the inner boundary of the bentonite.

A heat transfer model [73] has been used to calculate the temperature profile across the cast-iron insert, the disposal container, 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.

The estimation of the required interim storage period is dependent on the conservative assumptions made in developing the inventory for spent fuel (including those regarding burn-up and modelling of neutron fluxes that have been discussed in Section 3.4). In addition, there are several geological disposal facility variables that impact the thermal modelling used to estimate the required interim storage period, including:

- assumptions regarding the thermal conductivity of the host rock;
- assumptions regarding the depth of the facility; and
- assumptions regarding the spacing of waste packages, i.e. the expected design of a geological disposal facility disposal tunnels (e.g. spacing between waste packages), and the design of the disposal container and engineered barrier system.

The first two of these variables have been investigated in sensitivity calculations, in order to scope the range of uncertainty in the interim storage period.

These uncertainties could be reduced by further work, for example, through refinement of the assessment inventory, by taking into account the cooling of the spent fuel being stored prior to the end of the reactor operational period.

5.1.1 Results and Implications

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 RWM 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 consistency with other aspects of the GDA Disposability Assessment, the thermal calculations assumed burn-ups of 50GWd/tU and 60 GWd/tU.

Thermal Modelling Calculations

Table 31 summarises the results from the thermal modelling for the higher strength rock case. For this modelling, the assumed geological disposal facility depth is 650m, the container spacing is 6.5m and the disposal tunnel centre-to-centre spacing is 25m. In recognition of the uncertainty associated with the properties of potential higher strength rock settings in the UK, RWM has adopted a range of values for rock thermal conductivity that is believed to encompass the majority of likely values in the UK. This range extends from $2.2Wm^{-1}k^{-1}$ to $3.8Wm^{-1}k^{-1}$, and, these values are used in thermal modelling studies, in order to give some insight into the impact of the potential range of host rock thermal conductivities on spent fuel cooling times. In addition, to determine the cooling period to be

15 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 a geological disposal facility.

applied in inventory calculations, a central value of $3Wm^{-1}K^{-1}$ was used. Application of this central value thermal conductivity resulted in a cooling times of 61 years for the 50GWd/tU case and 78 years for the 60GWd/tU case.

Table 31 Thermal modelling results for higher strength rock

Outputs	50 GWd/tU		60 GWd/tU	
	High thermal conductivity 3.8 $Wm^{-1}K^{-1}$	Low thermal conductivity 2.2 $Wm^{-1}K^{-1}$	High thermal conductivity 3.8 $Wm^{-1}K^{-1}$	Low thermal conductivity 2.2 $Wm^{-1}K^{-1}$
Peak buffer temperature at 100 years cooling (˚C)	74	91	82	101
Cooling period required to achieve T<100˚C (years)	51	79	66	102

Table 32 summarises the results from the thermal modelling for the lower strength sedimentary rock case. For this modelling, the assumed geological disposal facility depth is 500m, the container spacing is 3m and the disposal tunnel centre-to-centre spacing is 27.25m. RWM has adopted a range of thermal conductivity representative of lower strength sedimentary rocks of $0.62Wm^{-1}k^{-1}$ to $3.2Wm^{-1}k^{-1}$, and, therefore, calculations have been made for both of these values. The values reported in Table 32 include values for 100°C to allow comparison with higher strength rock. The values for the 125°C thermal limit case are those on the bentonite/container interface, rather than at the mid-point of the buffer, and are therefore bounding.

Table 32 Thermal modelling results for lower strength sedimentary rock

Outputs	50 GWd/tU		60 GWd/tU	
	High thermal conductivity 3.2 $Wm^{-1}K^{-1}$	Low thermal conductivity 0.62 $Wm^{-1}K^{-1}$	High thermal conductivity 3.2 $Wm^{-1}K^{-1}$	Low thermal conductivity 0.62 $Wm^{-1}K^{-1}$
Peak buffer temperature at 100 years cooling (˚C)	78	130	87	149
Cooling period to achieve T<100˚C (years)	69	149	84	192
Cooling period to achieve T<125˚C (years)	49	106	62	126

Table 33 summarises the results from the thermal modelling for the evaporite rock case. For this modelling, the assumed geological disposal facility depth is 650m, the container spacing is 3m and the disposal tunnel centre-to-centre spacing is 21.5m. RWM has adopted a range of thermal conductivity representative of evaporite rocks of $3.5Wm^{-1}k^{-1}$ to $6Wm^{-1}k^{-1}$, and, therefore, calculations have been made for both of these values.

Table 33 Thermal modelling results for evaporite rock

Outputs	50 GWd/tU		60 GWd/tU	
	High thermal conductivity 6.0 $Wm^{-1}K^{-1}$	Low thermal conductivity 3.5 $Wm^{-1}K^{-1}$	High thermal conductivity 6.0 $Wm^{-1}K^{-1}$	Low thermal conductivity 3.5 $Wm^{-1}K^{-1}$
Peak buffer temperature at 100 years cooling (˚C)	78	99	86	110
Cooling period to achieve T<200˚C (years)	4	15	10	27

Thermal Modelling Conclusions

Based on a spent fuel waste package containing twelve UK ABWR fuel assemblies and adopting the spacing used in the illustrative designs for higher strength rock, it would require of the order of 50 and 100 years for the activity, and hence heat output, of the UK ABWR fuel to decay sufficiently to meet the existing temperature criterion. This period allows for both the range of predicted ABWR fuel burn-up (50-60GWd/tU) and the range of rock characteristics that may be encountered for a geological disposal facility at a depth of 650m.

The cooling time required to meet the temperature criteria in the lower strength sedimentary rock illustrative design has a greater range owing to a greater range in the thermal conductivity of the lower strength sedimentary host rocks that could be used to host a geological disposal facility. The cooling time required in lower strength sedimentary rocks is currently estimated to be of the order of 50 and 130 years. This range is for the same burn-ups as the higher strength rock case.

For the illustrative designs in evaporite host rocks, the cooling time required is estimated to be much shorter than for either higher strength rock or lower strength rock. This is because of the higher temperature criterion on disposal of spent fuel in evaporitic host rocks and the higher thermal conductivity of evaporitic rocks. Therefore, the cooling times are likely to always be the shortest for disposal of spent fuel in evaporite host rocks.

5.2 Spent Fuel Disposal Package Properties

5.2.1 Wasteform

Context

The provision of a 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 a geological disposal facility. 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 UK ABWR 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 UO_2 pellets within the spent fuel. The fraction of activity that is readily released upon initial 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 or gaseous 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

The fuel expected to be irradiated in ABWR stations is a UO_2-based fuel in the form of cylindrical pellets stacked on top of each other along the 'active' length of a fuel rod. This type of fuel is widely used in reactors worldwide and its characteristics are well understood.

