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NDA Technical Note no. 11261814

Geological Disposal

Generic Design Assessment: Summary of Disposability Assessment for Wastes and Spent Fuel arising from Operation of the UK EPR

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1 Introduction

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

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

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

The principal conclusions of this GDA Disposability Assessment are presented in this Summary Report, together with the details of the wastes and spent fuel and their characteristics applied in the assessment. More comprehensive details of the information supplied to RWMD by EdF/Areva, measures taken by RWMD to supplement this information, assessment methods and the detailed conclusions of the GDA Disposability Assessment are provided in a separate Assessment Report for the EdF/Areva EPR.

The GDA Disposability Assessment process is summarised in Appendix A and comprises three main components: a review to confirm waste and spent fuel properties; an assessment of the compatibility of the proposed disposal packages with concepts for geological disposal; identification of the main outstanding uncertainties and associated research and development needs relating to the future disposal of the wastes and spent fuel. A summary of the radionuclide assessment inventories for ILW and spent fuel derived for the purposes of this GDA Disposability Assessment is set out in Appendix B.

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

2 Nature of the Higher Activity Wastes and Spent Fuel

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

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

- Ion exchange resins;
- Spent cartridge filters (ILW) – higher activity filters from the reactor primary circuit;
- Spent cartridge filters (LLW and ILW)[1] – other designs of filter, typically with lower activity;
- Operational wastes with a dose rate >2mSv/hr – those general operational wastes that would be categorised as ILW, as determined by dose-rate;
- Wet sludges;
- Evaporator concentrates.

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

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

3 Proposals for Packaging

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

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

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

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

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

- Variant Case 2 – use of cast-iron casks as used in Germany for the packaging of similar operational wastes from LWRs.

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

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

Proposals for packaging waste and spent fuel are described in more detail in Appendix B.

4 Radionuclide Inventory of ILW and Spent Fuel

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

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

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

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

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

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

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

5 Assessment of Proposed ILW Packages

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

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

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

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

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

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

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

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

The assessment of long-term disposal system performance in the GDA Disposability Assessment has been undertaken by comparison with the assessment performed for legacy ILW. This was based on the assumed characteristics for a generic UK Geological Disposal Facility site. Since the properties of any selected site necessarily would need to be consistent with meeting the regulatory risk guidance level [5], based on the approach adopted for Letter of Compliance assessment, this assessment assumed a groundwater flow rate and return time to the accessible environment that would meet regulatory requirements when considering the inventory of legacy ILW. The additional radionuclide inventory associated with the ILW from an EPR represents only a small fraction of that of the legacy wastes, particularly for the majority of the radionuclides that determine risk in the long-term. Even considering the conservative approach to inventory assessment and recognising the potential for future optimisation of packaging proposals, the additional risk from the disposal of ILW from a single EPR in a site of the type described would be consistent with meeting the regulatory risk guidance level. The consideration of a fleet of six reactors does not alter this conclusion.

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

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

The potential impact of the disposal of EPR operational and decommissioning ILW on the size of a Geological Disposal Facility has been assessed. Although the impact has some dependence on the packaging variant considered for operational ILW, it has been concluded that in all cases the necessary increase in the ‘footprint area’ is small, corresponding to less than approximately 60m of disposal vault length for each EPR. This represents approximately 1% of the area required for the legacy ILW, per reactor, and less than 10% for the illustrative fleet of six EPR reactors. This is in line with previous estimates for potential new build reactor designs [6].

6 Assessment of Spent Fuel Packages

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

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

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

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

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

It is acknowledged that the cooling period specified above is greater than would be required for existing PWR fuel to meet the same criterion and RWMD proposes to explore how this period can be reduced. This may be achieved for instance through refinement of the assessment inventory (for example by considering a more realistic distribution of burn-up), by reducing the fuel loading in a canister, or by consideration of alternative disposal concepts. The sensitivity of the cooling period to fuel burn-up has been investigated by consideration of an alternative fuel inventory based on an assembly irradiation of 50 GWd/tU. For this alternative scenario it is estimated that the cooling time required will reduce to the order of 75 years to meet the same temperature criterion.

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

Consequently, although the EPR spent fuel may influence the necessary arrangements, for example through the need for additional shielding, it is judged that sufficient flexibility exists in the current concept to allow suitable arrangements to be developed.

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

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

The leaching of radionuclides from spent fuel is characterised by an initial ‘instant release fraction’ (IRF), and then by a more general dissolution rate. The IRF is the fraction of the inventory of more mobile radionuclides that is assumed to be readily released upon contact with groundwater and is influenced by the properties of the spent fuel. In the case of higher burn-up fuel, the increased irradiation of the EPR fuel would increase the IRF as compared to that for lower burn-up fuel. Generally available information [7] on the potential performance of higher burn-up fuel has been used to provide a suitable IRF for assessment.

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

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

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

The potential impact of the disposal of EPR spent fuel on the size of a Geological Disposal Facility has been assessed. The assumed operating scenario for an EPR (60 years operation) gives rise to an estimated 900 disposal canisters, requiring an area of approximately 0.15 km^2 for the associated disposal tunnels. A fleet of six such reactors would require an area of approximately 0.9 km^2, excluding associated service facilities. This represents approximately 8% of the area required for legacy HLW and spent fuel per EPR reactor, and approximately 50% for the illustrative fleet of six EPR reactors. This is in line with previous estimates for potential new build reactor designs [6].

RWMD is currently developing a Generic Disposal System Safety Case covering the Baseline Inventory of waste and wastes that may potentially arise in the future as set out in the Managing Radioactive Waste Safely White Paper [8]. RWMD is also considering an upper bound inventory reflecting the uncertainty around the Baseline Inventory, including the potential for wastes and spent fuel to arise from a new nuclear build power programme. This will provide information on the disposability of the various categories of waste in a single 'co-located' facility. It is planned that the Generic Disposal System Safety Case will be published in September 2010 to support the Geological Disposal Facility site selection and assessment process. This will provide a baseline for the ongoing provision of advice on the disposability of wastes, including for future interactions on EPR waste and spent fuel.

