Generic Design Assessment – New Reactors Programme

Assessment of the responses to RI-ABWR-0001 - Definition and Justification for the Radioactive Source Terms in UK ABWR during Normal Operations

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EXECUTIVE SUMMARY

This report presents the findings of the ONR assessment of submissions provided by Hitachi-GE in response to Regulatory Issue RI-ABWR-0001 - Definition and Justification for the Radioactive Source Terms in UK ABWR during Normal Operations.

The purpose of this assessment was three-fold;

- To document the assessment which underpins the recommendation made in closing RI-ABWR-0001, or otherwise;
- To serve as a record of the scope of the assessment undertaken for RI-ABWR-0001, and therefore the boundaries of the judgements made; and
- To identify any associated residual matters which may need to be satisfactorily addressed during the remainder of Generic Design Assessment (GDA) Step 4, or beyond, as appropriate.

The definition of the radioactive source term, namely the nature and amount of radioactivity, is a fundamental part of understanding and therefore being able to control the hazards associated with any nuclear facility. This definition should be based upon a suitable and sufficient justification, which should demonstrate that the derived values are appropriate to be used within the safety case, in whatever capacity is necessary. Failure to adequately define or justify the source term could ultimately mean that the design, operations or controls specified may not be soundly based. It would also prove difficult to demonstrate that associated risks have been reduced So Far As Is Reasonably Practicable (SFAIRP).

Based upon the submissions made by Hitachi-GE during Steps 2 and 3 of the GDA for UK ABWR, ONR judged there to be serious regulatory shortfalls associated with both the definition and justification of the source terms for UK ABWR (particularly for normal operations). These had the potential to prevent provision of a Design Acceptance Confirmation (DAC). In line with the guidance to requesting parties, ONR therefore raised RI-ABWR-0001, to make regulatory expectations clear and to ensure that these shortfalls were addressed during GDA.

In response Hitachi-GE provided a suite of documentation which defines and justifies the concentration of radionuclides around the UK ABWR plant during all modes of normal operations. This includes radioactivity with the reactor, water and gaseous auxiliary systems as well as deposited on piping surfaces and fuel cladding. In addition responses to numerous Regulatory Queries were submitted, providing additional clarification and evidence.

The main conclusions of the assessment are:

- The scope and approach adopted by Hitachi-GE in responding to the RI is adequate;
- The use of relevant plant operating experience, utilising the broadest data set that is considered pertinent, gives confidence in the defined values. Where suitable data does not exists recourse is made to other methods in an satisfactory manner;
- Throughout the development of the source term suitable and sufficient consideration has been given to safety, including consideration of all significant radionuclides that exist in the systems expected to contain radioactivity throughout the envisaged operational states;
- The defined UK ABWR source terms includes all appropriate sources of radioactivity within the plant, including mobile and fixed sources, and considers how the nature and quantities of radioactivity within the plant may change over time;

- Variations in radioactivity due to the different operational phases of the plant both in the short term and long term are appropriately considered, covering the entire fuel cycle;
- Both Best Estimate (BE) and Design Basis (DB) values are defined, representing an expected and more conservative estimate for the likely levels of radioactivity within UK ABWR. The BE values derived represent a reasonable estimate, for safety case purposes, of the likely performance of UK ABWR. A set of conservative DB values have been derived which should be suitable for use in the safety case;
- Where uncertainties still remain due to the methodologies, assumptions or approach these would not have a significant impact on the derived values;
- The derived values are further justified using additional OPEX, calculation, literature and sensitivity analysis. An adequate and proportionate degree of supporting evidence has been provided, which is focussed on those nuclides of highest safety significance;
- While the responses have been updated several times throughout my assessment, sufficient evidence has been documented to capture and understand the basis of the UK ABWR source terms should this need to be revisited in the future.

Overall, the defined UK ABWR source terms are now fit for purpose in making the UK ABWR safety case. While further changes may still occur, these should only be minor in nature.

While this assessment has identified a number of Residual Matters, none of these are significant enough to prevent closure of the RI.

To conclude, based on this assessment, Hitachi-GE have provided sufficient evidence to meet the intent of RI-ABWR-0001 and have addressed the issues which led to it being raised. RI-ABWR-0001 has therefore been resolved.

On the basis of this assessment, the following recommendations are made:

- <u>Recommendation 1</u>: RI-ABWR-0001 should be closed.

- <u>Recommendation 2</u>: The Residual Matters identified in this report should be considered by the relevant UK ABWR discipline inspectors and actioned as considered appropriate.

- <u>Recommendation 3</u>: The evidence provided as part of the resolution of RI-ABWR-0001 regarding management of the source terms should be considered as part of RO-ABWR-0006 Actions 3, 7 and 8.

LIST OF ABBREVIATIONS

ActP	Actinide Product
ALARP	As low as is reasonably practicable
AP	Activation Product
BE	Best Estimate
BWR	Boiling Water Reactor
CA	Cycle Average
CP	Corrosion Product
CPS	Condensate Purification System
CST	Condensate Storage Tank
CUW	Clean-Up Water (system)
DAC	Design Acceptance Confirmation
DB	Design Basis
DST	Deposit Source Term
EUST	End User Source Term
FP	Fission Product
FPC	Fuel Pool Cooling and clean-up (system)
FW	FeedWater
GDA	Generic Design Assessment
GEH	GE-Hitachi
HOW2	(ONR) Business Management System
HWC	Hydrogen Water Chemistry
IAEA	International Atomic Energy Agency
JBWR	Japanese Boiling Water Reactor
JABWR	Japanese Advanced Boiling Water Reactor
KK	Kashiwazaki Kariwa [power plant]
LCW	Low Conductivity Waste (system)
NMCA	Noble Metal Chemical Addition
NWC	Normal Water Chemistry
ONR	Office for Nuclear Regulation
OR	Operating Rule
ORIGEN	Oak RIdge Isotope GENerator (code)
OPEX	OPerational EXperience
PST	Primary Source Term
PrST	Process Source Term
RCCV	Reinforced Concrete Containment Vessel
RGP	Relevant Good Practice
RHR	Residual Heat Removal (system)
RI	Regulatory Issue

RO	Regulatory Observation
RP	Requesting Party [Hitachi-GE]
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
SAP	Safety Assessment Principle(s)
SFAIRP	So far as is reasonably practicable
SFP	Spent Fuel Pool
SS	Spent Sludge (system)
TAG	Technical Assessment Guide(s) (ONR)
TSC	Technical Support Contractor
UK ABWR	United Kingdom Advanced Boiling Water Reactor
WENRA	Western European Nuclear Regulators' Association

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1 INTRODUCTION

1.1 Background

1. This report presents the findings of my assessment of the submissions provided by Hitachi-GE in response to Regulatory Issue RI-ABWR-0001 - Definition and Justification for the Radioactive Source Terms in UK ABWR (United Kingdom Advanced Boiling Water Reactor) during Normal Operations (Ref 1). Assessment was undertaken in accordance with the requirements of the Office for Nuclear Regulation (ONR) How2 Business Management System (BMS) guide NS-PER-GD-014 (Ref. 2). The ONR Safety Assessment Principles (SAP) (Ref. 3), together with supporting Technical Assessment Guides (TAG) (Ref. 4), have been used as the basis for this assessment.

2. The definition of the radioactive source term, namely the nature and amount of radioactivity, is a fundamental part of understanding and therefore being able to control the hazards associated with any nuclear facility. This definition should be based upon a suitable and sufficient justification, which should demonstrate that the derived values are appropriate to be used within the safety case, in whatever capacity is necessary. Failure to adequately define or justify the source term could ultimately mean that the design, operations or controls specified may not be soundly based. It would also prove difficult to demonstrate that associated risks have been reduced So Far As Is Reasonably Practicable (SFAIRP).

3. Based upon the submissions made by Hitachi-GE (the Requesting Party (RP)) during Steps 2 and 3 of the Generic Design Assessment (GDA) for UK ABWR, ONR judged there to be serious regulatory shortfalls associated with both the definition and justification of the source terms for UK ABWR (particularly for normal operations). These had the potential to prevent provision of a Design Acceptance Confirmation (DAC). In line with the guidance to requesting parties (Ref. 5), ONR therefore raised a Regulatory Issue (RI), RI-ABWR-0001, to make regulatory expectations clear and to ensure that these shortfalls were addressed during GDA.

4. The purpose of this report is therefore three-fold;

- To document the assessment which underpins the recommendation made in closing RI-ABWR-0001, or otherwise;
- To serve as a record of the scope of the assessment undertaken for RI-ABWR-0001, and therefore the boundaries of the judgements made; and
- To identify any associated residual matters which may need to be satisfactorily addressed during the remainder of GDA Step 4, or beyond, as appropriate.

1.2 Scope

5. The scope of this report covers the assessment of only those matters identified within the scope of RI-ABWR-0001, as defined within the RI (see Annex 1). Importantly this means that the scope of source terms considered are those related to ‘normal operations’ (see Section 2.1 for further details). Overall, the scope of this report is to support my judgement on whether a suitable and sufficient definition and justification for the source terms for the UK ABWR has been provided. The full scope of the submissions provided in response to this RI have been sampled and assessed in order to make that judgement.

1.3 Methodology

6. The methodology for the assessment follows HOW2 guidance on mechanics of assessment within the Office for Nuclear Regulation (ONR) (Ref. 6). I have sampled all of the submissions made in response to this RI, to various degrees of breadth and

depth. I chose to focus my assessment on those aspects which I judged to have the greatest potential impact on the final source terms values. My sample has also been influenced by the identified uses of the source terms within the UK ABWR safety case, my previous experience in source terms for reactors and other nuclear facilities and the specific gaps in the original submissions made by the RP which led to the RI. Due to the approach adopted by the RP much of my effort focussed on the first part of the responses, the Primary Source Term (PST).

7. This assessment is therefore based on the main technical submissions relating to resolution of RI-ABWR-0001 as well as any further requests for information derived from assessment of those specific deliverables, in particular in responses to the Regulatory Queries (RQ) raised.

8. Due to the scope and nature of the main technical submissions, it was necessary for Hitachi-GE to update these several times throughout the resolution of RI-ABWR-0001. My assessment is therefore based upon a particular revision of each document, plus the subsequent RQ responses. I have also sampled the latest revision of each main submission to ensure that these reflect the RQ responses and any necessary changes or amendments. Further details of the submissions that formed the basis of this assessment are given in Section 3.2 of this report.

9. This assessment allows ONR to judge whether the submissions provided in response to the RI are sufficient to allow it be closed. This is not the same as concluding that all matters associated with the UK ABWR source terms are completely resolved. In fact, I would expect further changes may be necessary as the derived source terms are applied throughout the safety case – although an important part of recommending closure is to have sufficient confidence that these changes would only be minor. Where this assessment recognises that further evidence is required these are specifically identified, such that they can be satisfactorily addressed during the remainder of GDA Step 4, or beyond, as judged appropriate within the relevant ONR technical disciplines.