The wasteform evaluation concluded that the behaviour of the fuel is likely to be similar to that of other, well-researched fuels, currently considered as disposable [74]. In particular work done funded by organisations like SKB (e.g. [75]) or Nagra (e.g. [76]) has shown that typical instant release fractions of key soluble, long-lived radionuclides such as Cs-135 and I-129 are usually similar or bounded by values of the Fission Gas Release (FGR), typically in the range 0.1-10% for fuels which have been irradiated at power for prolonged periods of time. High IRFs have been historically associated with high burn-up (including speculations on the effect of the 'high burn-up' rim structure), although ongoing research is indicating that other factors (in particular actual power rating) may also affect (possibly even more strongly) the IRF. The levels of burn-up and power rating envisaged in UK ABWR rectors are not dissimilar to the values currently used in some existing light-water reactor (LWR) stations, which are the focus of ongoing studies and for which ongoing work indicates IRFs towards the upper end of the range reported above [77]. The use of burnable poisons (Gd) is not expected to negatively influence the leaching behaviour of the fuel.

As packaging proposals develop, it will be important to gather additional information on specific characteristics of the fuel that may affect its leaching behaviour, including expected power cycles/maxima and resulting temperatures in the fuel pellets, any available data on FGR, any specific additive or impurity in the fuel that may result in the presence of specific radioisotopes (e.g. Cl-36 or C-14) in the fuel, and any information that may be available relatively to the degree of oxidation / stoichiometry expected in failed fuel.

Estimates for the IRFs for spent fuel to be applied in the GDA Disposability Assessment were based on the IRFs used in the 2010 generic Disposal System Safety Case (DSSC), based on information initially reported in the Radionuclide Behaviour Status Report [11]. For this assessment, given the higher burn-up expected in UK ABWR fuel and recent developments indicating that higher values of IRF can be expected for caesium and iodine for these fuels, modified values for the IRF of caesium and iodine have been adopted. These values along with those used in the DSSC are presented in Table 34.

Table 34 Recommended IRFs for UK ABWR spent fuel

Radionuclide	2010 DSSC IRFs		UK ABWR GDA Disposability Assessment IRFs	
	Reasonable Probability IRF	Pessimistic Probability IRF	Reasonable Probability IRF	Pessimistic Probability IRF
Ag-108m	1	1	1	1
C-14	0.15	0.55	0.15	0.55
C-l36	0.06	0.12	0.06	0.12
Cs-135	0.03	0.06	0.1	0.1
Cs-137	0.03	0.06	0.1	0.1
I-129	0.03	0.06	0.1	0.1

Radionuclide	2010 DSSC IRFs		UK ABWR GDA Disposability Assessment IRFs	
	Reasonable Probability IRF	Pessimistic Probability IRF	Reasonable Probability IRF	Pessimistic Probability IRF
Nb-94	1	1	1	1
Ni-59	1	1	1	1
Ni-63	1	1	1	1
Pd-107	0.002	0.01	0.002	0.01
Se-79	0.03	0.06	0.03	0.06
Sn-126	0.02	0.04	0.02	0.04
Sr-90	0.0025	0.01	0.0025	0.01
Tc-99	0.002	0.01	0.002	0.01

Whilst any protection provided by the cladding is not usually considered (at least quantitatively) in safety assessments, siting and geological disposal facility design will ensure that rates of corrosion will be relatively slow, providing additional confidence in the containment provided by the engineered barrier system. Additionally, and perhaps more importantly, a low corrosion rate of fuel assembly materials would result in slow rates of release of any activation products (e.g. C-14) that may be present in the assembly.

The wasteform evaluation concluded that the likely characteristics and anticipated behaviour of such fuel during periods of interim storage, operations of a geological disposal facility and after closure of the facility is likely to be similar to that of other fuels routinely irradiated in modern power stations worldwide and considered disposable.

Future interactions between RWM and operators of a UK ABWR under the Disposability Assessment process will require additional information regarding the chemical characteristics of the fuel (e.g. the use of any specific additives that may result in the presence of key, specific radionuclides), details of the likely power/temperature averages and maxima (both at rod and pellet scale), information about Fission Gas release expected/measured for these fuels and about future interim storage strategies, including any drying/containerisation processes and storage conditions.

5.2.2 Spent Fuel Disposal Package Performance

Context

Impact Performance

Four repository operational accident cases were considered in the spent fuel disposal package performance assessment [43]:

- impact in an axis vertical orientation onto a flat unyielding target after a free fall from 8m;
- impact onto a flat unyielding target after free topple from an upright position;
- impact onto a mild steel ledge mounted on a flat unyielding target, after free topple from an upright position; and

- impact in an axis horizontal orientation onto a flat unyielding target after a free fall from 5.5m.

Fire Performance

The purpose of the fire performance assessment is to determine the release of radionuclides subjected to a 1-hour, 1,000°C, fire during geological disposal facility operations.

Results and implications

Impact Performance

Drop tests have not been conducted on the UK ABWR spent fuel in a spent fuel disposal container. Based on the work in developing a disposal container transport container (DCTC) outline design [78], there would be no loss of containment and the disposal container would not sustain any significant damage in a transport impact accident. Therefore, it has been assumed that, for a spent fuel disposal container held within a DCTC, there is no release from the disposal container or DCTC in a transport impact accident. Based on finite element impact analyses that were carried out as part of the technical underpinning of the conceptual RWM disposal container designs [79], the only repository operational impact accident case that would result in any release is the ‘topple onto a ledge’. Therefore, the facility in which the proposed UK ABWR disposal containers will be operated will be designed such that this impact accident scenario could not happen. Therefore, an RF of zero is assumed.

Fire Performance

The release of radionuclides from a spent fuel disposal container was included in the generic release fractions assessment [25]. It was concluded that, in the event of being engulfed in a 1000 °C fire for 1 hour, the container would reach temperatures of around 870°C. However, this is well below the melting point of either the copper or steel from which the container would be constructed. It was therefore concluded that the integrity of the container would remain intact and that there would be no release of radionuclides.

The disposal container envisaged for packaging UK ABWR spent fuel is similar in design to that assumed in the generic release fractions assessment [25]. It is therefore concluded that, in the event of a fire during operations at a geological disposal facility, there would be no release of radionuclides.

In the event of a fire during transport, the disposal container will be further protected from the heat of the fire by being inside a DCTC. The temperature reached by the disposal container following a fire during transport will therefore be less than that experienced following a fire during operations at a geological disposal facility. The release fraction following a fire during transport will therefore also be zero.

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 UK ABWR spent fuel on the design of the disposal facility [45].

A key issue impacting design, safety and potentially siting of a geological disposal facility is the change in volume of host rock required in the event that spent fuel from a new build UK ABWR is disposed of alongside historical and currently arising wastes and/or materials, and waste and/or materials from other new nuclear build reactors. The implication of this can be estimated by considering the number of disposal tunnels required to dispose of the waste compared to current estimates for the inventory for disposal.