7 Conclusions

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

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

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

8 References

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

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

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

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

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

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

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Appendix A
Protocol for GDA Disposability Assessment

The GDA Disposability Assessment of the EPR was based on a protocol agreed with Environment Agency and the Nuclear Installations Inspectorate (NII) [A1]. It was managed as a structured project using management procedures controlled under the RWMD Management System. A Project Board was established to provide oversight and a point of reference for key decisions.

The Project was run in a staged manner, based on three stages, as follows:

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 EdF/Areva, supplemented by existing RWMD experience. In particular, knowledge of radioactive waste management at the Sizewell B pressurised water reactor (PWR) was used to add value to the GDA Disposability Assessment for the EPR.

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

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

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 spent fuel.

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 (GDF) and operations in a GDF, including estimation of release fractions for a range of standard impact and fire scenarios.

The Disposal Facility Design evaluation considered the implications of operation of an EPR on the design of the GDF, including the following:

- the footprint area needed to accommodate the wastes, in both a standalone facility and in a disposal facility also incorporating legacy wastes;
- 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 impact of any novel conditioning or management methods.

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 the EPR with existing assessments of RWMD reference disposal concepts. The assessments provide the basis for judging the potential disposability of operational and decommissioning ILW, and spent fuel from the EPR.

- the Transport Safety assessment considered the logistics, regulatory compliance and risk of transport operations, with specific consideration of dose, gas generation, containment and heat output under normal and accident conditions;
- the Operational Safety assessment considered dose due to accidents, effects of gas generation and criticality;
- the Post-closure Safety assessment considered potential radioactivity impacts due to the groundwater and gas pathways, human intrusion and criticality, and environmental impacts due to chemotoxic species contained in the waste;
- the Consideration of Environmental Issues considered material usage in the GDF and commented on the consideration of options for waste management strategies and their implications for non-radiological environmental impacts;
- the Security Evaluation included determination of the likely security categorisation of the proposed waste packages, identification of nuclear material and commentary on proposals for accountancy and independent verification of the use of nuclear materials.

References

A1. NDA RWMD, *Disposability Assessment of Solid Waste Arisings from New Build*, April 2008.

Appendix B
EPR Operation, Wastes, Packaging Proposals and Package Characteristics

This Appendix provides a summary of the information used in the GDA Disposability Assessment for the EPR. RWMD used the information supplied by EdF/Areva, supplemented as necessary by information available to RWMD, to estimate package numbers, inventories and general characteristics.

This section contains the following information:

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

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

B1 Summary of EPR Design and Operation

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

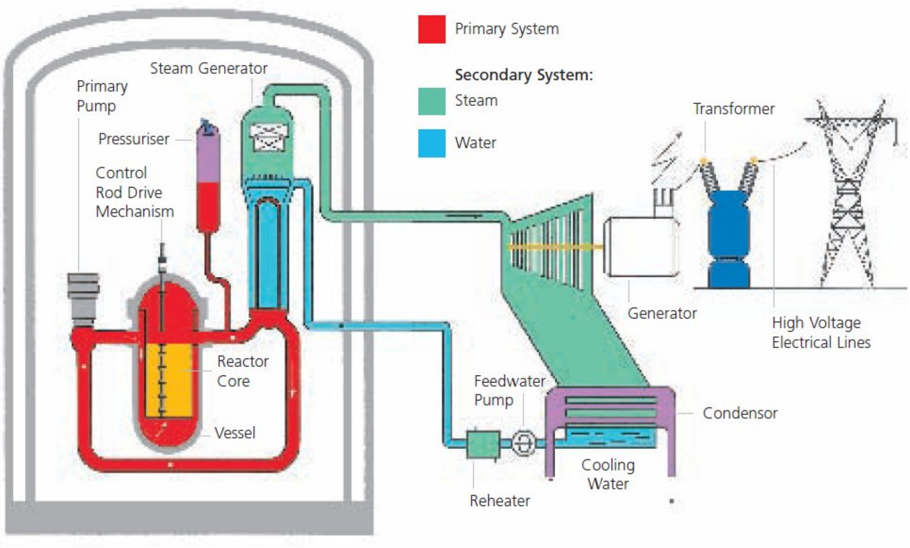
The EPR evolutionary design is based on experience from operation of Light Water Reactors (LWR) worldwide, primarily those incorporating the most recent technologies: the N4 and KONVOI reactors currently in operation in France and Germany respectively. The primary system design, loop configuration, and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

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

The EPR design is furnished with a four-loop, pressurised water ‘reactor coolant system’ composed of a reactor vessel that contains the fuel assemblies, a pressuriser including control systems to maintain system pressure, one reactor coolant pump per loop, one steam generator per loop, associated piping, and related control and protection systems (Figure B1). This equipment would be standardised for all EPRs.

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

Figure B1 – Principal components of an EPR. Figure reproduced from [B1]



B2 Assumptions

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

- The EPR would be operated for 60 years. During the operation of the reactor, fuel assemblies would be periodically rotated within the reactor core, and then removed and replaced with other fuel assemblies. Ninety spent fuel assemblies would be removed from the reactor every 18 months during planned shutdown periods and require storage.
- The date at which operation of power production from an EPR would commence in the UK is uncertain. Government and industry estimates suggest that the first reactor could be operational by 2017. In the GDA Disposability Assessment for the EPR, estimates of time-dependent properties, e.g. those related to radioactive decay, are assessed from the time of generation of the waste. In discussion of the implications for management of radioactive waste, RWMD has assumed a start date for a single reactor of 2020.
- Spent fuel characteristics have been determined on the assumption that the reactor would be operated to achieve a maximum fuel assembly irradiation (burn-up)[1] of 65 GWd/tU. In the absence of data to the contrary, the GDA Disposability Assessment has assumed that all fuel will be irradiated to the maximum fuel assembly burn-up. This is a conservative approach and ensures that the conclusions from the assessment are bounding for a wide range of possible operational behaviours.