1.4 Structure

10. The assessment report structure differs from for previous reports produced within GDA, in particular from the end of step Reactor Chemistry assessments (for example, Ref. 13). The reasons for these differences is that the focus of this report is much narrower – addressing RI-ABWR-0001 only, rather than a whole technical discipline, and specifically addressing the question over closure of the RI. This report has therefore been structured in such a way that it can be referenced from any subsequent assessment reports produced at the end of GDA Step 4.

2 ASSESSMENT STRATEGY

11. The intended assessment strategy for resolution of RI-ABWR-0001 is set out in this section. This identifies the scope of the assessment and the standards and criteria that have been applied.

2.1 Assessment Scope

12. This report presents only the assessment undertaken for resolution of Regulatory Issue RI-ABWR-0001, related to the definition and justification of source terms for UK ABWR during normal operations (Ref. 1). The overall aim of the assessment is therefore to come to a judgement on whether the source terms for UK ABWR have been adequately defined and whether suitable and sufficient evidence has been provided to justify the defined values.

13. Annex 1 of this report contains the full text of the Regulatory Issue and Actions. Hitachi-GE have produced a resolution plan which details the methods by which they intended to resolve the RI through identified timescales and deliverables; see Ref. 7.

14. It is also important to be clear on a number of terms used throughout this report. In the context of RI-ABWR-0001, and therefore this assessment, 'source terms' have been defined as "*the types, quantities, and physical and chemical forms of the radionuclides present in a nuclear facility that have the potential to give rise to exposure to radiation, radioactive waste or discharges*". Similarly the RI refers to 'operational states', which are defined as "*including "normal operations" and "anticipated operational occurrences". For a nuclear power plant, this includes start-up, power operation, shutting down, shutdown, maintenance, testing and refuelling*". This is an important aspect of the scope of RI-ABWR-0001, and therefore this assessment. In effect this means that the scope excludes the changes in radioactivity during an accident (e.g. after the initiating event), however it should be recognised that the conservative operating conditions at the start of an accident is within the scope of RI-ABWR-0001 (e.g. what radioactivity may be allowed within the plant under any operating rules – up to the initiating event). Accident source terms are considered within the affected ONR technical disciplines throughout Step 4, for example in the context of Regulatory Observation (RO) RO-ABWR-0066 (Ref. 10) for Reactor Chemistry.

15. As described in more detail below, this assessment report therefore does not represent the full judgement on all aspects of this topic for GDA of UK ABWR. This will be reported in the Step 4 assessment reports on a discipline basis. Therefore it is not appropriate for this report to identify any Assessment Findings or Minor Shortfalls (Ref. 8), but this report does record Residual Matters that need to be addressed during the remainder of GDA Step 4, or beyond, as judged appropriate within the relevant ONR technical disciplines. These are identified throughout my assessment and are collated in Annex 2. The reason for doing this now is to ensure that such matters are tracked and resolution can be traced. Doing so does not prejudice the eventual resolution of these.

16. This assessment does not assess how the derived source terms are used or applied within particular technical aspects of the safety case. These matters are not within the scope of RI-ABWR-0001 and should be considered within affected ONR technical disciplines throughout GDA Step 4; **[Residual Matter 1]**. Should subsequent detailed assessment of these uses suggest that the source term needs to be amended in any way this would need to be reviewed and assessed at that point.

17. A number of source term related matters remain within the scope of RO-ABWR-0006 (Ref. 9), relating to management of the source term and justification that all reasonably practicable measures had been taken to reduce radioactivity. These are not within the scope of RI-ABWR-0001 and are not assessed here; although Section 4.2.7 does

discuss some matters associated with the management of source terms data that are relevant.

18. The responses to this RI will need to be reflected into the UK ABWR PCSR. However the next submission of this document will not be made until March 2017. Assessment of the PCSR was therefore not part of this assessment; **[Residual Matter 2]**. Similarly the declared Master Document Submission List for GDA needs to reflect the responses to RI-ABWR-0001. This will also need to be reviewed at the end of GDA Step 4; **[Residual Matter 3]**.

19. Section 3.1 of this report provides a brief overview of the background to RI-ABWR-0001, which provides further context on the detailed assessment scope.

2.2 Assessment Approach

20. The assessment was undertaken by examining the evidence provided by Hitachi-GE in responding to RI-ABWR-0001. This was assessed against the expectations and requirements of the SAPs and other guidance considered appropriate. The basis of the assessment undertaken to prepare this report is therefore:

- Submissions made to ONR in accordance with the resolution plan;
- Consideration of internal and international standards and guidance, international experience, operational feedback and expertise and assessments performed by other regulators, especially their findings;
- Interaction with other relevant technical areas (where appropriate);
- Consideration of relevant outputs from any Technical Support Contractor (TSC) work;
- Raising and issuing of Regulatory Queries (RQs) as appropriate, followed by assessment of Requesting Party (RP) responses; and
- Holding technical meetings to progress the identified lines of enquiry.

21. Further details of TSC work on source terms or related matters, and its relevance to the assessment conducted, is given in Section 2.4 of this report.

22. The following subsections provide an overview of the outcome from each of the information exchange mechanisms in further detail.

2.2.1 Regulatory Queries

23. A total of 38 Regulatory Queries (RQs), comprising 116 individual queries, were raised with Hitachi-GE for the assessment of RI-ABWR-0001. These are detailed in Table 1.

24. The responses provided by Hitachi-GE to the RQs were assessed as part of this assessment. Commentary on the most important and relevant RQ responses is included in the assessment section later in this report as appropriate. The responses provided further evidence to support resolution of the RI.

2.2.2 Technical Meetings

25. A number of technical meetings with Hitachi-GE were held during assessment of the RI-ABWR-0001 responses. Approximately 17 days of main technical exchange meetings were undertaken, including a 3 day workshop to make regulatory expectations clear and to agree the Hitachi-GE resolution plan. The principal focus of these meetings was to discuss progress and responses, to facilitate technical exchanges and to hold discussions with the RP's technical experts on emergent issues.

2.3 Standards and Criteria

26. The relevant standards and criteria adopted within this assessment are principally the Safety Assessment Principles (SAP) (Ref. 3), internal ONR Technical Assessment Guides (TAG) (Ref. 4), relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites and during previous GDA assessments. The key SAPs and any relevant TAGs are detailed within this section. National and international standards and guidance have been referenced where appropriate within the assessment report. Relevant good practice, where applicable, has also been cited within the body of the assessment.

2.3.1 Safety Assessment Principles

27. The key SAPs applied within the assessment are included within Table 2 of this report. Given the nature and scope of this RI, the most relevant SAPs are those associated with production of an adequate safety case.

2.3.2 Technical Assessment Guides

28. The key TAGs applied within the assessment are included within Table 3 of this report. Given that the definition of the source term (the hazard) is such a fundamental aspect of producing an adequate safety case, there is no specific TAG which deals with this. As with the SAPs, this is encompassed within those guides that deal with producing an adequate safety case.

2.3.3 National and International Standards and Guidance

29. There are no specific international standards or guidance that deal with the definition or justification of source terms. As with the SAPs and TAGs the main international standard and guidance of relevance are those associated with production of an adequate safety case (Ref. 11).

2.4 Use of Technical Support Contractors

30. No TSC support was undertaken to review the responses to RI-ABWR-0001 directly.

31. However, during Step 3 a TSC contract was let with Studsvik to provide an independent estimate of some aspects of the likely source terms for UK AWBR (Ref. 12). The output from this work has been used as part of this assessment, as an independent estimate with which to compare the UK ABWR values.

32. In addition, ONR commissioned an independent analysis of corrosion product transport within UK ABWR (Ref. 45). The focus of this report is not in addressing matters associated with this RI, however part of this contract reviewed the Hitachi-GE corrosion product model. This model (and simplifications thereof) is discussed further as part of my assessment, as it does form part of the RP's response.

2.5 Integration with Other Assessment Topics

33. A number of other ONR technical disciplines have provided input into the overall assessment of this RI (for example Radiological Protection, Radiological Consequences and Fault Studies). This report is consistent with that assessment. Where necessary, for example for more significant assessment items, this is reported in more detail elsewhere as referenced in the assessment section of this report (Section 4).

2.6 Out of Scope Items

34. Hitachi-GE identified no items as outside the scope of the response to RI-ABWR-0001, aside from those defined by the RI itself. This means that the responses to the RI specifically exclude:

- The inventory of the fuel at any point during the fuel route (i.e. in the core and during decay cooling storage).
- The inventory due to activation of the core components, such as reactor internals, Reactor Pressure Vessel (RPV) and Reinforced Concrete Containment Vessel (RCCV).

35. Details of the scope of my assessment are described in Section 2.1, which identifies those aspects I did not assess. In particular this excludes:

- Source terms associated with plant states not considered part of normal operations.
- The use of the derived source term values within particular technical aspects of the UK ABWR safety case.

3 REQUESTING PARTY'S SAFETY CASE

3.1 Background to the Regulatory Issue

36. The assessment of the levels and control of radioactivity within the UK ABWR design was one of the first topics sampled within GDA, given the importance of this topic to the RP being able to make an adequate safety case.

37. The Step 2 reactor chemistry assessment report (Ref. 13) discusses these aspects. It notes that, while the claims made regarding radioactivity were reasonable for that step of GDA, there was no information presented in the submissions on the likely radioactivity levels in the UK ABWR. The reactor chemistry Preliminary Safety Report (PSR) (Ref. 14) did not quantify the level of radioactivity expected in UK ABWR; neither did any of the other Step 2 submissions. For example, the PSR for radiation protection (Ref. 15) defines what the sources of radiation are and claims they are conservative, but does not quantify how much there is. The basis of the UK ABWR source terms at that time was a report summarising industry experience up to 1973 (Ref. 16), which therefore did not consider the chemistry or materials proposed for UK ABWR, nor even similar designs of Boiling Water Reactors (BWR). Even given the focus of the assessment on fundamental claims at that time, serious gaps were evident.

38. Three main areas were identified where further justification and evidence would be required from Hitachi-GE, namely:

- To define and justify the source terms for UK ABWR, including how these are used;
- To demonstrate the impact of the material choices, operating chemistry and operating practices on radioactivity in the plant and to show that these reduce radioactivity So Far As Is Reasonably Practicable (SFAIRP); and
- To show that the source term information is adequately managed and controlled throughout the safety and environmental cases.