The Design Impact evaluation considered the impact on a geological disposal facility of a single UK ABWR based on the assumption that the spent fuel is packaged prior to consignment. The impact of a fleet of four UK ABWRs has also been considered [45].

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 [8]. RWM expects to revisit this design as implementation of geological disposal progresses, and RWM will review the design based on information relevant to a specific site and specific setting.

Results and implications

For a packaging assumption of twelve fuel assemblies per container, the 9,600 fuel assemblies would require 800 disposal containers. These would be placed in individual disposal holes within the deposition tunnels. This arrangement is the same as that adopted for historical and currently arising spent fuel and new nuclear build spent fuel in previous disposal assessments, and the length of the containers is bounded by designs for spent fuel already considered in the inventory for disposal.

Other design impacts associated with this change include [45]:

- the mass of a laden spent fuel disposal container is 25.1t; this exceeds the current design limit for the DCTC (which is 25t[16]), but issues with the design of the DCTC are known and it is expected that the mass limit of the DCTC will be changed in the future; nonetheless, it is necessary to check that this is feasible and for a suitable mass of a laden spent fuel disposal container for UK ABWR to be identified during future interactions between RWM and operators of a UK ABWR under the Disposability Assessment process; and
- the thermal output of the disposal container exceeds the current limit for a DCTC of 1,200W, however, reductions in the pessimisms associated with the estimation of the heat output are likely to overcome this issue.

Based on the thermal modelling done as part of the design impact assessment (see Section 5.1), disposal of spent fuel from a UK ABWR to a geological disposal facility in higher strength rock could commence from 2131. This would allow the spent fuel to be readily incorporated into a disposal schedule consistent with the current assumption of closure of a geological disposal facility commencing in 2190.

The potential impact of the disposal of UK ABWR spent fuel on the size of the geological disposal facility has been assessed. The industry ambition of 16GW of nuclear new build has been estimated previously to produce spent fuel containers that will fill approximately 202 disposal tunnels in a geological disposal facility in high strength rock. The assumed operating scenario for a single UK ABWR gives rise to an estimated 800 spent fuel disposal containers, requiring approximately 18 disposal tunnels for disposal in higher strength rock. For the illustrative fleet of four UK ABWR reactors, representing 5.40GW,

[16] Although the design limit is not met, the requirements for transport using a four-axle rail wagon (<90t) are met by this mass.

this would be equivalent to 72 disposal tunnels. This indicates that the required number of disposal tunnels is in agreement with the estimates for other new build reactors.

5.3.2 Transport Safety

Context

Based on the assumption that spent fuel will be packaged for disposal before dispatch to a geological disposal facility (Section 3.2), it follows that arrangements will be required to transport spent fuel packaged in disposal containers safely through the public domain. As described earlier (Section 3.4.2), RWM is planning the transport system that will be required to move all higher activity wastes from sites of interim storage through the public domain to a geological disposal facility. This will be achieved in the case of spent fuel by provision of a shielded transport container that meets the requirements of the 2012 IAEA Transport Regulations [22] as implemented by UK transport legislation.

The DCTC (described in Section 3.4.2) is the transport container concept developed by RWM for transport of spent fuel through the public domain [78]. 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 UK ABWR spent fuel for compatibility with the existing DCTC concept and against the generic Transport Safety Case [47].

Results and implications

External Dose Rates

The external dose rate from a loaded transport container (DCTC) has been calculated and compared to the limit of 0.1mSv/hr at 1m from the transport container specified in IAEA Transport Regulations for non-exclusive use [22]. For gamma radiation the dose rate is 0.1mSv/hr at 1m from the transport container, while for neutron radiation the dose rate is 0.02mSv/hr at 1m. The dose rate at 1m from the DCTC for UK ABWR spent fuel subject to a burn-up of 50GWd/tU is 7.66E-02mSv/h (dose rate calculated for 61 years cooling) and for UK ABWR spent fuel subject to a burn-up of 60GWd/tU is 7.60E-02 mSv/hr (dose rate calculated for 78 years cooling), both of which are less than the 0.1mSv/hr dose rate limit.

An additional conservatism in the spent fuel assessment inventory pertains to the management of the spent fuel following discharge from the reactor core. Hitachi-GE may build a large central pond to store spent fuel prior to packaging. This would provide an ability to “mix-and-match” fuel assemblies when packaging in disposal containers, i.e. mix older fuel with younger fuel to provide lower maximum heat outputs and dose rates from the packages. This management strategy has not been considered in this Disposability Assessment.

Gas Generation under Normal Conditions

The disposal container 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. Furthermore, any gaseous fission products developed during irradiation in a reactor (noble gases such as krypton and xenon) are likely to remain in stable quantities inside the cladding and, even if released in the container, would result in trivial amount of pressurisation (due to the very small amounts available). An additional source of pressurisation expected in spent fuels during disposal, however, is represented by any helium generated by α-decay. Given the very long timescales over which this process is likely to occur, any pressurisation is likely to occur only after substantial periods of time (100,000-1,000,000 years). For example, a study carried out to evaluate helium pressurisation in AGR and PWR fuel indicated trivial level of pressurisation for periods up to at least 1,000,000 years [80].

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 UK ABWR (see Section 5.2.2). 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 Disposability Assessment process, the operator will need to confirm zero or low release fractions from the disposal package in accident conditions through testing and/or modelling of the waste packages.

Heat Output

The GDA Disposability Assessment estimated that the heat output from the disposal container will be approximately 1.72kW for UK ABWR spent fuel subject to a burn-up of 50GWd/tU after 50 years cooling time, and 1.55kW for UK ABWR spent fuel subject to a burn-up of 60GWd/tU after 70 years cooling time[17]. Both of these values are below the 2.0kW envelope demonstrated to be acceptable in achieving safe handling of the DCTC during transport.

Transport Operational Risks

The additional transport movements associated with transport of UK ABWR spent fuel to a geological disposal facility have been compared with the generic transport safety assessment [46]. 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 RPPM [58]. No increase has been observed for accident risk since radioactive release in accident conditions is expected to be zero.

Criticality

Although nuclear fuel is most reactive prior to irradiation, fresh fuel is readily transported to reactor sites prior to use. Subsequent to irradiation, the increased irradiation anticipated for a UK ABWR would reduce the nuclear reactivity compared to spent fuel at the same initial enrichments from current PWRs. Furthermore, it has been reported that fresh fuel from the Swedish programme contained in a sealed (water-tight) disposal container would be sub-critical [81].

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 container, in particular water ingress. However, the DCTC is being developed to incorporate Multiple Water Barriers to mitigate the potential for water ingress. Criticality scenarios involving water leakage into the DCTC or disposal container therefore can be excluded.

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[17] These cooling times were selected to be consistent with the approximate cooling periods estimated by the thermal modelling presented in Section 5.1.