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[1] The fuel assembly average irradiation (burn-up) represents the total irradiation associated with all the fissile material in an assembly divided by the initial mass of uranium in the assembly. It takes into account the variation in irradiation both axially along a fuel rod and the variation from one fuel rod to another. For simplicity, whenever fuel irradiation or burn-up is referred to in the remainder of the report what is meant is fuel assembly average irradiation or burn-up. Thus, the statement that the maximum fuel assembly burn-up is 65 GWd/tU means that the highest fuel assembly average burn-up will be 65 GWd/tU.

- The fuel used in the EPR would be manufactured from freshly mined uranium enriched to an initial U-235 content of 5%[2].
- It is assumed that ILW and spent fuel from the EPR will arrive at the GDF in a packaged state, ready for disposal.

B3 ILW Streams, Packaging Assumptions, Package Numbers and Characteristics

B3.1 Operational ILW Streams and Packaging Assumptions

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

- Ion exchange resins – organic resins that arise from the clean-up of primary circuit water and water from the Liquid Waste and spent fuel Pit Treatment Systems;
- Spent cartridge filters (ILW) – filters from the clean-up of primary circuit water and water from the Liquid Waste and spent fuel Pit Treatment Systems. The filters consist of a stainless steel support, with a glass fibre or organic filter media;
- Spent cartridge filters (LLW and ILW)[3] – filters, similar to, but typically smaller in size than spent cartridge filters (ILW);
- Operational wastes >2mSv/hr – a range of materials, including contaminated metal, plastics, cloth, glassware and rubble, arising from operations during planned shutdown periods (hence ‘operational wastes’);
- Wet sludges – sludges arising from cleaning the bottoms of liquid waste treatment tanks and various sumps;
- Evaporator concentrates – residues from the evaporation of waste water.

The raw waste volumes of these materials as determined by EdF/Areva are provided in Table B1.

The GDA Disposability Assessment for the EPR considered three scenarios for conditioning and packaging of operational ILW arising over 60 years, referred to as the Reference Case, Variant Case 1 and Variant Case 2. Waste stream identifiers for each scenario are specified in Table B1.

Reference Case

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

Two types of cylindrical pre-cast concrete casks, designated C1 and C4, were assumed in the Reference Case (Figure B2). Both of these casks can include internal mild steel shielding of flexible thickness (40-100 mm of shielding was assumed for the GDA Disposability Assessment) to provide shielding against different concentrations of gamma emitting radionuclides. The C1 Cask is 1.4 m in diameter, 1.3 m high, and has a 0.15 m thick concrete shield wall. The C4 Cask has the same dimensions apart from the diameter; it is

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

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

Table B1 – Total lifetime raw waste volumes for operational ILW from an EPR and identifiers used for different management scenarios

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

1.1m in diameter. The C1 and C4 Casks are assumed to be used as Industrial Package Type 2 (IP-2) transport packages as defined by IAEA Transport Regulations.

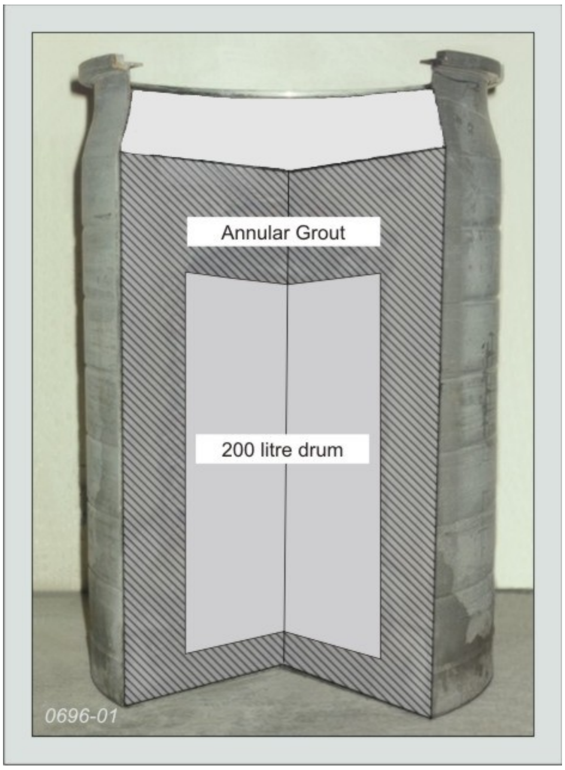
In the Reference Case scenario, the operational ILW would be immobilised using epoxy resin (EPR01), or cement grout (EPR02, EPR03, EPR04 and EPR05). The range of wastes comprising EPR04 would be placed into plastic bags and compressed to reduce volume before grouting. The EdF/Areva submission assumed that, in the Reference Case, evaporator concentrates would be incinerated leaving no radioactive residue, which is the current practice in France. The Reference Case identifier for Evaporator Concentrates in table B1 is therefore ‘NA’. The absence of radioactive residue following this practice will need to be confirmed in future.

Figure B2 – Illustration of the C1 and C4 concrete casks proposed for packaging of operational ILW in the EPR Reference Case

Variant Case 1

For Variant Case 1, it is assumed that EPR operational ILW (waste streams EPR11-EPR16) will be packaged in 200 litre Drums, which would subsequently be placed in UK standard stainless steel 500 litre Drums with an annular grout lining cast into place during packaging and assumed to be 100 mm thick (Figure B3). Evaporator concentrates are assumed to be packaged and disposed of rather than incinerated. Waste streams EPR11 (Ion exchange resins), EPR14 (Operational wastes), EPR15 (Wet sludges) and EPR16 (Evaporator concentrates) would be dried and packaged directly in the 200 litre drums (i.e. without immobilisation). Waste streams EPR12 (Spent cartridge filters, ILW) and EPR13 (Spent cartridge filters, LLW and ILW) would be grouted into the 200 litre drums. For transport, the 500 litre Drums would be placed, in groups of four, inside a Standard Waste Transport Container with 285 mm of shielding (SWTC-285) [B2] for transport as Type B transport packages. The SWTC-285 has been designed with a cavity size and shape suitable for transport of four 500 litre Drums using a handling frame.