39. To address these aspects the regulators (ONR and the Environment Agency) jointly raised a Regulatory Observation (RO) related to the source terms in the UK ABWR, RO-ABWR-0006 (Ref. 9) in April 2014. This RO was associated with all three of the aspects bulleted above, including the definition (Action 1) and supporting evidence that was considered necessary to justify (Action 2) the source terms for the UK ABWR design during "operational states" and "expected events". Note that "expected events" is a term used by the Environment Agency, not ONR, and is defined in their documentation as "*events that are expected to occur over the lifetime of the plant. This does not include events that are inconsistent with the use of best available techniques such as accidents, inadequate maintenance and inadequate operation*" (Ref. 46). There is therefore overlap with how ONR would define "normal operations", which would also include minor deviations from desired operating conditions provided these are appropriately justified in the safety case (i.e. they are similar to what the IAEA terms "Anticipated Operational Occurrences").

40. Other actions under RO-ABWR-0006 deal with the management of source term information and the justification that radioactivity is reduced SFAIRP, but as these are not within the scope of RI-ABWR-0001, these are not discussed further in this report.

41. Responses to Actions 1 and 2 were received during January 2015 (Refs 17 and 18). These responses are not discussed in detail here, except to note that the approach within these reports was almost exclusively based on theoretical calculations, contained very limited justification and no recourse was made to plant Operating Experience (OPEX).

42. The assessment of these submissions indicated a number of significant shortfalls in the responses. A number of regulatory interventions took place over subsequent months to clarify regulatory expectations, including letter REG-HGNE-0077R (Ref.19) and a detailed ONR presentation of feedback (Ref. 20). In summary, the main deficiencies with Refs 17 and 18 were:

- The approach taken, of calculation of the source terms, meant that there were inherently many assumptions, some of which appeared to impose a significant sensitivity on the results. These weren't appropriately justified;
- The definition of an "average" source term did not cover all potential transients, operational occurrences or operations expected at the plant, as requested in RO-ABWR-0006;
- The amount of fixed radioactivity (contamination) was inadequately defined and substantiated, with no supporting evidence;
- The scope of the defined source terms was incomplete with some significant aspects missing;
- The corrective factor applied when the source terms are used for specific purposes did not appear to be conservative;
- There was no link between the defined source terms and the extant UK ABWR safety and environmental cases; and
- A suitably robust demonstration and justification for the adequacy of the defined source terms was not provided.

43. Given the significance of the identified regulatory shortfalls the definition and justification for the UK ABWR source terms during normal operations was escalated to a Regulatory Issue, RI-ABWR-0001, in line with the Guidance to Requesting Parties (Ref. 5) in June 2015.

44. RI-ABWR-0001 is given in Annex 1. The expectations of the RI are essentially the same as those for RO-ABWR-0006 Actions 1 and 2. In fact, the RI states that the responses should meet the regulatory expectations defined in the RI, address the regulatory expectations of RO-ABWR-0006 (Ref. 9) and address the feedback given in letter REG-HGNE-0077R (Ref. 19). Further regulatory guidance on source terms more generally was provided to Hitachi-GE during development of their resolution plan (Ref. 20), which contains further information on broader regulatory expectations of relevance.

45. ONRs expectations were clear in that, in summary, an adequate definition of the UK ABWR source terms should:

- Cover all significant radionuclides;
- Cover all systems which are expected to contain radioactivity;
- Cover all operational states;
- Cover all appropriate sources of radioactivity within the plant, including mobile and fixed sources;
- Consider how the nature and quantity of radioactivity within the pant may change over time;
- Cover all aspects of the safety or environmental case for UK ABWR;
- Be consistent with how the defined source terms are used by, and support, these cases; and
- Be consistent with the design and operations of UK ABWR.

46. Similarly an adequate justification should:

- Provide an appropriate degree of robust supporting evidence for the defined source terms;

- Cover the full scope of the definition, but be targeted towards those radionuclides, systems or operations which have the highest safety or environmental impact; and
- Be demonstrated to be appropriate for the UK ABWR and consistent with the extant safety and environmental cases.

3.2 Submissions Provided in Response to the Regulatory Issue

47. The basis, derivation methodology and calculated values for the radioactive Source Term for the UK ABWR is presented in a suite of documents, as shown in Figure 1 below. The document structure uses a tiered approach that includes a high-level Strategy Report, a Source Term Manual, Supporting Reports and Source Term value data sets. This figure also shows how the various documents are linked.

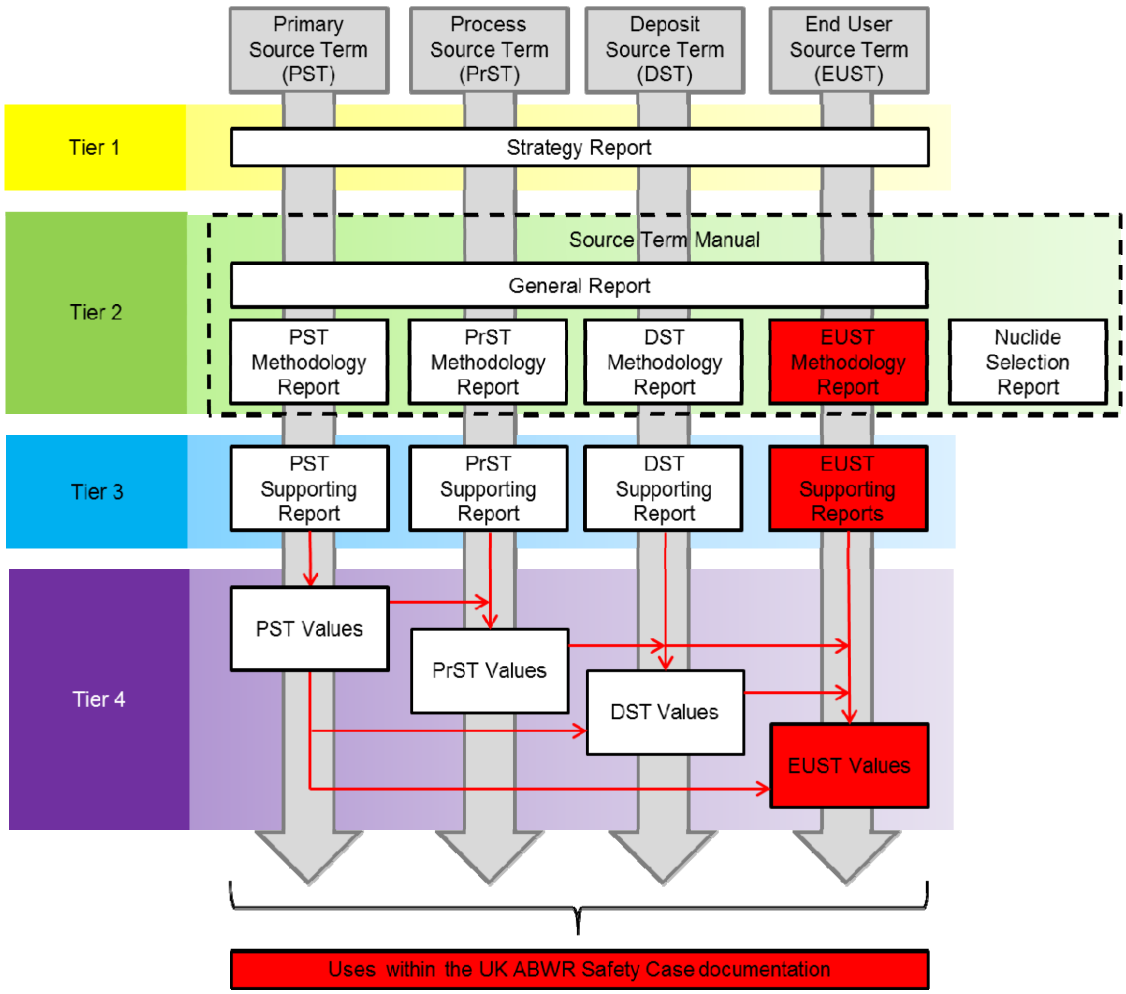

Figure 1: Overview of the RI-ABWR-0001 Submission Structure

48. Note that those boxes shown above with a red background were produced as part of the overall source terms work by Hitachi-GE, but were not within the scope of RI-ABWR-0001 and are not assessed as part of this report.

49. The various submissions can be split according to either content (vertically in Figure 1) or source term category (horizontally in Figure 1). Splitting by content leads to four tiers;

- The first tier comprises a single high-level source term “Strategy Report” (Ref. 22) which describes the role of the source term in supplying data to support the safety and environmental cases for the UK ABWR during GDA, site specific licensing and permitting studies, and future reactor commissioning, operation and decommissioning phases. This report also describes how the suite of documents will be maintained throughout the lifecycle of the plant.

- The second tier is described collectively as the "Source Term Manual" which consists of six documents: a "General Report" (Ref. 23), a "Methodology Report" for each of the four categories of source terms derived; Primary Source Term (PST), Process Source Term (PrST), Deposit Source Term (DST) and End User Source Term (EUST) (Refs 24 to 26). The general report outlines the technical basis of the source term and the structure adopted by each of the methodology reports which themselves provide a detailed account of how each category of source term is derived. Also included is a "Nuclide Selection" report which justifies which nuclides are relevant for UK ABWR (Ref. 27). The source term manual is not in itself a single report, but this suite of six documents.
- The third tier comprises a suite of supporting reports which provide detailed evidence that underpins the information presented in the source term manual. In the context of this assessment there is one "Supporting Report" for each source term category (Refs 28 to 30). Examples of topics covered in the supporting reports are: OPEX data selection methodology, key assumption sets, derivation and justification data and key radionuclide selection methodology.
- The fourth tier of the Source Term document suite presents the value data sets. These provide the calculated radionuclide concentration values for the full range of source term categories presented in the methodology reports (Refs 31 to 33). Radionuclide values are presented in tabular form.

50. The second means of splitting the responses is in terms of source term category. These are:

- Primary Source Term (PST) - defined as the level of radioactivity within the nuclear boiler system in UK AWBR (i.e. within the Reactor Pressure Vessel (RPV)). More specifically the PST quantifies the concentration of each radionuclide present in the reactor water and reactor steam that leaves the RPV. As can be seen from Figure 1, the PST is key to the UK ABWR source terms as it is an input to all other defined source terms.
- Process Source Term (PrST) - defined as the level of radioactivity within each of the systems in the UK ABWR. The PrST quantifies the concentration of each radionuclide present within circuit pipes, ancillary equipment and plant systems. The PrST essentially uses a mass balance approach to determine how radioactivity that exit the NB system changes due to processes such as decay, removal or accumulation.
- Deposit Source Term (DST) - defined as the level of activity deposited within each of the systems in the UK ABWR. The DST quantifies the concentration of each deposited radionuclide on internal pipework, ancillary equipment, plant systems and fuel pins. The DST uses both the PST and PrST as inputs.
- End User Source Term (EUST) - defined as the final level of radioactivity considered for a particular assessment within a technical area of the safety and environmental case for the UK ABWR. The EUST uses relevant aspect of the PST, PrST and DST. This category of source term was not within the scope of RI-ABWR-0001, so is not assessed in this report.