On the basis of these arguments, it has been concluded it should be possible to construct a criticality safety case for the transport of UK ABWR 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 Disposability Assessment process.

5.3.3 Operational Safety

Context

The Disposability Assessment process uses RWM's generic Operational Safety Case [51] and a series of generic Operational Safety Assessments [52, 53, 54, 55] to test proposed waste packages and to check compliance with assumed performance and accident consequence criteria. A similar approach has been adopted for the UK ABWR GDA Disposability Assessment.

The aim of the operational safety assessment is to examine the information supplied by Hitachi-GE, including any enhancement undertaken during earlier stages of the Disposability Assessment, to assess whether disposal of waste from the reactor under consideration does not undermine assumptions within the generic safety assessments.

When spent fuel packages arrive at a geological disposal facility site they would be transported underground in a DCTC to the transfer hall (the Underground Transfer Facility). Here, they would be transferred to a shielded deposition machine. The deposition machine would then transfer the disposal container to the disposal tunnel and emplace the waste.

At all times when operators may be present the waste inventory is kept behind shielding in the DCTC, the deposition machine or the Underground Transfer Facility. The spent fuel disposal package accident performance evaluation has concluded that there are no fault conditions that will lead to loss of containment. The operational safety assessment has therefore considered the operational safety implications of normal operations (doses to workers under normal conditions of operators) and criticality.

Results and implications

Doses to Workers under Normal Conditions

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. It would be expected that the shielding would be designed to maintain dose rates on the surfaces of these facilities to below 2.5μSv/hr.

For UK ABWR spent fuel packaged inside a disposal container, the dose rates at 1m from the surface of a DCTC for the burn-up value 60GWd/tU is 7.60E-02mSv/hr and for 50GWd/tU it is 7.66E-02mSv/hr, which is less than the limiting criteria under IAEA and UK radioactive material transport regulations of 0.1mSv/hr at 1m from the transport container. The dose rate at 3m from the disposal container is estimated to be 1.86E+00mSv/hr and 2.3mSv/hr respectively.

For the high burn-up scenario of 60GWd/tU, if it is assumed that an operator spends a total of 10 minutes in proximity to each transport container, the operational dose associated with a single container would be 12.6μSv. Six hundred and sixty-seven of these packages are anticipated over a 60-year operational lifetime. If it is conservatively assumed that ten percent (67) of total waste packages are received in one year, the operational dose contribution could be approaching 0.84mSv.

For the corresponding consideration for the average burn-up scenario of 50GWd/tU, the operational dose associated with a single container would be 12.8μSv. Eight hundred of these packages are anticipated over a 60-year operational lifetime. Based on the assumption of ten percent (80) of total waste packages being received in one year, the operational dose contribution could be 1.02mSv.

These doses are of a similar order to the current Basic Safety Objective (BSO) of 1mSv for Target 1 '*Normal operation - any person on the site*' from the ONR's Safety Assessment Principles (SAPs) [82]. However, given the uncertainties regarding the package inventories and the lack of information regarding the design of the transport container, this is not necessarily considered to be a significant issue at this early stage. However, further assessment of the dose to workers under normal conditions will be required during future interactions under the Disposability Assessment process when uncertainties in the spent fuel inventories are reduced, and the design and operational safety assessment of the spent fuel handling and emplacement system has been progressed further.

Criticality

The disposal packages containing UK ABWR spent fuel would be handled and emplaced 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 a geological disposal facility.

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 container, 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 geological disposal facility operations. Criticality scenarios involving water leakage into the DCTC or disposal container therefore can be excluded.

Based on the above, it may be concluded that a criticality safety case for the handling of disposal packages containing UK ABWR spent fuel during operations at a geological disposal facility 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 Disposability Assessment process.

5.3.4 Environmental Evaluation

Context

The environmental evaluation 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 a geological disposal facility.

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 spent fuel. Further assessment, including consideration of site-specific effects, would be required in the future to meet Environmental Impact Assessment (EIA) requirements.

It is assumed that the environmental implications of waste treatment, packaging and storage prior to disposal will be addressed by the waste producers – including by discharging their obligations under the relevant UK EIA Regulations.

The environmental implications of disposing of UK ABWR wastes will, to a large extent, be determined by the design changes (to both a geological disposal facility and its associated transport system) required to accommodate the wastes. The environmental evaluation therefore draws on the Design Impact Assessment for UK ABWR wastes (Section 5.2.1).

Results and implications

Based on the submitted proposals and in terms of generic (non-radiological) environmental effects, the environmental evaluation concluded that the wastes from a single UK ABWR reactor should be considered as disposable. Current assumptions for treatment and packaging of the wastes are consistent with those made in generic design and assessment work. The proposals present no novel treatment and packing options that are likely to give rise to environmental (or socio-economic) effects that go beyond the scope of RWM's current generic assessment work.

The disposal of UK ABWR wastes will result in a change in the underground footprint of a geological disposal facility to accommodate the additional wastes and an associated increase in excavated rock spoil, although this may not be significant in overall terms. This could result in additional environmental and socio-economic effects (both positive and negative) – largely associated with the transport of additional or reduced rock spoil off-site for disposal and with an extended operational period for the facility.

5.3.5 Physical Protection and Safeguards Evaluation

Context

The objectives of the physical protection evaluation were to determine whether the physical protection requirements for the wastes that could arise from the operation and decommissioning of an ABWR would be consistent with regulatory requirements and with plans for the transport of waste packages to a geological disposal facility.

Nuclear Material is defined as *"(a) any fissile material in the form of uranium metal, alloy or chemical compound, or of plutonium metal, alloy or chemical compound; or (b) any other fissile material which may be prescribed by regulations made by the Secretary of State"* [61].

The quantity of Nuclear Material contained within waste packages is required to be controlled such that they can be transported subject to standards of physical protection no higher than those defined for the transport system.

The Nuclear Industries Security Regulations (NISR) [62]) lay down the requirements for security of nuclear premises, security of transport of nuclear material and security of sensitive nuclear information. The Office for Nuclear Regulation (ONR) has issued National Objectives, Requirements and Model Standards (NORMS) for the protective security of civil licensed nuclear sites, other nuclear premises and nuclear material in transit [63] to support implementation of the NISR.

The security standards in NORMS are offered as a benchmark (i.e. Model Standard) to reflect internationally agreed recommendations on the physical protection of Nuclear Material published by the IAEA [64]. These standards also reflect the United Kingdom's obligations under the *Convention on the Physical Protection of Nuclear Materials* and its commitments under the Nuclear Suppliers Group Guidelines and Plutonium Management Guidelines.

The NORMS specifies mass limits for the quantities of Nuclear Material and ILW/LLW containing Nuclear Material that can be transported with four 'categories' of physical protection (Categories I to IV, Category I being the most restrictive). The generic Disposal System Technical Specification states the assumption that a geological disposal facility for LLW, ILW, HLW and spent fuel will be a Category II facility as a minimum under the current system [12].