Figure B3 – Illustration of 200 litre drum cement grouted within a 500 litre Drum as proposed for packaging of operational ILW in the EPR Variant Case 1



Variant Case 2

For Variant Case 2, the waste is assumed to be packaged in cylindrical cast-iron casks. Containers of this type, for example MOSAIK Casks, are approved and are currently used for the packaging of operational waste in Germany (Figure B4). These casks are made from Ductile Cast Iron, and have dimensions of 1.06m (diameter) by 1.5m (height) and have walls, base and lid thicknesses of 0.16m. The cast-iron casks may be used as either Industrial Package Type 2 (IP-2) or Type B[4] transport container, the latter requiring suitable over-packing arrangements to ensure appropriate performance. For reasons of efficiency, this assessment has assumed that Type B arrangements would be used to ensure optimum waste loading. It is recognised further development work would be required to confirm the appropriateness of this assumption.

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

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

cartridge filters would be conditioned in the same manner as for Variant Case 1, i.e. conditioned by grouting within the cast-iron casks.

Figure B4 – Illustration of a MOSAIK Cask as proposed for packaging of operational ILW in the EPR Variant Case 2

B3.2 Decommissioning ILW Streams and Packaging Assumptions

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

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

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

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

- EPR06 (Reactor Vessel), which consists of ferritic steel associated with the mid-height section of the pressure vessel and from the vessel cladding. The pressure vessel steel will be in the form of thick (~0.2m) curved steel plate, possibly with its stainless steel cladding, typically a few mm thickness, still attached. These wastes would be grouted into 4 metre Boxes with a 100 mm concrete wall (Figure B5) and would be transported as IP-2 transport packages.

- EPR07 (Upper and Lower Reactor Internals), which consists of low cobalt stainless steel in the form of plates with thickness of the order of 0.01m. These wastes would be grouted into 3m^3 Boxes (Figure B5); and would be transported in a standard waste transport container (SWTC) as Type B transport packages.
- EPR08 (Lower Reactor Internals including Heavy Shield), which consists of two similar grades of stainless steel: a low cobalt grade steel used for all the components closest to the core that receive the highest irradiation, and a grade containing higher concentrations of cobalt used for the more distant components. These steels are expected to have plate-like structures with thickness of the order of 0.01m. These wastes would be grouted into 3m^3 Boxes (Figure B5); and would be transported in a SWTC as Type B transport packages.

Raw waste volumes for decommissioning ILW are presented in Table B2.

B5 – Illustration of a 3m^3 Box and 4m Box as proposed for packaging of decommissioning ILW from the EPR

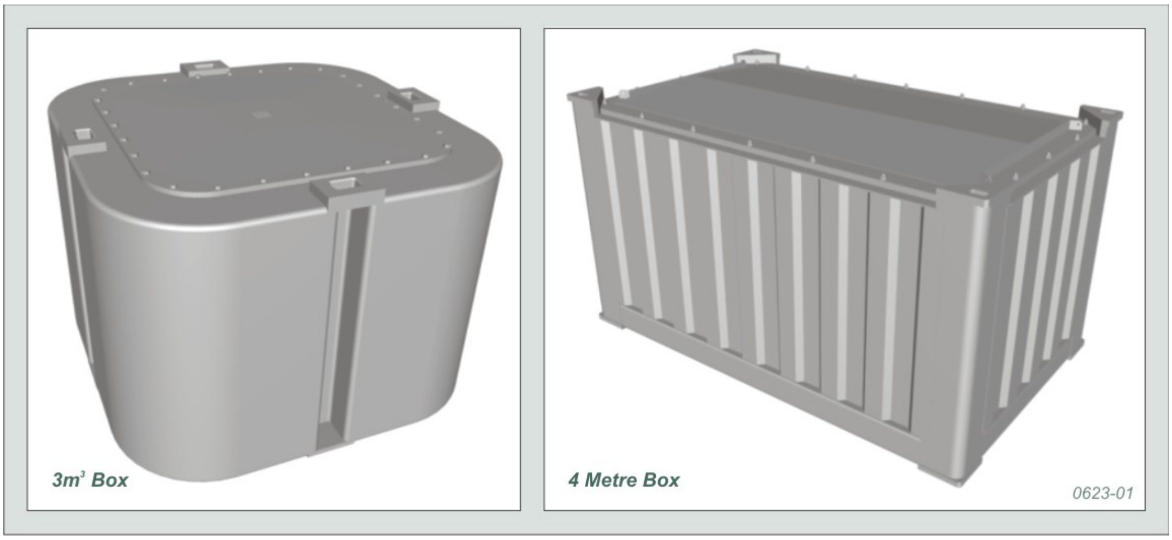


Table B2 – Raw waste volumes for decommissioning ILW from an EPR and identifiers used

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

The inventory of C-14 associated with decommissioning ILW has been estimated through activation calculations based on an assumed concentration of the relevant pre-cursor species (primarily nitrogen). The concentration of nitrogen in steels has been assumed to be 800 ppm. In the absence of other data this is thought to be a conservative assumption.

B3.3 ILW Package Numbers and Characteristics

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

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

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

The EPR ILW waste package radionuclide-related parameters and waste quantities (package numbers and total packaged volume) are given in Tables B3 to B6. Radionuclide related parameters (e.g. dose rate) are calculated at the time of arising (i.e. zero-decayed for operational ILW and 40 year decayed for decommissioning ILW appropriate to the assumed times if transport). The fissile content of waste is not included in the summary tables as it is estimated to be well below the 15g fissile exception level for non-fissile transport packages.