51. Further relevant details on the contents of each of these submissions are contained with the assessment section of this report (Section 4). As noted earlier these submissions were updated several times throughout the assessment period. Refs 22 to 33 are those revisions which form the basis for the assessment. The most current revisions at the time of preparing this report were also reviewed, as described later in my assessment.

52. In addition to the submissions detailed above, which formed the basis for Hitachi-GEs resolution plan, responses to the various relevant RQs also formed the basis for this assessment. These are referenced throughout Section 4, and given in Table 1.

3.3 Overview of the Requesting Party's Responses

53. The suite of documentation submitted by Hitachi-GE to resolve RI-ABWR-0001 is described above. The RP's intention in submitting this documentation was to demonstrate that they have derived an appropriate suite of source terms for UK ABWR, and these are supported by suitable and sufficient justification. The RI responses are completely different from the original RO-ABWR-0006 responses (Refs 17 and 18), in terms of scope, content and approach. An overview of these responses is described below.

3.3.1 Scope and Content

54. As noted previously, the approach taken by Hitachi-GE is to categorise the source terms into PST, PrST, DST and EUST with appropriate links to capture the inputs and outputs necessary. Collectively, this captures how radioactivity is generated, moves and accumulates throughout the various plant systems.

55. In addition to categorising the source terms, the RP's approach also splits the source term into the various nuclide categories, based on their generation mechanism; namely Fission Products (FP), Corrosion Products (CP), Actinides (ActP) and Activation Products (AP). Given their different generation mechanisms and subsequent behaviour in the plant, the definition and justification approach differs for each.

56. For each category of source term and nuclide Hitachi-GE considers all phases of the operational cycle (i.e. system start-up, power operation, normal hot stand-by, shutdown and outage). Not every phase is explicitly defined however, as the RPs approach is to simplify by using bounding values for some phases of operation (for example the shutdown spike bounds what is expected during start-up).

57. In addition, a range of derived values are calculated, as follows:

- Best Estimate (BE) - this is what Hitachi-GE describe as "*an overall best estimate of the Source Term expected in the UK ABWR over a defined period. This will be a representative condition that is realistic and reasonable*". The RPs intention in defining such a source term value is to ensure that a value is produced which can be used when more realistic estimates of radioactivity within systems is needed, so as not to result in over-specification of plant systems. The RP notes that the BE value can be used for areas such as disposability assessments and routine discharges.
- Design Basis (DB) - this is what Hitachi-GE describes as "*a conservative maximum value for the Source Term which can be considered to be a bounding limit for the plant design*". In ONR terminology this would therefore align with the operating rules for UK ABWR such that it is expected that this level would not be exceeded during operation, even when "expected events" such as single pin fuel failures occur. This DB value is therefore important for key safety related applications such as shielding calculations or radiological consequences of accidents to ensure that doses to the operators and public are minimised.
- Cycle Average (CA) – Hitachi-GE have also chosen to derive a source term which defines the radioactivity over an entire fuel cycle (i.e. 18 months), including start-up, steady power operation, shut-down and refuelling outage phases. Both BE and DB CA values are defined, and will include "expected events" such as fuel pin failures and unplanned shutdowns. The CA values are used as inputs to some PrST and DST source terms.

58. Given the nature of BWRs, namely with boiling of the coolant, it is important that the source terms considers this phase change appropriately. The approach taken by the RP in this regard is to define the concentration of a given nuclide according to its solubility and volatility. This accounts for the fractions of radionuclides which remain in the water within the NB system and that which is transported with the steam. The soluble and insoluble fraction are also important in determining the deposited radioactivity.

59. The combination of these factors is shown in Figure 2, below. While this figure only considers the PST similar figures could be prepared for the PrST and DST. What this figure demonstrates is that the approach taken in the RI submissions means that Hitachi-GE have defined a wide range of source term values. For example the concentration of any given soluble nuclide in the reactor water, under a particular operating phase can be found.

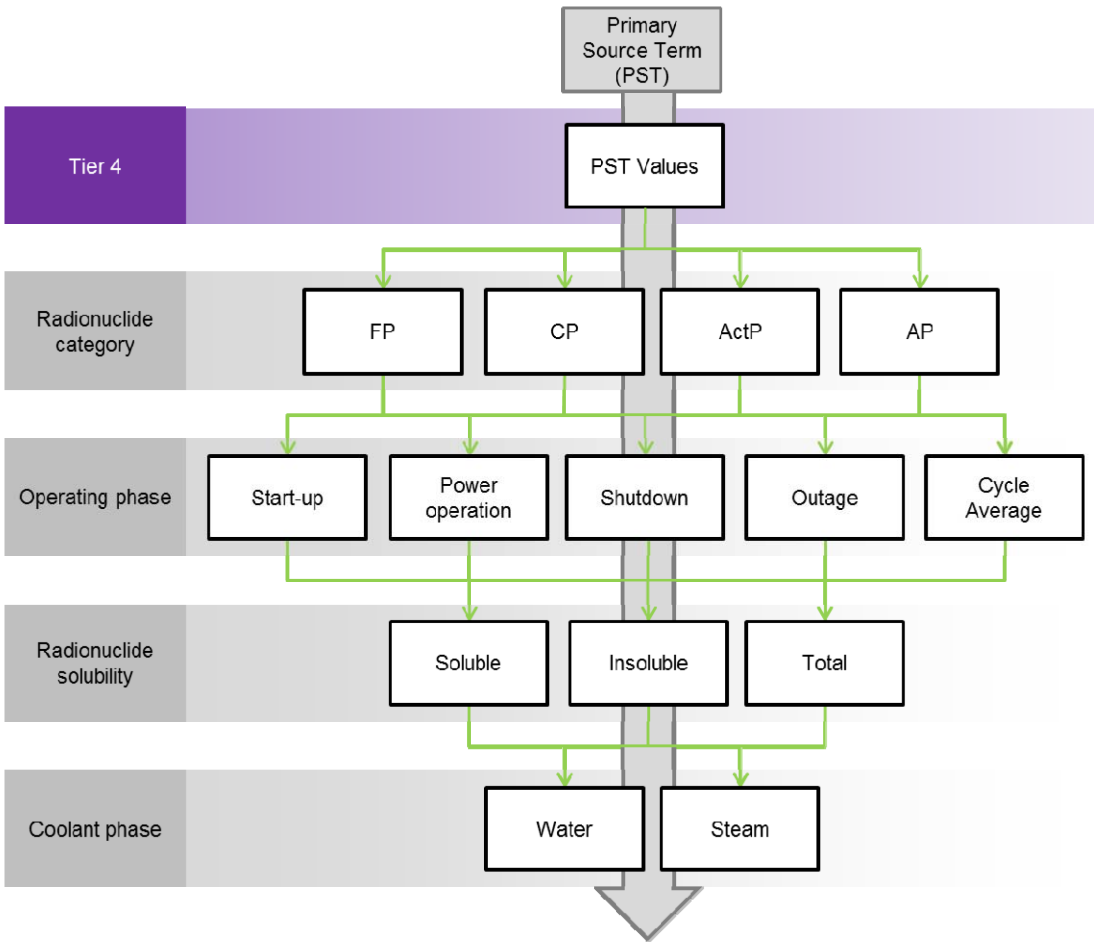

Figure 2: Overview of the PST Approach

3.3.2 Approach

60. The approach taken in the original response to RO-ABWR-0006 (Refs 17 and 18) was to use theoretical calculations. This was changed in responding to the RI.

61. There are some differences between the PST, PrST and DST, but the basis of the approach is to use statistical analysis of OPEX data gathered from existing BWR plants. Hitachi-GE select a range of OPEX to provide the broadest data set that they

judge pertinent to the design and operation of the UK ABWR (influencing factors include proposed water chemistry and material selection). The limitation with this approach is that the OPEX data of sufficient quality and providence is typically limited to a sub-set of the most significant radionuclides, which does not cover all of the nuclides necessary for UK ABWR. These gaps in the OPEX data are filled using verified and validated models, computer codes and supporting calculations. The underlying assumptions and parameters are defined and justified within the various supporting reports, alongside the OPEX selection methodology.

62. Hitachi-GE contends that the use of OPEX for derivation provides a degree of justification. However, additional independent justification for the source term values is also provided. The approach to this justification varies much more than for the definition aspect, including from nuclide to nuclide, but some combination of the following methodologies are employed, in this order of preference:

- Further OPEX data analysis using additional (independent) plant information;
- First principle computer codes and modelling calculations incorporating the proposed neutron flux, water chemistry and material selection for the UK ABWR;
- Published literature; or
- Sensitivity analysis.

4 ONR ASSESSMENT

63. This assessment has been carried out in accordance with HOW2 guide NS-PER-GD-014, “Purpose and Scope of Permissioning” (Ref. 1).

4.1 Scope of Assessment Undertaken

64. The scope of my assessment is described in Section 2.1, alongside the description of the submissions which formed the basis for that assessment in Section 3.2. The overall scope of this assessment is to provide responses to the following aims:

- To provide the assessment which underpins the judgement made in closing RI-ABWR-0001;
- To serve as a record of the scope of the assessment of RI-ABWR-0001, and therefore the boundaries of the judgements made; and
- To identify any associated residual matters which may need to be satisfactorily addressed during the remainder of GDA Step 4, or beyond, as appropriate.

4.2 Assessment

65. This section describes my assessment of the RI responses.

66. I have structured my assessment around the responses provided by Hitachi-GE. I therefore consider each category of source term in sequence; PST, PrST and DST. The bulk of my assessment has concentrated on the PST as this feeds into all the other areas, therefore the level of rigour needed here is greatest. I consider definition and justification separately to align with the RI Actions, although they are intrinsically linked.

4.2.1 Radioactivity Generation, Transport and Behaviour in BWRs

67. The following section provides a brief overview of radioactivity in BWRs. This is to give context to the assessment that follows, and the RP’s responses need to be consistent with this understanding.

68. Overall, how the radioactivity that is produced in an operational BWR behaves is a complex interaction between many variables. Some of these are well understood, whereas others can only be empirically followed based on plant experience. Importantly there is no single unified understanding of all aspect of radioactivity transport, in particular relating to how any changes to operating conditions may impact. This therefore makes defining and justifying the source terms for UK ABWR less straightforward, because although ABWRs have been operated providing useful data it is often not directly applicable to UK ABWR because of differences in the design or operating conditions. This latter point is particularly important in UK ABWR due to the changes to the operating chemistry (from Normal Water Chemistry (NWC) to Hydrogen Water Chemistry (HWC)) and material choices (away from carbon steels). It is unnecessary for the scope of this assessment to describe the impact of these changes in detail, although they are mentioned briefly below and considered as part of my assessment.