The categorisation of Nuclear Material depends on whether or not the Nuclear Material can satisfy the NORMS definition of 'waste' which requires the material to be:

> *'….Nuclear Material arising from operations which have been or are to be discarded as Intermediate or Low Level Waste………..provided that:*
>
> *a. the waste is in solid form including sludges without free liquid;*
>
> *b. the Nuclear Material is well dispersed and is not readily separable or recoverable;*
>
> *c. the mass of the Nuclear Material content is less than 1% of the total mass of the waste;*
>
> *d. the waste is stored or transported within the UK.'*

Furthermore, the waste must be transported in '*Concreted waste disposal containers'* which '*include containers where the waste is immobilised in a cementitious grout'*.

With respect to safeguards, all Nuclear Material is subject to safeguards, unless the safeguards status can be terminated. Termination can be achieved, following agreement between the site operator and Euratom, on the grounds of low Nuclear Materials concentration (e.g. 0.1%w/w for DU, or 4ppm for Pu), or if the Nuclear Material is in a form unsuitable for further use (e.g. finely dispersed in a cement matrix, or as widely spread surface contamination).

Results and implications

The maximum Nuclear Material of any of the spent fuel waste packages will be ~2t, comprising mainly uranium with an enrichment of 0.34% containing up to 20kg of plutonium and traces of thorium.

Reference to the NORMS suggests that these waste packages, by virtue of their plutonium content, would require physical protection during transport to the standards defined by Category I. However, if the external dose rate of the fuel at 1m from its surface exceeds 1Gy/h, and the fuel could be defined as 'irradiated material', Category III standards of protection would be adequate.

In addition, under current definitions presented in the NORMS [63], the spent fuel from a UK ABWR is not deemed to be waste because:

- the Nuclear Material is not well dispersed and is assessed to be readily separable or recoverable;
- the mass of the Nuclear Material content is greater than 1% of the total mass of the waste; and
- the material is not planned to be transported in 'Concreted waste disposal containers' which 'include containers where the waste is immobilised in a cementitious grout'.

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 at a geological disposal facility. Furthermore, it can be assumed that the presence of spent fuel in a geological disposal facility will result in a range of safeguards-related measures being applied to the geological disposal facility 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 a geological disposal facility resulting from the disposal of spent fuel from a UK ABWR 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. This will provide the first indication of the extent of the measures that could be applied to a geological disposal facility in the UK.

There are no safeguards-relevant characteristics present in the UK ABWR spent fuel that are likely to make it significantly different to spent fuel from any other civil reactor type. However, in common with other spent fuel, the safeguards status of spent fuel is unlikely to be capable of termination and current thinking is that spent fuel will remain 'on inventory' after packaging and possibly even after disposal. RWM therefore recommends that Hitachi-GE consider the extension of safeguards provisions through disposal, particularly for spent fuel.

5.4 Post-closure Safety

Context

As described earlier, the post-closure safety assessment is one component of the ESC which is required to demonstrate safety of the disposal system in the long-term following backfilling, sealing and closing of a geological disposal facility. 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 container, the buffer and the geological environment. Understanding 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' GRA [71], 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 RWM on the disposal of spent fuel in the UK has included the development of a generic post-closure safety assessment [68]. The post-closure safety assessment of UK ABWR spent fuel was undertaken by considering whether the disposal of UK ABWR spent fuel would challenge any of the conclusions from this previous assessment. The assessment considered potential radiological and non-radiological impacts due to the groundwater and gas pathways, human intrusion, chemotoxic species, and criticality. Quantitative assessment of risks to humans from the groundwater pathway was conducted using the GoldSim [69] code.

As noted above, the post-closure assessment is a component of the ESC, development of which is at an early generic stage. The assessment is based on a "illustrative" geological disposal facility design and host environment. This also includes assumptions regarding the nature of the geology and hydrogeology pertaining to the near-field and far-field environments and regarding the biosphere. The assessment of the performance of the higher strength rock illustrative geological disposal facility design under development by RWM 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, and was therefore used as the basis for the post-closure performance of UK ABWR spent fuel.

Results and implications

The disposability assessment has considered how spent fuel packages would evolve in the very long-term future, recognising that radionuclides would be released only subsequent to a breach in a disposal container. Subsequent to any container failure, which is likely to occur after many thousands of years, allowing a large fraction of the original inventory to decay to very low levels, the remaining 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 a geological disposal facility. 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 historical and currently arising ILW (which provides the bounding case for groundwater flow rate and return time).

In the GDA Disposability Assessment for the UK ABWR, the quantitative assessment considered spent fuel from a single reactor packaged in disposal containers each containing twelve spent fuel assemblies all irradiated to either 50 GWd/tU or 60 GWd/tU [69].

Groundwater Pathway

The risks from the UK ABWR spent fuel are comparable with, but lower than, the risks obtained for all of the historical and currently arising spent fuel in the 2010 generic post-closure safety assessment. The key radionuclides in terms of contribution to risk in both cases are I-129 and Cs-135. The disposal container inventories for these radionuclides for ABWR exceed those considered in the 2010 generic post-closure safety assessment, but not to a significant degree. In this sense, the UK ABWR disposal container activities for I-129 and Cs-135 are considered to be consistent with the container inventories for PWR spent fuel considered in the 2010 generic post-closure safety assessment.

If more than one reactor were constructed, then the risks from the ABWR wastes would increase in proportion.

Gas Pathway

No formal assessments of the consequences of gas generation from spent fuel have been undertaken by RWM. Therefore, the GDA Disposability Assessment undertook a scoping calculation, which, building on the understanding of ILW metallic waste streams, considered the potential consequences of the release of C-14 following loss of integrity of spent fuel disposal containers[18]. The assessment made several pessimistic assumptions in order to scope the maximum possible impact from release of C-14, including:

- all of the spent fuel containers failed in the period 50,000-100,000 years following closure;
- that all of the released C-14 is released as radioactive methane; and
- there is no decay of C-14 during the release period.

[18] The GPA03 gas assessment concluded that other gases generated in a geological disposal facility would have small dose contributions relative to C-14-substituted methane, and this assumption has also been adopted for the GDA Disposability Assessment. Other radioactive gases, such as Rn-222, are assessed as part of the groundwater pathway.

Using these assumptions annual risks were estimated as 1.2E-11 for release over the plan area of the vaults, and 4.8E-10 for release over 10,000m^3. These risks are significantly below the risk guidance level, and, on the basis of this scoping calculation, the risks from C-14 originating from UK ABWR spent fuel are of no concern given the current assumptions regarding post-closure performance.

Results for Human Intrusion

The siting process adopted by Government [70] 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' GRA requirements [71] for human intrusion requires that any practical measures to reduce the risk from human intrusion are implemented in a geological disposal facility and that potential risks from human intrusion are optimised. These requirements do not relate, therefore, to the fundamental disposability of spent fuel.