Information on the raw waste volumes, package types, package numbers and radionuclide content have been derived from consideration of information on waste packages from existing PWRs provided by EdF/Areva and enhanced by RWMD. These data are presented in the form of average waste package inventories.

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

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

Table B3 – EPR Waste Stream Data: Operational ILW Reference Case(1) (2)

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

Notes:

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

(2) Radionuclide data for the maximum package may be obtained as M times the average package data where approximately M=12 for EPR01 & EPR02, M=10 for EPR03 & EPR05 and M=7 for EPR04.

Table B4 – EPR Waste Stream Data: Operational ILW Variant Case 1(1) (2)

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

Notes:

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

(2) Radionuclide data for the maximum package may be obtained as M times the average package data where approximately M=12 for EPR11 & EPR12, M=10 for EPR13, EPR15 & EPR16 and M=7 for EPR14.

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

Table B5 – EPR Waste Stream Data: Operational ILW Variant Case 2(1) (2)

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

Notes:

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

(2) Radionuclide data for the maximum package may be obtained as M times the average package data where approximately M=12 for EPR21 & EPR22, M=10 for EPR23, EPR25 & EPR26 and M=7 for EPR24.

Estimates of the quantities and characteristics of decommissioning ILW (Table B6) have been developed based on modelling of the neutron flux, power history and material composition data for the core of an EPR reactor. The activation calculations used the highest total flux experienced by each component to derive its total inventory. Therefore, the inventories for EPR decommissioning ILW considered to be upper bound estimates (maximum package inventories).

Table B6 – EPR Waste Stream Data: Decommissioning ILW(1)

Waste Stream	Package Type	Number of Packages	Total Packaged Waste Volume (m^3)	Maximum Package Alpha Activity (TBq)	Maximum Package Beta/ Gamma Activity (TBq)	Maximum Package A_2 Content	Maximum Package Heat Output (Watts)	Maximum Package Dose Rate at 1m from Package (mSv/hr)(2)
EPR06	4mBox100	10	215.00	1.21E-04	1.98E+00	3.29E-01	3.14E-02	1.45E-01
EPR07	$3m^3$ Box	25	82.50	1.88E-02	1.12E+03	8.38E+01	7.93E+00	4.41E-03
EPR08	$3m^3$ Box	46	151.80	6.04E-02	3.59E+03	3.33E+02	3.57E+01	2.32E-02
TOTALS		81	449.30					

Notes:

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

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

Tables B3, B4, B5 and B6 are underpinned by a detailed evaluation of the radionuclide inventory of each of the waste streams.

B3.4 Comparison of EPR ILW with Sizewell B ILW

In order to place the information on the radioactivity of the ILW that would arise from an EPR in context, a comparison has been made with ILW from Sizewell B, which is the pressurised water reactor operated in the UK by British Energy. The Sizewell B design net electrical power output is 1188 MW(e) [B3] and an assumed operating life of 40 years, whereas the EPR's electrical power output is 1,600 MW(e) for an assumed operating life of 60 years.

Information on the Sizewell B ILW inventory has been taken from the 2007 National radioactive Waste Inventory [B4].

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

The activity of EPR stainless steel decommissioning ILW (streams EPR07 and EPR08) is compared with the activity of the equivalent Sizewell B PWR waste [B4] (2007 National Inventory stream 3S306) in Table B7. The basis for Table B7 is as follows:

- radionuclide activities have been estimated for 40 years after reactor shutdown;
- the activity data have been normalised to the total electrical output of the two reactors (Sizewell B – 1.18 GW(e) for 40 years, EPR 1.6 GW(e) for 60 years), this allows a like-for-like comparison of the radionuclide inventories between the two types of reactors, and highlights any differences that would result from the design of the reactor or the operational practices (e.g. intensity of neutron flux);
- all the EPR normalised activity values (measured in TBq per GW(e).yr) were reduced by a factor of three as EDF/Areva had used peak neutron fluxes to calculate radionuclide activities, whereas the Sizewell B data had used average neutron flux data in the activation calculations;
- the radionuclides considered in Table B7 are the top 10 most active in the EPR wastes for which estimates were also available for the Sizewell B PWR wastes;
- the cell colouration displayed in the sixth column of Table B7 is used to indicate the closeness of the agreement that presents the ratio of EPR to Sizewell B normalised activities as follows: green 0.33 to 3, yellow 0.1 to 0.33 & 3 to 10, pink <0.1 & > 10.

As can be seen from Table B7, with the exception of H-3, the activities of the five radionuclides with the highest activities are similar (within a factor of three) and the activity of the radionuclide with the sixth highest activity, Nb-93m, is only just outside that range. Like H-3 the total activity of Mo-93 and Tc-99 is considerably higher in the EPR stainless steel wastes than that from Sizewell B. This can be explained by the application of conservative upper bound trace element concentrations in the RWMD inventory enhancement work.

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

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

Table B7 – Comparison of radionuclide activities for Stainless Steel decommissioning ILW from an EPR with Equivalent ILW stream from Sizewell B PWR (3S306)

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

B4 Description of Spent Fuel, Packaging Assumptions, and Package Numbers and Characteristics

B4.1 Description of Spent Fuel

The core of an EPR consists of 241 fuel assemblies providing a controlled fission reaction and a heat source for electrical power production. Each fuel assembly is formed by a 17×17 array of Zircaloy M5 tubes, made up of 265 fuel rods and 24 guide thimbles, as illustrated in Figure B6.

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

The EPR fuel assembly and fuel rod are illustrated in Figure B6 and some additional dimensional information is provided in Table B8.