69. However, the same fundamental processes that govern radioactivity behaviour will be true in UK ABWR, as for other BWRs:

- The most important is that the water coolant is allowed to boil inside the Reactor Pressure Vessel (RPV) and the steam that is produced is transferred to the turbine, condensed, purified and returned to the RPV as feedwater. This means that any volatile impurities within the coolant can be carried within the steam to the turbine systems, as can any material entrained within the steam even though it is dried to very low levels.

- Due to the volume of the RPV and steam flow rate the residence time within the RPV is large. This has two effects that are important for radioactivity; firstly, water flowing through the core is exposed to ionizing radiation (especially neutrons) and a wide variety of radiolysis products are produced. This has a pronounced effect on the environmental conditions within the coolant, which become oxidising (although chemistry controls, particularly hydrogen addition, attempt to reduce this). The environment can affect the behaviour of radionuclides, in particular their volatility, solubility or removal efficiency. Secondly, this residence time exposes the coolant to the radiation field within the core, which causes activation of species within the coolant (for example metal corrosion products) or of the coolant itself (for example O^{16} in water to N^{16}).

Fission Products

70. Fission Products (FP) are produced within the fuel elements and will remain within the fuel cladding provided there is no damage or leaks (except for tritium which can diffuse through zirconium, although this is small compared to that produced from H^2). The concentration in the coolant will never be zero as some small amount of leakage inevitably occurs and fission products can also be produced by trace levels of uranium contamination outside the fuel rods, either from manufacture or previous fuel damage (known as tramp uranium). Thus the factors which most directly influence FP radioactivity are the extent of fuel leakage or contamination. The type of damage occurred is important in determining the extent of FP release from the fuel, and also the amount of the different nuclides released (in fact FP coolant ratios can help in determining the state of the core).

71. A range of FPs are produced in BWRs, including the noble gas isotopes of krypton and xenon, soluble species such as strontium and potentially volatile isotopes of iodine. Their behaviour in a BWR can therefore vary. Of particular note are the noble gases which are effectively removed from the coolant by the boiling in the core and are transported with the steam. Other species will tend to stay with the water and can therefore be subject to clean-up, but the extent of this varies depending on volatility and speciation. Material choices have negligible effect on the behaviour of FPs, and the operating chemistry is limited to affecting the speciation and volatility; the overall behaviour and concentration is mainly determined by the plant design and operations.

Actinides

72. Actinides are generated as part of the fission process, similar to other fission products, but are released into the coolant due to processes that lead to contact between the water and fuel itself. They are treated as a separate category of nuclides, because they are only sparingly soluble and tend to deposit on surfaces, even suspended solids. Their behaviour under the operating chemistry for UK ABWR may vary. Due to their production route the main impact on actinide concentration is likely to be fuel failure rates (via tramp uranium). The long lived nature of the nuclides means they tend to be of concern for waste disposal and internal doses.

Corrosion Products

73. Activated corrosion products (CP) are produced when metal impurities within the coolant are activated within the core. The vast majority of these impurities are produced by corrosion of the feedwater plant, with the exception of cobalt which also has important contributions from cobalt based alloy releases. Even though the release rates themselves may be very small the large surface areas and feedwater flow rate over a fuel cycle may introduce 10's of kg of iron or nickel into the coolant, for example. The amount of each corrosion product nuclide is primarily a function of the alloys used within the feedwater system, the clean-up system efficiency and their

behaviour within the core. This latter point is important because minimising the residence time of the corrosion product within the core will directly reduce the level of activation.

74. CPs will deposit on the fuel surfaces (and to a much lesser extent on other in core surfaces), deposit on out of core surfaces (such as pipework), be carried within the coolant (both soluble and insoluble) or become trapped within clean-up systems. The chemistry conditions will change how the different corrosion products distribute between these different locations by changing the solubility and form of oxide deposits produced. CPs have low volatility so most of the radioactivity will remain in the water phase, although a small amount of transfer to steam can occur through droplet carryover. Out of core deposits are particularly important as Co^{60} is the main source of dose to workers during outages, whereas the majority of CP radioactivity in a BWR will tend to accumulate on the fuel surfaces.

Activation Products

75. Activation Products (AP) are generated from the activation of the reactor coolant or entrained impurities. The originating source differs for each radionuclide, although in many cases they are formed via the neutron activation of naturally occurring parent nuclides. The range of nuclides covered by the AP category is wide and covers volatile, soluble and insoluble nuclides. Therefore APs can remain in the water or partition with steam, depending on the particular nuclide. The most important determining factor in the concentration of the AP nuclides is the parent nuclide concentration, especially where the source is from an impurity.

76. The most important AP in BWRs is N^{16}, which is produced by neutron activation of O^{16} in the coolant water molecules. This nuclide dominates doses during operations, but rapidly decays once the reactor is shutdown (half-life of 7 seconds). The amount of N^{16} is a function of reactor power, but its distribution is directly influenced by the operating chemistry, with more reducing conditions changing the chemical form to volatile species which are transferred with the steam to the turbine.

4.2.2 Approach and Scope

77. In this section of my assessment I consider whether the RP has adopted a suitable approach and scope to resolve RI-ABWR-0001. The approach and scope implemented is described in Section 3.3 of this report, so is not repeated in detail here.

78. Paras 43 and 44 outline my regulatory expectations. When comparing these with the RP's responses I am content that the fundamental approach and scope adopted by Hitachi-GE is appropriate to respond to these. Inevitably, there are some more detailed questions regarding methodologies and evidence which I consider further in subsequent parts of my assessment, but the change to an OPEX led approach including specific consideration of both definition and justification, covering the scope suggested, is reasonable.

79. This change in approach and scope also removes a number of the deficiencies identified in the original RP responses on source terms (Refs 17 and 18), as defined in letter REG-HGNE-0077R (Ref.19). Importantly these changes have resolved the concerns regarding:

- The sensitivity of and justification for the assumptions used within the calculations used to derive the source terms – the approach is now OPEX led. While assumptions are still necessary, the case presented is more transparent, less open to challenge and questions now become centred on the representivity of the selected OPEX to UK ABWR;

- The definition of an "average" source term did not cover all potential transients, operational occurrences or operations expected at the plant, as requested in RO-ABWR-0006 – individual plant states are now explicitly considered; and
- The scope of the defined source terms was incomplete with some significant aspects missing – the scope now meets my expectations, and is aligned with the general behaviour of radioactivity expected in BWRs described in Section 4.2.1.

80. An important linkage made by the RP in their responses to RI-ABWR-0001 is with the requirements of the safety case. The strategy report (Ref. 22) makes this link early in the process and this is carried through to the other responses. Ensuring that the definition and justification is aligned with the safety (and environmental) cases for UK ABWR, ensuring appropriate targeting and proportionality, is a significant improvement.

4.2.3 Nuclide Selection

81. Before describing my assessment of each of the source term categories, I first consider whether the RP has identified all of the relevant nuclides. This is detailed in the nuclide selection report (Ref. 27). The purpose of Ref. 27 is to detail and justify the selection methodology used to identify those nuclides which form the UK ABWR source terms. The nuclides selected here are then considered further throughout subsequent stages of the source terms.

82. The original response to RO-ABWR-0006 (Ref. 17) used the Oak Ridge Isotope GENerator (ORIGEN) code to quantify a near exhaustive list of the radionuclides likely to be present within UK ABWR. This resulted in a list of over 600 nuclides, many of which were at immeasurably small concentrations. This in itself was one of my concerns with the original responses, in that Hitachi-GE were not focussing on those nuclides which were significant to the safety (or environmental) case. Hitachi-GE make use of this original ORIGEN analysis as a starting point for the nuclide selection process, however this is screened to leave only those nuclides which the RP considers to be "significant".

83. The approach to screening considers the requirements of each of the EUSTs. These are shown below, including what are the key aspects of the safety (and environmental) case covered by each. It is worth noting that the "environmental discharges" EUST also covers releases during an accident (i.e. for radiological consequence analysis).

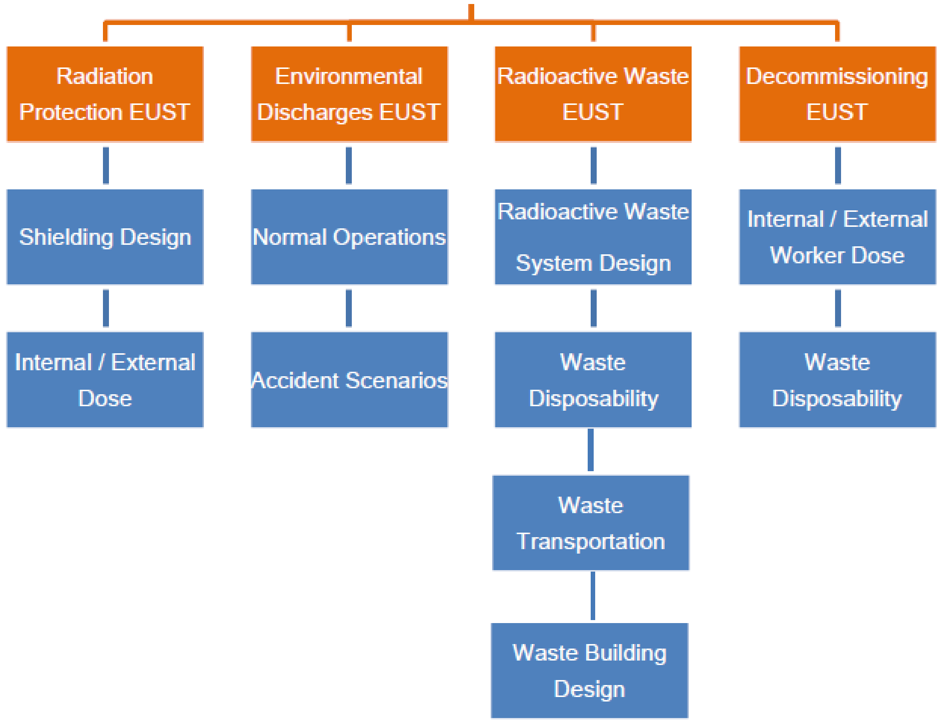

Figure 3: Identified safety and environmental areas covered by each EUST

84. For the FP, CP and ActP the screening process uses various criteria and limits. For APs the process is based more on judgement and OPEX, as the concentration of these nuclides is mainly a function of impurity ingress. A number of other factors are also considered including regulatory requirements and benchmarking to previous source terms (including for other reactor types). In addition a number of specific assumptions are applied for each EUST, to account for factors such as decay. The output from this process is therefore a list of significant nuclides for each EUST, some of which overlap, but when taken collectively comprise the nuclides considered significant. Using this process Ref. 27 identified a total of 96 nuclides.