Chemotoxic Assessment

Consideration of the potential impact of chemotoxic elements contained in the UK ABWR spent fuel was undertaken by comparing the mass of chemotoxic species in the spent fuel with the masses considered in the 2007 screening assessment [67]. With the exception of zirconium, the ratio of the masses of chemotoxic species in the UK ABWR spent fuel to the totals in the 2007 screening assessment were small (approximately 1:1,000 to 1:10,000). The ratio of the mass of zirconium in the UK ABWR spent fuel to the mass of zirconium in the 2007 screening assessment was approximately 0.5, but zirconium is considered to be of low concern in chemotoxic assessment of a geological disposal facility. The UK ABWR spent fuel is therefore unlikely to significantly increase the chemotoxic hazard presented by a geological disposal facility.

Further confirmation should be sought during future interactions under the Disposability Assessment process that all chemotoxic species have been identified in the UK ABWR ILW, and also that the 2007 screening assessment remains valid following updates to the inventory for disposal.

Criticality Safety

UK ABWR spent fuel would contain about 23.2kg of fissile material per disposal package for the 50GWd/tU case and 17.6kg of fissile material per disposal package for the 60GWd/tU case. The inventory of fissile material per disposal container for Sizewell B PWR spent fuel has been estimated as approximately 17.6kg and that for UK AGR spent fuel as 24.1kg [83]. Therefore, the quantity of fissile material in UK ABWR spent fuel is broadly similar to the quantity in packages containing UK AGR and Sizewell B PWR spent fuel.

A recent study undertaken by RWM has shown that the likelihood and consequences of a criticality event in a geological disposal facility are low for PWR and AGR spent fuel [84]. Thus, as the inventories of fissile radionuclides in UK ABWR disposal containers are similar to those in the PWR and AGR disposal containers considered in the 2010 generic post-closure safety assessment, the conclusion of low likelihood of a criticality event will still hold.

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

5.5 Summary of the Disposability of UK ABWR 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 a geological disposal facility discussed in Section 5.3 and the performance of the packages following sealing and closure of a geological disposal facility discussed in Section 5.4, packages containing spent fuel from a UK ABWR have been judged to be potentially disposable.

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

5.5.2 Inventory

The GDA Disposability Assessment for the UK ABWR has shown that the principal radionuclides present in UK ABWR spent fuel are the same as those present in current and historic reactors, and, in particular, are consistent with the anticipated arisings from the existing PWR at Sizewell B. This conclusion reflects that both the UK ABWR and PWRs are light-water reactors, and the expectation that similar operating regimes would be applied.

Hitachi-GE has indicated that the GDA Disposability Assessment for the UK ABWR should assume that the reactor would operate to achieve a maximum fuel pin burn-up of 65 GWd/tU, which corresponds to a maximum fuel assembly average burn-up of 60 GWd/tU and an average burn-up of 50 GWd/tU fuel assembly. This burn-up is slightly higher than that for the existing PWR at Sizewell B, which are assumed to be 45GWd/tU for stocks and 55GWd/tU for future arisings.

Although the length of UK ABWR spent fuel is similar to the length of PWR spent fuel from Sizewell B (approximately 4.5m versus 4.0m respectively), the cross-sectional area is somewhat different (approximately 0.14x0.14m versus 0.21x0.21m respectively). As a result, there will be a larger number of UK ABWR spent fuel assemblies in each disposal container, 12 compared to 4, and there will be a greater mass of uranium, 2.158tU for UK ABWR spent fuel disposal containers and 1.834tU for PWR disposal containers.

As illustrated in Table 11, the differences in disposal container contents are consistent with the estimated radionuclide activity per disposal container for the UK ABWR, which is slightly greater than that of the Sizewell B fuel. However, the detailed methodology used to develop the assessment inventory has led to some significant differences in the radionuclide content of spent fuel from an UK ABWR compared to that from Sizewell B, in particular the use of pessimistic chlorine and tin concentrations in precursor materials. However, concentrations of Ni-59 and U-233 were assessed to be lower for the UK ABWR owing to lower quantities of Inconel in the UK ABWR fuel assembly compared to a Sizewell B assembly, and the difference in the assumed precursor concentration of thorium in spent fuel and precursor concentration of thorium in the spent fuel cladding.

5.5.3 Waste Packages

The GDA Disposability Assessment for the UK ABWR was based on an assumption that spent fuel would be overpacked for disposal. Under this concept, spent fuel would be overpacked into durable disposal containers 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 container material remains to be confirmed, the assessment has considered the potential performance of copper containers, which would be expected to provide very long term containment in suitable geological environments.

The materials used as part of the engineered barrier system, and the characteristics of the host rock, will affect the thermal criteria used to determine the acceptability of the heat output from waste packages consigned for disposal. In the current generic phase of the programme, generic thermal criteria are used to determine approximate cooling times required before disposal of spent fuel. Different thermal criteria are applied in the illustrative disposal concepts for different host rocks. In higher strength rock, the temperature criterion requires that the temperature of the inner surface of the bentonite buffer should not exceed 100°C. In lower strength sedimentary rock, the temperature criterion is that the buffer temperature should not exceed 125°C at its mid-point. In evaporites, the temperature criterion is that the temperature of the host rock should not exceed 200°C.

Based on a spent fuel waste package containing twelve UK ABWR fuel assemblies and adopting the spacing used in the illustrative designs for higher strength rock, it would require between 50 and 100 years for the activity, and hence heat output, of the UK ABWR fuel to decay sufficiently to meet the existing temperature criterion. This period allows for both the range of predicted ABWR fuel burn-up (50-60GWd/tU) and the range of rock characteristics that may be encountered for a geological disposal facility at a depth of 650m.

The cooling time required to meet the temperature criteria in the lower strength sedimentary rock illustrative design has a greater range owing to a greater range in the thermal conductivity of the lower strength sedimentary host rocks that could be used to host a geological disposal facility. The cooling time required in lower strength sedimentary rocks is currently estimated to be between 50 and 130 years. This range is for the same burn-ups as the higher strength rock case.

For the illustrative designs in evaporite host rocks, the cooling time required is estimated to be less than 40 years. This is because of the higher temperature criterion on disposal of spent fuel in evaporitic host rocks and the higher thermal conductivity of evaporitic rocks. Therefore, the cooling times are likely to always be the shortest for disposal of spent fuel in evaporite host rocks.

These cooling times are dependent on a number of uncertainties, in particular the conservative assumptions made in developing the inventory for spent fuel, the uncertainty in the thermal conductivity of the host rock of a specific site, and the details of the underground design (e.g. package spacing). These uncertainties could be reduced by further work, for example, through refinement of the assessment inventory, by taking into account the cooling of the spent fuel being stored prior to the end of the operational period, or by consideration of alternative geological disposal facility designs.