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

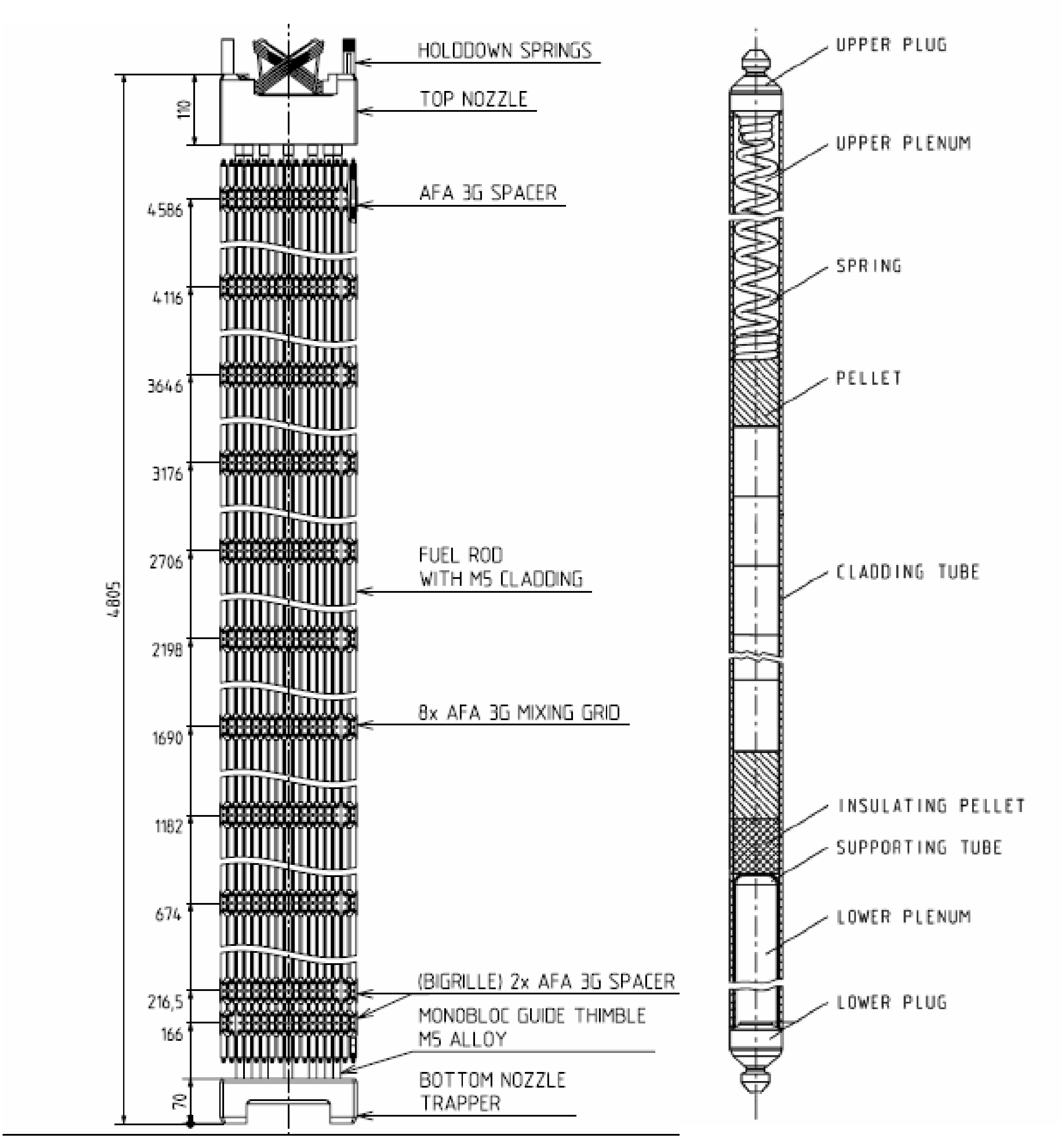
The stack of UO_2 pellets extends over a height of 4.2m known as the active height of the fuel. Above and below the UO_2 stack are the upper and lower fission gas plenums designed to accommodate any volatile fission products released during the irradiation process. An Inconel (believed to be Grade 718)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

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

Table B8 – Dimensional information for EPR fuel assemblies and rods

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

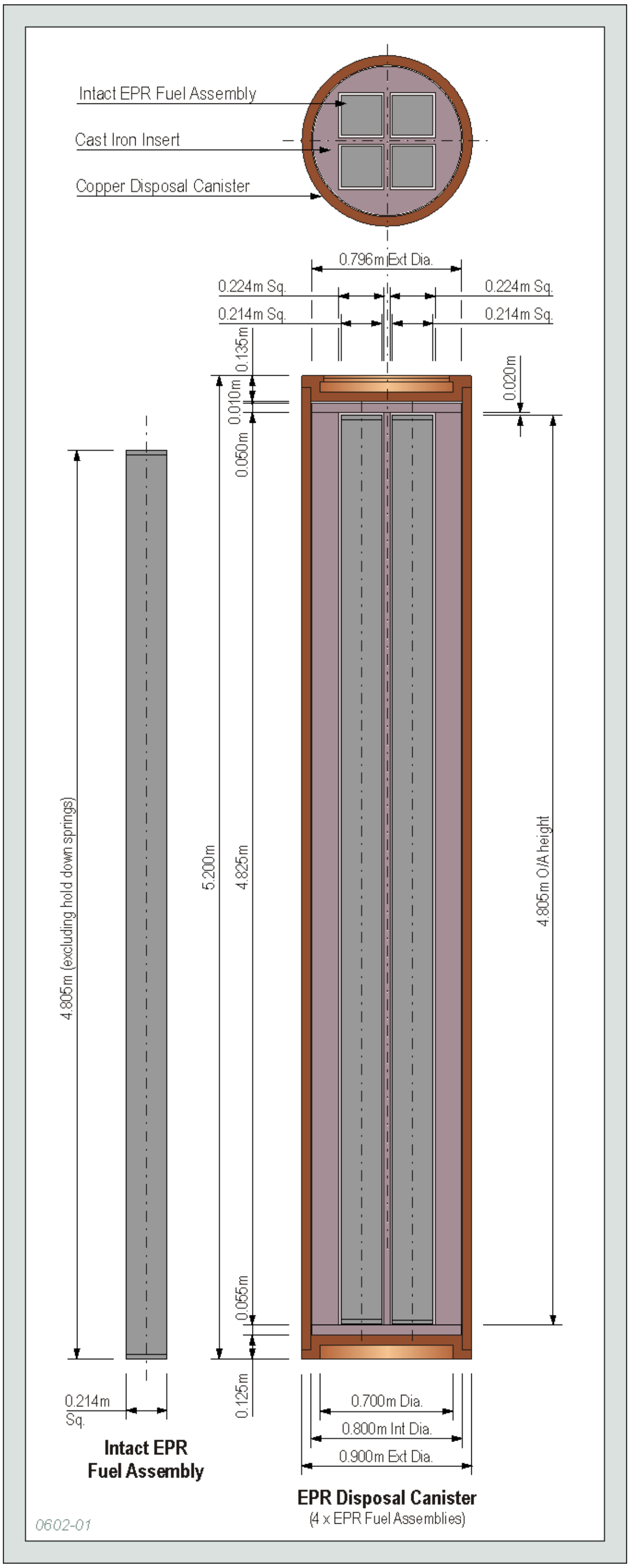
Figure B6 – Components of an EPR fuel assembly (left) and a single EPR fuel rod (right)