85. Based on my assessment of Ref. 27, although I could challenge some of the factors and assumptions for each EUST list, I was content that the consolidated list of nuclides for UK ABWR was generally reasonable and certainly considered those nuclides of most importance. However, I requested further evidence to justify the omission of some other nuclides in RQ-ABWR-0691 (Ref. 34). The response provides the rationale for this decision, but also conceded that Na^{24} should be included in the list of nuclides due to its dose significance and generation due to the use of noble metal chemistry in UK ABWR. I further queried why P^{32}, Cu^{64}, Kr^{83m} and W^{187} were excluded given their use in ANSI/ANS-18.1-1999 for defining light water reactor source terms, in RQ-ABWR-0797 (Ref. 34). The response argues that the concentration of these nuclides will be lower in UK ABWR than in the plants used to derive the ANSI standard values due to design improvements such as material choices. On this basis I was content to accept that the inclusion of these additional nuclides would not have a large impact on the UK ABWR source terms.

86. In a similar manner RQ-ABWR-0682 query 1 (Ref. 34) asked why some APs were not considered. These were screened out on the basis that Hitachi-GE considered tem to be “masked” by N^{16}. However, this neglects their differing behaviour and half-lives. I was particularly concerned by C^{15} which is a high energy γ emitter (see Ref. 12). The response makes a reasonable case for the continued exclusion of most of these, but

also concedes that C^{15} will be included in the nuclide list given it contributes approximately 20% of the main steam line radiation dose.

87. The latest revision of the nuclide selection report (Rev. 2), Ref. 35, identifies 100 nuclides of significance; the 96 identified in Rev. 0 plus C^{15} and Na^{24} as noted above and two further which are included as surrogates for some other nuclides for which OPEX data does not exist. The consolidated list of nuclides is therefore given below:

[redacted]

Figure 4: Identified nuclides for UK ABWR

88. I have also compared this list to the nuclides considered in a range of other sources, including the Sizewell B, AP1000® and UK EPR™ safety cases, ANS/ANSI-18.1-1999 and other relevant literature (for example, Refs 12, 16, 36 and 37) and guidance (Refs 38 and 39). While there may be some additional nuclides that could be considered, these are likely to be both of low safety significance and/or low concentration.

89. In RQ-ABWR-0740 I queried how this final consolidated list of nuclides would be used within each EUST. While I have not assessed the EUST documents, it became apparent that while the source term documents use this complete list of 100 nuclides, each EUST goes back to the relevant smaller subset in the nuclide selection report. The response to this RQ (Ref. 34) confirmed this. I am not convinced of the benefit in

this approach, but the RP notes that this position is reviewed for each EUST at time of use. As this is outside the scope of my assessment, I consider it would be beneficial for each end user to confirm their contentment with this; **[Residual Matter 4]**.

90. Overall, I judge that Hitachi-GE have identified and considered the most safety significant nuclides as part of their responses to RI-ABWR-0001.

4.2.4 Primary Source Terms

91. The Primary Source Term (PST) is defined by Hitachi-GE as "*the level of mobile activity within the Nuclear Boiler system (NB) and thus, quantifies the concentration of each mobile radionuclide present in the reactor water and reactor steam in the NB*". This therefore covers the soluble, insoluble and total concentration of the identified FPs, CPs, APs and ActPs for each mode of operation. A Best Estimate (BE), Design Basis (DB) and Cycle Average (CA) value for each radionuclide present in the PST is defined.

92. The PST documentation consists of a methodology report (Ref. 24), supporting report (Ref. 28) and values (Ref. 31).

93. As described previously, the PST is the first part of the overall methodology used by Hitachi-GE and hence is an input to all other aspects of the defined source terms. A large part of my assessment effort therefore focused on the PST.

Methodology

94. The methodology used by the RP to define the PST is described in Ref. 24, and further detailed in Ref. 28. In particular Ref. 28 includes additional details of the plant OPEX and the selection process. The methodology differs for each nuclide category, so I assess each of these separately below. First, I consider a number of general points which apply across the PST.

General

95. One of the main differences for UK ABWR is the choice of operating chemistry. While the choice of suitable OPEX should account for the impact of this choice on nuclide concentrations, I asked if Hitachi-GE had evidence for the impact of this on the timings or profiles of changes that might be expected (RQ-ABWR-0684 query 1 (Ref. 34)). The response cites Electric Power Research Institute (EPRI) data for US BWRs which shows that the effect of HWC is not important when considering the scale of any changes. It does have an effect on the potential timings of shutdown transients. The RP argues that this need not be considered due to the assumptions used in defining the shutdown spikes, discussed later in my assessment. I am satisfied with this response.

96. I asked a similar question but relating to the effects of noble metals or zinc addition in query 5 of RQ-ABWR-0685 (Ref. 34). The response refers to the Topic Reports on NMCA (Ref. 42) and Zinc (Ref. 43). These reports were not produced to respond to RI-ABWR-0001 and consider matters outside of the scope of this present assessment. However, I am not convinced that these reports provide sufficient information to respond to this query. On the basis of information presented therein and elsewhere I do not at present judge the impact to be significant, but consider that this needs to be resolved and suitable and sufficient evidence provided as part of the wider safety case; **[Residual Matter 5]**.

97. The material choices for UK ABWR are also actively being considered at present (via RO-ABWR-0035). The choice of materials will affect the amount, composition and behaviour of radioactivity. Broadly the RP makes material assumptions within the RI-ABWR-0001 responses that it considers to be reasonable; however where uncertainty

remains in the final choice a conservative approach is taken. The base material choice is taken to be carbon steel, although the Clean-up Water system (CUW), Condensate Storage Tank (CST) and Spent Sludge system (SS) are stainless steel. I am content that in the context of RI-ABWR-0001 these are reasonable assumptions to make. I consider specific points around material choices as part of my detailed assessment.

98. In the second query of RQ-ABWR-0684 (Ref. 34) I ask Hitachi-GE what they consider to be the main sensitivities associated with the methodologies used in the responses to RI-ABWR-0001. The response considers each nuclide category in turn and identifies a number of assumptions, identifying those with the RP considers to be key. I judge the lists provided to be reasonable in terms of content but I could challenge the RP's consideration of significance (in terms of impact and uncertainty). For example the RP notes that the DB FP methodology has a low uncertainty, but I challenged this and it was subsequently changed (see Para. 122). Notwithstanding this point there is generally a good alignment between those assumptions identified by the RP and the subject of many of my RQs discussed in subsequent sections of my report. Overall, I take comfort that the RP has considered many of these assumptions as part of the derivation and justification processes and has satisfied itself that the approach remains fit for purpose.

<u>Corrosion Products</u>

99. Hitachi-GE rationalise the normal operating phases of relevance to CPs to power operation and shutdown. It is claimed, with supporting arguments and evidence, that these two phases bound the others. I am content that this is a reasonable approximation to make given the mechanisms associated with CP behaviour.

100. In order to define the likely CP concentrations in UK ABWR the RP first identifies a range of plant OPEX. Directly measured OPEX data were initially sourced from a large number of Japanese ABWRs, Japanese BWRs and worldwide BWRs. This OPEX dataset is then screened to improve the quality, usefulness and relevance which results in a reduction in the number, but not range of plants considered for power operation phases. Importantly this screening removes plants which are known to have a high cobalt burden and those which do not operate within the expected UK ABWR feedwater iron control band of 0.1 to 1 ppb. A similar screening process is applied to the shutdown OPEX, using the same criteria. While less plants are included, all of the plants used to define the shutdown phase are included within the power operation dataset.

101. Despite this large dataset, not all CP nuclides identified for UK ABWR have OPEX data as this tends to focus on those nuclides of most significance. There are also differences in the quality and quantity of data. All values are for reactor water concentrations, not steam. The RP therefore uses methods which are broadly similar but do differ to define the concentrations given these constraints:

- Measured CP during power operation - the approach is simply to perform a statistical analysis of the data, with the BE value given as the [redacted] and DB value equal to the [redacted] Soluble and insoluble data is treated separately. Steam concentrations are defined by assuming a steam carryover of [redacted] % for BE and [redacted] % for DB.
- Unmeasured CP during power operation – the approach is to scale the unmeasured radionuclides to measure radionuclides based upon the release rate (using material composition, corrosion rate and surface area) and an activation calculation using ORIGIN, thus creating a "fingerprint" for UK ABWR. Co^{60} is used as the reference. Where data is not available for the soluble or insoluble split Hitachi-GE assume it to be the same as an analogous nuclide (e.g. Fe^{55} is the same as Fe^{59}). The same steam carryover fractions as above are assumed.

- Measured CP during shutdown - the peak (total) values for nuclides in the OPEX data are determined. The soluble and insoluble split is determined for Co^{60} from 5 plants and this ratio is applied to all nuclides. BE values are the [redacted] and DB are [redacted]. Steam carryover uses the same assumptions as previously.
- Unmeasured CP during shutdown – the same approach as for the unmeasured power operation phases is used. The only difference is that the Co^{60} value that is used as reference is the shutdown peak value (using the same statistical approach as above for BE or DB).
- Shutdown spike removal – as described above the shutdown values are peak values but it is also necessary to define the removal half-life. This is done by analysing the behaviour of the plants selected for shutdown conditions. This leads to values of [redacted] and [redacted] days for BE and DB respectively.

102. During development of the CP methodology (in later revisions of Ref. 24 and 29), Hitachi-GE also recognise that some other nuclides are generated to some extent by corrosion. These are included in the appropriate method described above. However, they are not listed as CP by the RP, but included in the FP category. I consider this distinction arbitrary. I welcome the consideration of this source, although it is relatively minor compared to that due to fission and affects less significant nuclides only.

103. In general, I consider the methodologies described above to be a reasonable starting point to adequately define the CP concentrations for UK ABWR. I did query a number of aspects of this methodology in more detail as part of my assessment. These RQs tended to focus on testing the validity and sensitivity of assumptions.

104. Firstly, the assumption that the operating chemistry does not have a significant impact on the concentration of CP was tested in RQ-ABWR-0687 query 6 (Ref. 34). The response demonstrates that there is no significant relationship that can be established from plant OPEX on the impact of the operating chemistry, with the exception of those plants which operated with very low or very high feedwater Fe levels. This observation on Fe is further supported by published papers. I am therefore content that the screening criteria were reasonable. I note that the importance of feedwater Fe is obvious and Hitachi-GE need to ensure this is reflected in the safety case through suitable and sufficient evidence; **[Residual Matter 6].**

105. A number of important assumptions are made by Hitachi-GE in their methodology for determining the concentrations of unmeasured CP during power operation. Queries 2 and 3 of RQ-ABWR-0687 (Ref. 34) were raised to test the assumptions on material compositions and corrosion rates respectively. In response the RP undertakes a number of sensitivity analyses, changing both of these assumptions. The conclusions are that:

- material composition (within the range examined) does not have a significant effect in CP concentrations;
- increasing the feedwater iron concentration has a proportionate effect on CP concentration; and
- changing the stainless steel corrosion rate has a much smaller impact on the resultant CP concentrations.