The consequences for cooling times can also be managed by consideration of alternative container and geological disposal facility designs.

The disposal container will provide full containment for an extended time period, allowing a large fraction of the radioactive inventory to decay to very low levels. The fuel can only be leached after the containment has been breached. The loss of radionuclides from spent fuel is characterised by an initial 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 container failure and is influenced by the properties of the spent fuel. The increased irradiation of the higher burn-up UK ABWR fuel could increase the IRF as compared to that for lower burn-up fuel. Available information on the performance of higher burn-up fuel has been used to provide suitably conservative IRFs for the assessment.

5.5.4 Impact on Design

An assessment has been made of the potential impact of the disposal of UK ABWR spent fuel on the size of a geological disposal facility. The 16GW of nuclear new build has been estimated to produce spent fuel containers that will fill approximately 200 disposal tunnels in a geological disposal facility in high strength rock. The assumed operating scenario for a single UK ABWR gives rise to an estimated 800 spent fuel disposal containers, requiring approximately 18 disposal tunnels for disposal in higher strength rock. For the illustrative fleet of four UK ABWR reactors, representing 5.40GW, this would be equivalent to 72 disposal tunnels. This indicates that the required number of disposal tunnels is in agreement with the estimates for other new build reactors.

5.5.5 Transport Safety

RWM is planning for the transport of packaged spent fuel to a geological disposal facility. Development of designs of suitable reusable shielded transport overpacks has commenced, although is at an early stage of development. Consequently, although the UK ABWR 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 current assumptions regarding handling and emplacement of spent fuel in a geological disposal facility. The disposal container 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 containers in regions of the disposal facility where the disposal container 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 UK ABWR 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 issues have been identified that challenge the viability of the disposal of spent fuel from a UK ABWR.

5.5.8 Security and Safeguards

No security or safeguards issues have been identified for UK ABWR spent fuel in addition to those already recognised for legacy spent fuel.

5.5.9 Post-closure Safety

The GDA Disposability Assessment has considered how spent fuel disposal packages would evolve in the very long term following closure of a geological disposal facility, recognising that radionuclides would be released only subsequent to a breach in a disposal container. Subsequent to any container 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 and the environment.

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 container failure and is influenced by the properties of the spent fuel. The increased irradiation of the higher burn-up UK ABWR fuel could increase the IRF as compared to that for lower burn-up fuel. Available information on the performance of higher burn-up fuel has been used to provide suitably conservative IRF's for the 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 would need to be consistent with meeting the regulatory risk guidance level, this assessment assumed the same site characteristics as assumed for the existing RWM generic assessment. On the basis of the information provided by Hitachi-GE and conservative calculations of spent fuel waste package performance, it was calculated that the spent fuel from a fleet of four UK ABWR reactors would give rise to an estimated risk below the risk guidance level.

The risks calculated for the disposal of spent fuel reflect the assumed performance of the proposed packaging options. Sensitivity analysis has demonstrated that while the calculated risk would be influenced by the container material performance, coupled with the performance of other engineered barriers and the geological barrier, the risk was calculated to be below the regulatory guidance level. This outcome is insensitive to any impact that the high burn-up experienced by the fuel assemblies would have on the IRF.

RWM recognises that the performance of disposal containers will be an important element of a safety case for the disposal of spent fuel. Consequently, it is anticipated that RWM will continue to develop container designs, including the designs of containers for UK ABWR spent fuel, with the intention of substantiating the continued robustness of current assumptions and tailoring the designs to whatever site is ultimately identified.

6 CONCLUSIONS

RWM has undertaken a GDA Disposability Assessment for the higher-activity wastes and spent fuel expected to arise from the operation of a UK ABWR. 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 RWM by Hitachi-GE and discussed in detail. This information has been used to assess the implications of the disposal of the proposed waste packages against the waste package standards and specifications developed by RWM, and the supporting safety assessments for a proposed geological disposal facility. The safety of transport operations, handling and emplacement at a geological disposal facility in the UK, 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.

RWM has concluded that sufficient information has been provided by Hitachi-GE to produce valid and justifiable conclusions under the GDA Disposability Assessment. RWM has concluded that ILW and spent fuel from operation and decommissioning of a UK ABWR should be compatible with plans for transport and geological disposal of higher-activity wastes and spent fuel. It is expected that these conclusions would be supported and substantiated by future refinements of the 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, it is expected that more specific and detailed packaging proposals would be assessed, and potentially endorsed, through the established Disposability Assessment process for assessment of waste packaging proposals.

The GDA Disposability Assessment for the UK ABWR has not identified any significant issues that challenge the fundamental disposability of the wastes and spent fuel expected to be generated from operation of such a reactor. This conclusion is supported by the similarity of the wastes to the expected arisings from the existing PWR at Sizewell B. Given a disposal site with suitable characteristics, the wastes and spent fuel from the UK ABWR are expected to be disposable.

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Appendix A: The Disposability Assessment Process

Introduction

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

In the case of higher activity waste coming forward from the UK ABWR it is expected that the GDA Disposability Assessment commissioned by Hitachi-GE 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 Disposability Assessment interactions.

Disposability Assessment Stages

Disposability Assessment 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 RWM Assessment will in addition to the Disposability Assessment, include RWM'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.

Disposability Assessment Bibliography

The Disposability Assessment process is well established and is supported by a suite of published guidance that operators will find helpful in undertaking Disposability Assessment interactions with RWM. 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.

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[19] ONR/EA/SEPA/NRW, *The Management of Higher Activity Radioactive Waste on Nuclear Licensed Sites, Guidance from the ONR, EA, SEPA and NRW to Nuclear Licensees*, Rev. 2, February 2014

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Copies of WPSGD are available on request from RWM or can be downloaded from NDA Publications:

http://www.nda.gov.uk/publications/

Appendix B: Issues to be Addressed during Future Disposability Assessment 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 Disposability Assessment 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 ABWR 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 Disposability Assessment process, as summarised in Appendix A. A range of information and guidance has been developed by RWM, describing the requirements of the Disposability Assessment 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 Disposability Assessment for the relevant wastes or materials. However, this Appendix should not be assumed to represent a comprehensive basis for fulfilling the requirements of Disposability Assessment.

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

The following key issues for ILW are identified in the assessment:

1. The optimum time for disposal of the ILW. In particular, Hitachi-GE has proposed disposing of the wastes shortly after they arise. For some of the waste streams, this raises concerns in meeting transport limits and operational limits at the GDF. These could be addressed by a period of decay storage for the relevant wastes.
2. Hitachi-GE proposed that the RPV decommissioning wastes were packaged in 4m boxes. The evaluations found that a significant period of decay storage would be required before some of the wastes from this waste stream could be transported and placed in the proposed GDF if these containers were used. It was therefore recommended that these wastes should be placed in $3m^3$ boxes and transported in Standard Waste Transport Containers.
3. The control rods in the ABWR design differ from those in the previously assessed PWR designs where the potential exists to dispose of them with the spent fuel. In the case of the ABWR, the control rods, both hafnium and boron carbide variants, are separate from the fuel assemblies and are proposed to be disposed of as ILW. The nature of these wastes is inherently challenging and they will require a period of decay storage prior to Hitachi-GE's proposal for grout encapsulation in $3m^3$ boxes. While they raise no insurmountable issues precluding disposal, they will need to be subject to further assessment as the disposal plans are further developed.