B4.2 Spent Fuel Packaging Assumptions

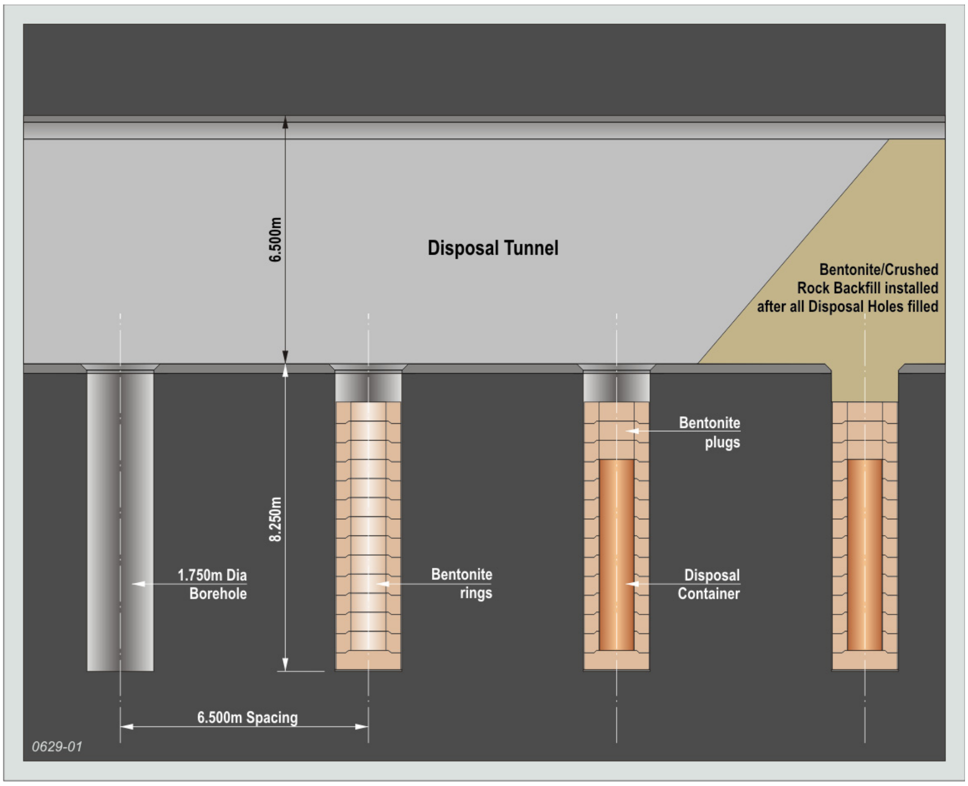
The packaging assumptions for EPR spent fuel are based on concepts developed by RWMD to date [B5]. Under these concepts, spent fuel would be over-packed into durable, corrosion-resistant disposal canisters manufactured from suitable materials, which would provide long-term containment for the radionuclides contained within the spent fuel (Figure B7). Although the canister material remains to be confirmed, the assessment has considered the potential performance of copper and steel canisters. In both cases, it is assumed that an additional

Figure B7 – Illustration of an EPR spent fuel disposal canister



cast-iron inner vessel is used to hold and locate the spent fuel assemblies, and in the case of the copper canister would provide mechanical strength as well. These canisters would be emplaced in disposal holes lined with a buffer made from compacted bentonite, which swells following contact with water (Figure B8). The concept is based on the KBS-3V concept developed by SKB for disposal of spent fuel in Sweden [B6].

Figure B8 – Longitudinal section of a disposal tunnel illustrating the disposal holes and immediate emplacement of backfill following disposal of spent fuel



Disposal of EPR spent fuel assumes a disposal canister with a length of 5.2 m (Figure B7). This is approximately 0.6 m longer than the longest disposal canister envisaged for legacy (Sizewell B PWR) spent fuel. The reference assumption is for four spent fuel assemblies to be packaged in each canister.

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

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

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

B4.3 Spent Fuel Package Numbers and Characteristics

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

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

The dimensions of one fuel assembly are 0.214m x 0.214m x 4.805m (Figure B7), so the raw waste volume associated with 3,600 fuel assemblies is 792 m^3. Regarding packaged volume, the envelope volume of a canister capable of accommodating four fuel assemblies is 3.33 m^3, and the packaged volume of the waste consisting of 900 canisters is therefore 2,997 m^3.

The GDA Disposability Assessment for the EPR has assumed that the concentration of chlorine impurities in the fuel was 25ppm. In the absence of other data this is thought to be a conservative assumption.