106. I do not agree with the first conclusion as this is only true because the RP effectively only considers changes downwards from that assumed for UK ABWR. In effect I consider that the analysis shows that both corrosion rate and material composition are important (in fact the key driver is the product of these, the elemental release rate). I am satisfied that the assumptions made by Hitachi-GE in their methodology are reasonable, when considering the other uncertainties that are likely within this analysis. While they could be challenged and the results show a high degree of sensitivity to these, they are already conservative and the DB values offer a further degree of

conservatism over and above that which could be influenced by reasonable changes in these assumptions.

107. In RQ-ABWR-0685 query 1 (Ref. 34) I questioned the use of different carryover values for BE and DB concentrations of CP (and FP). In their response, the RP explained how this assumption is based upon the statistical analysis of moisture carry-over in BWRs. The design value for UK ABWR (based on the steam dryer and separator design) is 0.1%, but OPEX (from JABWRs) suggests it is much lower than this. The OPEX analysis bound this design value. I judge these arguments to be reasonable. However, the response notes that the moisture carryover can vary throughout an operating cycle, with increases at the end due to increased core flow for reactivity management. I consider it would be beneficial to confirm the scale of this effect expected for UK ABWR with the RP; **[Residual Matter 7]**.

108. Query 4 of RQ-ABWR-0685 (Ref. 34) concerns the behaviour of Cr^{51} during a shutdown. This nuclide will behave differently to other CP nuclides during a shutdown in UK ABWR, when the environment changes from reducing to oxidising leading to an increase in solubility (similar to a PWR). The response provides further details on the OPEX and confirms that the definition is based upon plants that operate under HWC, noble metals and zinc addition. While I am content that the total peak value may be reasonable, albeit with greater uncertainty than the other CP, I am not confident that the stated split of soluble and insoluble activity is correct. This is based on the Co^{60} data, which will be different. This is particularly true for Cr^{51} but is also true for the other defined CP (except Co^{60} itself). I specifically queried this point in RQ-ABWR-0687 query 4 (Ref. 34). Hitachi-GE argues that the use of Co^{60} data is conservative as the change observed during shutdown is greatest for this nuclide; from [redacted]% soluble during power operation to [redacted]% during shutdown. Further it is argued that this split is most significant when considering the impact on retention in filters and ion exchangers. To demonstrate this a series of calculations are performed for each CP nuclide assuming different soluble fractions. While tis changes the results they all remain an order of magnitude lower than the DB values (assuming the Co^{60} split fractions). In a similar vein RQ-ABWR-0687 query 1 (Ref. 34) questioned the assumption that Ag^{110m} is almost entirely insoluble in the environmental conditions of UK ABWR (both power operation and shutdown). The response provides theoretical arguments, but more relevantly notes that no measurements are made in the entire OPEX data set available to Hitachi-GE ([redacted] plants) except for 2 measurements at [redacted] during start-up. Both these values are low and below the DB concentrations. I consider these to be satisfactory answers, but note that I would expect the RP to be clearer in its safety case over the uncertainties and their impact.

109. Query 4 of RQ-ABWR-0687 (Ref. 34) also questions the approach used to determine the shutdown peak value. This seems to assume that there is no relationship between the power operation concentrations and the peak observed at shutdown. This relationship is reviewed in the response, which shows that while such a relationship (albeit weak) exists for Co^{60} and Zn^{65}, the same is not true for the other CP. I consider that this is partially down to the lack of data. Nonetheless the RP goes on the provide further evidence to justify their method by comparing the power operation values from OPEX to those used in the UK ABWR method and by determining the Co^{60} and Zn^{65} concentrations using the relationship that is established earlier. These are comparable with the results from the existing methodology. I am therefore satisfied that the approach adopted is fit for purpose.

110. I examined the rationale for how the defined shutdown times of [redacted] and [redacted] days respectively for BE and DB values compared with what would be expected using UK ABWR operating parameters in RQ-ABWR-0687 query 5 (Ref. 34). The response further screens the OPEX data to consider plants which operate with a similar clean-up capacity as UK ABWR (2%). This showed that the expected shutdown durations could be achieved by these plants. However, the response did not provide the more

parametric analysis I expected, nor calculated the actual performance anticipated for UK ABWR. While this is sufficient for the response to RI-ABWR-0001, I consider that further evidence of the UK ABWR clean-up systems performance in this regard should be provided as part of the overall safety case; **[Residual Matter 8]**.

Fission Products and Actinides

111. As with CP, Hitachi-GE rationalise the number of operating phases that need to be defined for FP and ActP to power operation (bounding of start-up and hot standby) and shutdown, while stating that outage concentrations are zero. The RP acknowledges that there is often a start-up spike of FPs, but this is claimed to be much smaller than that observed during shutdown and is comparable to the eventual steady state power operation values.

112. To define the FP and ActP concentrations likely in UK ABWR directly measured OPEX data was sourced for both the power operation and shutdown phase. This included data for JABWRs, JBWRs and worldwide BWRs. However, to make use of this data a number of corresponding plant parameters are needed which were not available to the RP for the majority of the data. This restricted the screened OPEX in this case to the Japanese plants (7 JBWR which included 2 ABWRs), for operations at power.

113. The OPEX data available from this review is further split into four sub-categories; noble gases, soluble and insoluble fission products, halogens and actinides because the available data, methodology and assumptions used to define the values for UK ABWR differs for each. A summary of the BE methodologies is:

- Measured noble gases – the assumption used by the RP is to use the [redacted] of plant data for periods with no fuel failures, after correcting the OPEX to UK ABWR conditions (to account for power and fuel area). All noble gases are assumed to be 100% transferred to the steam.
- Measured FP (excluding noble gases) – the same assumptions and processing of OPEX as above is performed, however additional steps are necessary to also account for the decay and removal or these nuclides from the reactor water. The BE value is defined as the average of each corrected plant concentration. It is assumed that [redacted]% of halogens and [redacted]% of soluble or insoluble nuclides are transferred to the steam.
- Unmeasured FP – the approach is to calculate the concentration based upon a relationship to the measured nuclides. This is based upon the fact that the concentration is dependent upon many factors, such as release rate, fission yield and decay constant. This is undertaken on a sub-group basis (i.e. for noble gases, soluble FP, insoluble FP and halogens). The respective carryover fractions to the steam as above are utilised.
- Actinides – the first step is to determine the reactor water concentration of Np^{239} from relevant OPEX. As this is the only measured ActP this is used to scale the others, via a "fingerprint" determined from an ORIGEN calculation. It is assumed that [redacted]% is transferred to the steam.

114. As with CPs, I had a number of detailed queries regarding the approach to definition for FP and ActP.

115. I queried the assumption that the operating chemistry does not have a significant impact on the concentration of FP in particular for the volatile nuclides (RQ-ABWR-0690 query 1 (Ref. 34)). The response provides plant OPEX that demonstrates that there is no discernible relationship between operating chemistry and the concentrations observed in the reactor water. In effect any change is within the normal cycle to cycle variation observed. I am therefore content that founding the definition on NWC plants is a reasonable simplification to make.

116. RQ-ABWR-0685 query 1 (Ref. 34) is discussed in Para. 107. This also referred to FP, and my considerations earlier apply equally here. In addition query 2 of this RQ asked a similar question regarding the assumed steam carry-over factor applied for halogens. This appeared not to consider transfer in moisture (i.e. should the DB value be [redacted]% ([redacted]% from volatility plus [redacted]% from moisture carryover)). The response notes that there is no available data on which to distinguish the precise contribution of each mechanism. The rationale for using a single value comes from the analysis of plant data; in fact two values are calculated from this data set for the BE ([redacted]; [redacted]%) and DB values ([redacted]; [redacted]%) but when rounded they become the same . On this basis I am satisfied with the use of a single value.

117. Even when fuel failure events are removed from the OPEX considered the concentration of FP and ActP is not zero. This is to some degree due to tramp uranium. However the RP does not know how much the contribution of this source is. RQ-ABWR-0689 query 3 (Ref. 34) requested further information on the impact of this source as I wished to understand the sensitivity of the defined UK ABWR source terms to increases in tramp uranium. The response provides information on the potential impact from either fuel fabrication contamination or zirconium impurities but chooses to ignore fuel damage. I subsequently followed this up in RQ-ABWR-0796 query 2 (Ref. 34), the response to which completes the analysis to include consideration of uranium released from fuel damage. Hitachi-GE considered the impact of increases in all three potential tramp uranium sources on FP (I^{131}) and ActP (Np^{239}). While the analysis is somewhat simplified and makes many assumptions (some of which I would not agree with), it does demonstrate the trends and impact possible from each potential source. Fuel rod contamination leads to an immediate increase but is rapidly removed, whereas the impact of zirconium impurities and fuel contamination is long lived. This matches plant OPEX. Assuming quite conservative values for these sources leads to calculated values that are similar to the defined BE values for UK ABWR. The most likely source of any increase would be fuel damage in previous cycles. The analysis shows that large quantities of tramp uranium (grams) would be needed to generate the DB values for either I^{131} or Np^{239}. This is further supported by the independent analysis performed by my TSC in Ref. 12, which examined changes in tramp uranium contamination from various sources. I am therefore satisfied that this analysis responds adequately to the concern. Importantly it demonstrates that the methodology used by the RP is adequate to account for the likely variations that could be reasonably expected in tramp uranium contamination (i.e. excluding large scale fuel damage).

118. For ActP the important assumption is that Np^{239} is a suitable surrogate for all other actinides. I question this in RQ-ABWR-0681 query 1 (Ref. 34). The RP's response provided theoretical arguments why this was the case; mainly due to nuclear properties and similar chemical behaviour amongst the actinides. I further queried this in RQ-ABWR-0796 query 1 (Ref. 34) as there is evidence that measurements of Np^{239} under HWC conditions may be considered unreliable (see Refs 36 and 40). The response confirms that as the UK ABWR ActP methodology is based on JBWR data this is for NWC plants. The response also provides further OPEX from plants which operate under both NWC and HWC for comparison including other actinides. Analysis of this data showed no significant variation in ActP concentrations between the different chemistry regimes. While there is an appreciable degree of scatter in data, within one order of magnitude, Hitachi-GE argue that the methodology used is fit for purpose given the concentrations and nuclear safety significance of the ActP. I accept these arguments as reasonable.