In more detail, issues identified for further consideration in any future submission for UK ABWR ILW wastestreams are:

Resins and Cruds

4. Based on experience from Sizewell, RWM is of the view that the conditioning factor of 3 that has been applied for the resins may be optimistic; a conditioning factor of 10 may be more appropriate. More information and substantiation of the conditioning factors will be required during further interactions under the Disposability Assessment process.
5. Zinc (added for water chemistry control) could be incorporated into crud waste streams (e.g. if it plates out on steel surfaces) instead of, or as well as, being taken up by ion exchange resins. The zinc could potentially act as a cement set retardant in crud and resin wasteforms if present in sufficient quantities. Therefore, future interactions under the Disposability Assessment process should evaluate this potential route for zinc contamination of the crud and resins.
6. Information on the types of resins present in the wastes, and discussion of the expected degradation products, their potential for producing complexants and their impact on wasteform properties and radionuclide behaviour would be required as part of future submissions.
7. The Decontamination Resins waste stream inventory used in the assessment has a relatively high fissile content per packages, Exceeding screening levels and not being declared fissile excepted packages. This would require further evaluation in any future Disposability Assessment.
8. In future interactions under the Disposability Assessment process a method for calculation of the maximum package inventories for Cruds and Resins should be proposed by the operator.

Control Rods (Hafnium and Boron Carbide)

9. The Hitachi-GE submission states that 40 control rods will be packed in each 3m^3 box. Given the high levels of activity and heat generated by the control rods, RWM advises that to facilitate grout encapsulation and cooling, the control rods should be located in the boxes using internal frames. If such furniture is used to locate size reduced control rods, RWM estimates 15 control rods can be placed in each box. On this basis, the currently assumed packing density of control rods is considered optimistic. Future operators would need to consider this.
10. Hf-178n is not modelled by ORIGEN, but is a significant contributor to dose at short timescales (half-life, approximately 31 years), and, therefore, should be included in any future inventories for the UK ABWR.
11. The assumption in this assessment that control rod metals contain 0.26% cobalt, leads to relatively high activities for Co-60 in the waste package inventories. The estimated activities can, in certain cases, challenge the limits on transport included in the IAEA Transport Regulations [21], and the assumptions in RWM's operational safety case. In future, RWM would expect to work with a future operator to reduce pessimisms in the inventories for control rods and activated metals. This might include consideration of the steel alloys to be used in the UK ABWR, for example, consideration of low-cobalt steel for the control rods.
12. The methods for size reduction of control rods and activated metals should be described in a future submission. In particular, these should define how boron carbide control rods are to be cut without release of the powder .

[21] International Atomic Energy Agency, *Regulations for the Safe Transport of Radioactive Material*, 2012 Edition, Specific Safety Requirements No. SSR-6, 2012.

13. Future operators of a UK ABWR should give consideration to distributing the highly irradiated tips of control rods with lower activity parts of the size-reduced control rods.

Activated Metals

14. For some of the steels, Hitachi-GE had provided good information on the steel compositions, but for some steels no compositions had been provided. For the assessment, the composition of Type 304 steel had been used to fill gaps and to ensure that the inventory was pessimistic. The steel used in the Activated Metals is unlikely to be Type 304, and a more corrosion-resistant metal is likely to be used by reactor operator. Future submissions should provide more detail on the composition of the steels to be used for each component.
15. The neutron sources included within the Activated Metals waste stream does not include antimony. This is a common element in modern neutron sources, although it is acknowledged that different sources may be used in the UK ABWR when it is implemented. Similarly, the monitoring probes may include fission chambers containing uranium. Although it is unlikely that the monitoring probes will contain significant quantities of uranium, any uranium within the probes should be reported in more detailed disposability assessments.
16. Antimony and beryllium, which may be present in the Activated Metals wastes, may present chemotoxic hazards. Such materials are unlikely to be present in large amounts, but might lead to special consideration being required for the relevant waste packages. The possibility of such materials being present in the waste will need to be considered in future interactions under the Disposability Assessment process.
17. The number of Mixed Metal ILW packages is based on a packing density of 8.25t of raw waste per package. This packing density is considered unlikely to be feasible on volume grounds and should be reassessed in future interactions under the Disposability Assessment process.
18. The heat output from the Mixed Metal waste stream exceeds transport limits at the proposed time of disposal. This would require further consideration in future Disposability Assessments.
19. The Disposability Assessment has not considered removal of metal items from storage baskets, but it is feasible that the storage baskets will be packaged with Activated Metals. This would reduce the dose. The management of Activated Metals including cutting for size reduction prior to packaging and the consequent impact on package inventories should be considered in more detailed interactions under the Disposability Assessment process.

General issues

20. No details on the presence of toxic or hazardous materials in the wasteforms are given in the submission. These would be required in any future submission.
21. Details need to be provided on the use of in-box furniture and the resulting residual void space in Decommissioning ILW packages at more detailed stages of the Disposability Assessment process.
22. Details of specific grouts, their properties and formulation development, will be required in future Disposability Assessment submissions.
23. Hitachi-GE has stated that there will be no miscellaneous contaminated items that would be classified as ILW. If it remains an assumption for future disposability assessments, this will need to be stated in the design and operation protocols, as some reactors do produce ILW contaminated items during operations. Potential

future operators need to be aware that this is a potential route for generation of operational ILW and should confirm that such material is to be consigned as LLW.

B.2 Issues Relating to Spent Fuel

1. The storage of spent fuel in water ponds means that drying techniques will need to be put in place to avoid the potential for internal pressurisation of storage/disposal containers and to ensure that they would comply with existing transport regulations.
2. Storage conditions will need to be managed to maintain integrity of the fuel assembly (particularly the fuel cladding) and any storage/disposal container during storage operations. If a wet storage strategy were to be implemented, a key requirement would be to maintain conditions to preserve the integrity of any stainless steel components (e.g. tie bars). If a dry storage regime were implemented, control of temperature and relative humidity would be required to minimise the potential for degradation (e.g. by hydride embrittlement) of fuel assembly components and any disposal container.
3. RWM recommends that a future operator considers the extension of safeguards provisions through to disposal, particularly for spent fuel, and considers, working with RWM, whether and how the safeguards status of spent fuel will be terminated.
4. Further confirmation would be sought during future interactions under the Disposability Assessment process that all chemotoxic species have been identified in the UK ABWR SF.