The component mass estimates for an EPR fuel assembly are provided in Table B9. This table includes a small additional quantity of Zircaloy M5 used to balance the individual component masses with the total spent fuel assembly mass. Table B10 presents the mass data for each fuel assembly and each canister (for illustrative purposes, copper has been assumed as the material of manufacture in this case), summed for each material type. Package data are summarised in Table B11.

Table B9 – Estimates of component mass for an EPR fuel assembly

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

Table B10 – Material mass breakdown for an EPR fuel assembly and for a copper canister (assuming four assemblies per canister)

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

Table B11 – EPR Waste Stream Data: Spent Fuel(1)

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

Notes:

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

Although EdF/Areva is designing and planning for a burn-up of fuel to 65 GWd/tU, this is the maximum burn-up that a fuel assembly would experience. The lifetime thermal energy production for an EPR at a load factor of 93% would be 9.17E+04 GWd. These 3,600 EPR fuel assemblies would contain 1,899 tU. Therefore, assuming that 3,600 fuel assemblies are generated over the lifetime of a reactor implies that the average burn-up of the assemblies is 48.3 GWd/tU. In calculating the total spent fuel inventory for the post-closure performance assessments, it was assumed that all 3,600 spent fuel assemblies had been irradiated to 65 GWd/tU, rather than 48.3 GWd/tU. This is clearly conservative although the conservatism only amounts to about a factor of 1.3 for most of the post-closure significant radionuclides.

B4.4 Comparison of EPR Spent Fuel with Sizewell B PWR Spent Fuel

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

For example, an EPR operating for 60 years at 1.6 GW(e) would produce 3,600 spent fuel assemblies, which is equivalent to 37.5 spent fuel assemblies for every GW(e) year. In comparison, assuming the PWR at Sizewell B operates for 40 years at 1.188 GW(e) and produces 2,228 spent fuel assemblies [B7], 46.9 spent fuel assemblies would be produced for every GW(e) year. Thus the efficiency gains can be seen, however it should be noted that this does lead to a higher concentration of activity in EPR spent fuel assemblies in comparison to Sizewell B PWR spent fuel assemblies.

Table B12 provides a comparison of the radionuclide inventories for the most significant post-closure radionuclides in spent fuel from an EPR, with radionuclide inventories for spent fuel from Sizewell B PWR. The comparison is based on the inventory of radionuclides estimated to be present in one spent fuel canister at 90 years cooling[5] The data for the Sizewell B PWR are derived from the Low Burn-up PWR data presented in [B8], the fission product and actinide data from which were used in a previous assessment of the implications associated with new build reactors undertaken by Nirex [B9].

Table B12 – Comparison of radionuclide activities for spent fuel from an EPR with spent fuel from Sizewell B

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

The only comparison of EPR and Sizewell B spent fuel inventories that could readily be made involves EPR's maximum fuel assembly average burn-up inventory with the batch average fuel burn-up inventory associated with Sizewell B, as reported in [B3]. It is recognised that it would have been more appropriate to compare either the two maximum fuel assembly average burn-up cases or two batch average fuel burn-up inventories. However, relevant information was not available for such comparison at the time of this assessment. Since the burn-up assumed for EPR spent fuel is about twice that assumed for the Sizewell B spent fuel, for many radionuclides the ratio of EPR to Sizewell B fuel activities is about two, as shown in Table B12. Ratios a little below and above two reflect non-linearity effects that arise from, for example, the higher proportion of fissions coming from Pu-239 in

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[5] 90 years was selected at the outset of this assessment to provide a reasonable approximation of the amount of cooling time expected before disposal.

the higher burn-up fuel. A few of the activity ratios are outside the range that might be expected from the different burn-ups and these, perhaps unexpected differences are attributable to five separate causes which are discussed below. Yellow, pink, blue and green shadings have been used in Table B12 to identify the causes of the apparently anomalous activity ratios.

Yellow cells: C-14, Cl-36 and Ni-59. These radionuclides arise mainly as activation products of trace impurities or in the case of Ni-59, from trace impurities and the small amount of a nickel alloy (Inconel 718) used for grid springs. The stable elements responsible for these activation products are: nitrogen for C-14; chlorine for Cl-36; nickel for Ni-59. In general, EdF/Areva has adopted more conservative specification limit values for the trace impurities in their spent fuel inventory calculations than has been adopted by RWMD in previous studies of PWR fuel. This has led to EPR inventories that are more than the factor of two greater than those coming from the Sizewell B calculations (identical impurity levels would have resulted in EPR inventories being about twice the Sizewell B inventories because of the two-fold higher irradiation). For example, for its calculations EdF/Areva adopted chlorine concentrations of approximately 25ppm and 20ppm for the UO_2 and Zircaloy M5 cladding respectively, whilst the Sizewell B calculations used approximately 5ppm chlorine for the UO_2 and neglected the chlorine content of the cladding. Based on an extensive Cl-36 research project conducted by Nirex in the 1990's the chlorine concentrations adopted for the Sizewell B calculations are considered more justifiable (i.e. the upper bound chlorine concentration for LWR UO_2 and Zircaloy were assessed to be approximately 5ppm and 1.7ppm respectively [B10],[B11]).

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

Pink cells: Se-79. Differences in the estimated activities of Se-79 are associated with changes to data on the fission yield and half-life of this radionuclide, and these parameters have been revised in recently published nuclear data libraries. For a given fission yield in terms of number of atoms, the associated activity is inversely proportional to half-life. The estimated activity of Se-79 for an EPR used a half-life for the radionuclide of about 3.3E+05 years. However, the Sizewell B estimates used a Se-79 half-life of 6.5E+04 years, and the difference in Se-79 activity presented in Table 12 is in accord with the difference in half-lives and burn-ups associated with the two with the two spent fuel calculations used to develop the estimates.

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

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

242 and Am-243 activity with burn-up is still fundamentally down to the fact that they are produced by multi-step activation reactions.

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

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