119. I queried how Hitachi-GE deal with FP generated by activation of other radionuclides in RQ-ABWR-0690 query 5 (Ref. 34). The approach adopted in Ref. 24 is to use the "f-value" Operating Rule (OR) to scale the BE to DB concentrations. The f-value is the sum of the release rate of six noble gases measured at the off-gas system, so is not

necessarily directly transferable to other nuclides. Their response demonstrates that the approach adopted yields approximately the same answer as using the concentration of the parent nuclide. I am satisfied that this resolves the query.

120. For the DB values for FP and ActP the approach is to assume that the plant is at the f-value Operating Rule (OR) level throughout the cycle. In effect this is a limit on the release rate from any defective fuel, as the plant will not be allowed to operate above this level. The RP acknowledges that this does not in itself limit the number or type of defective fuel rods, but provide evidence to suggest that this OR encompasses at least [redacted] failed rod (see Figure 5 below). The methodology employed in Ref. 24 is essentially to scale the BE values using the ratio of the f-value under BE and DB (i.e. OR) conditions. For the DB values steam carry-overs of [redacted] [redacted] and [redacted]% are assumed for noble gases, halogens and soluble/insoluble FP respectively.

121. The f-value is therefore of particular importance in defining the UK ABWR source terms. Figure 5 below shows the historic trend in f-value measured in JBWRs over the period of 1975 – 2010. This data includes both JBWR (black data) and JABWRs (green data). As is evident the general trend is for improvements (lower f-values). Typical values (no fuel failures) are around 1E6 (Bq s^{-1}). The highest peak value recorded in recent times is of [redacted] during a fuel failure at [redacted]. The various horizontal lines represent different limits; black representing the limits in Japan and the two red lines represent the different UK ABWR limits described further below (at [redacted] and [redacted]).

[redacted]

Figure 5: f-value Behaviour in JBWRs

122. The important assumption in this approach of using the f-value OR to scale is that it is possible to simply proportionately increase the concentration of all FP nuclides based upon the increase in f-value. In effect, this assumes that there is no change in the release characteristics due to fuel failures. I queried this in RQ-ABWR-0689 query 2 (Ref. 34). The information provided was insufficient to answer the query and I repeated the question in RQ-ABWR-0798 query 2. The response provides OPEX for [redacted], [redacted], [redacted] and [redacted] for periods of known fuel failure events and specific numerical comparison for the [redacted] event. The arguments put forward by Hitachi-GE are that the

trend predicted is observed in plant data, namely that there is an increase. It is further argued that the increase predicted for UK ABWR (i.e. the difference between BE and DB) is conservative, although I note that this is based on an example where the f-value increase is much smaller than would be allowed in the safety case for UK ABWR (i.e. < OR). What this response did not consider is how the specifics of any fuel damage may affect the release rates for individual nuclides. A closer analysis of the response to RQ-ABWR-0798 shows that the ratios between the various noble gas nuclides that together comprise the f-value changes when a fuel failure event occurs. This is not surprising on a scientific basis, as the release of fission products trapped in the fuel rod will lead to a proportionate increase in the longer lived nuclides within the coolant (see Ref. 40, for example). In fact, characterisation of the ratios between various nuclides is an important tool in understanding the nature, severity, location and origin of any fuel degradation and therefore should change. This means that simply assuming the same ratio before and after is incorrect and importantly means that the data is not suitable for use in radiological consequence assessment of accidents, where the longer lived nuclides (especially noble gases) tend to dominate.

123. RQ-ABWR-0870 (Ref. 34) was therefore raised to resolve this matter. This consisted of three queries which asked for additional OPEX data, a comparison of this against the methodology given in Ref. 24 and 29, and queried what will be done to resolve any identified discrepancies. The response is comprehensive and provides additional OPEX data and analysis, including JABWR and US plants. It concludes that:

- The methodology in Ref. 24 and 29 is inadequate;
- A revised methodology is proposed, based on US OPEX; and
- This new method will be reflected in updates to the RI-ABWR-001 responses.

124. The response notes that there are differences between the different data sets. In effect this is due to the severity of fuel damage observed, which ranges from small pin-hole type defects to secondary failures and fuel pin wash out. For these reasons the methodology in Ref. 24 is adapted to use the more conservative data, which increases the proportion of longer lived nuclides significantly. In principle the revised method proposed is reasonable and resolves a number of concerns. Also, it is proposed to increase the f-value from [redacted] to [redacted] for the DB values (i.e. from the lower to upper horizontal line in Figure 5). This resolves the concerns regarding sufficient operating margin (Para. 160) and iodine spike methodology (Para. 127) discussed elsewhere. The net effect of all these changes to the DB FP methodology is to increase the concentration by at least an order of magnitude and increase the proportion of longer lived nuclides. I am satisfied that this method is now fit for purpose.

125. Query 4 of RQ-ABWR-0689, and later query 3 of RQ-ABWR-0798 (Ref. 34) raised the question on the assumption that the fuel failure would be a pin-hole type leak and would be stopped by initiating power suppression. The response to RQ-ABWR-0870 above removes the first of these points, as it is no longer based solely on a pin-hole type leak. It also provides evidence the answer the first part of query 3 of RQ-ABWR-0798 regarding the impact of different fuel failure modes on the source terms. While I am satisfied that the RP has demonstrated that power suppression is suitable to arrest a fuel failure in the short term I would expect further evidence to be provided to show that it stops fuel failures from escalating, that suitable methods are available to detect further degradation before it become significant and that such operations reduce risks SFAIRP (including when it may be applied). The RP notes that these are considered in Ref. 41. Assessment of this was outside the scope of RI-ABWR-0001 but does need to be considered; **[Residual Matter 9]**.

126. I questioned the use of f-value scaling as an appropriate method to determine the DB ActP concentrations in RQ-ABWR-0681 query 3 (Ref. 34). The RP agreed that the methodology is not representative of the likely mechanism for ActP release; however it is argued that it is conservative. This is dependent on the use of power suppression

described above (to stop secondary failures) and has been analysed in the sensitivity studies conducted for RQ-ABWR-0796 query 2 (para. 117). When considered alongside these responses I consider it to be unreasonable to expect an improved method given the importance of this deficiency.

127. Due to the significant increase in FP radioactivity seen during shutdowns the RP utilises a separate method to calculate the shutdown values for UK ABWR. I^{131} is used as a reference radionuclide in order to determine the specific amount of activity relating to all other radionuclides released during the shutdown phase. The rationale for this is that a large inventory of OPEX exists for this nuclide. Empirical studies highlight a relationship between the f-value and the total activity arising from a spike in activity of I^{131}. A factor of [redacted] (Bq/(Bq/s)) is determined for other radionuclides based upon the [redacted] value from OPEX, predominantly made up of JBWRs. The BE and DB values are calculated using this same approach, but using the relevant BE or DB f-value, and by accounting for other factors such as reactor water clean-up. The relevant steam carry-over fractions from the corresponding power operation values are used.

128. This method is clearly very reliant on the validity of the empirical relationship that has been established for I^{131}. I asked for further evidence to support this in RQ-ABWR-0689 query 1 (Ref. 34) and followed this up further in RQ-ABWR-0798 query 1 (Ref. 34). The latter RQ response provides both the rationale behind the assumption plus additional plant data from JBWRs and JABWRs. This data supports the assumed factor of [redacted] and importantly closes a gap in the original analysis by extending it to consider more recent experiences. While a number of individual data points show that the factor may be larger these tend to be for larger initial f-values (and hence indicative of a greater degree of fuel damage). When considering the f-value OR declared for UK ABWR the factor of [redacted] is conservative in most instances. This additional information significantly strengthens the safety case in this regard. This also resolves query 4 in RQ-ABWR-0690 (Ref. 34) (which asked why the BE and DB values both use the same factor) as this relationship is valid over a large range of initial I^{131} concentrations. In combination with the additional margin added in response to RQ-ABWR-0870 (Ref. 34), Para. 122, I am content that sufficient has been provided to justify this approach is adequate. However I do note that the analysis does not consider plants outside of Japan or which operate with the UK ABWR operating chemistry of HWC, noble metals and zinc addition. I consider it would be beneficial to resolve this gap; **[Residual Matter 10]**.

129. I queried some facets of the applicability of this relationship for I^{131} to other nuclides in a number of RQs:

- RQ-ABWR-0690 query 3, followed up by RQ-ABWR-0823 query 1 (Ref. 34) asked about the assumption made regarding the behaviour of noble gases, which are assumed to be released twice as fast as iodine. This is based on Graham's law for effusion of gases, which inherently assumes that iodine escapes as a gas. Additional evidence is provided that for this behaviour. I am not convinced by the arguments and evidence presented here.
- The responses to RQ-ABWR-0689 query 5 and RQ-ABWR-0798 query 4 (Ref. 34) considered the use of I^{131} for determining the shutdown spike. The arguments made are mainly theoretical as, at the time of writing, the RP does not have access to relevant OPEX to substantiate iodine behaviour during shutdowns, particularly for the UK ABWR chemistry conditions. I consider this to be a weakness in the case made.

However, when balanced against the other changes made to the methodologies (in particular the response to RQ-ABWR-0870) I do not consider these gaps to be significant enough to hinder resolution of the RI. The RP commits to consider the response to RQ-ABWR-0798 query 4 further; **[Residual Matter 11]**.

130. While RQ-ABWR-0692 query 2 (Ref. 34) was raised regarding the resultant FP values from this shutdown methodology, the response did clarify to which nuclides this spike methodology applies, namely the [redacted] [redacted] and [redacted] I am content with the rationale provided for this approach. This was clarified in later revisions of Refs 24 and 29.

131. I noted that the methodology for shutdown FP spike does not appear to consider the timing of this event (cf. the CP spike assumes [redacted] or [redacted] days for BE and DB clean-up respectively) so I queried this (RQ-ABWR-0690 query 2 (Ref. 34). The RP confirmed this was the case but stated this was because the concertation is assumed to stay at the peak value for the whole [redacted] hours of shutdown. This is clearly very conservative as the [Reactor] Clean-Up Water system (CUW) will continue to remove radioactivity over this time. Evidence is provided from plant OPEX on the likely timings of such FP spikes. There is great variation, but less than [redacted] hours is typical along with two spikes coinciding to the reduction in power and pressure respectively. Arguments are made as to why the flow rate of the UK ABWR CUW will further reduce this time. I am content with these arguments given the assumption made, but do not understand why a different approach is taken for CP to FP. Residual Matter 7, Para. 107, is also relevant here.

Activation Products

132. The APs are only produced by neutron activation, hence are only produced during operations at power. For this reason Hitachi-GE only define a power operation concentration. This is considered bounding for all other phases. With the exception of H^3 the RP does not define an outage value, which is effectively set to zero.

133. The approach taken to define the concentration of AP in UK ABWR differs from nuclide to nuclide in terms of plant data, inputs and methodology although all AP concentrations are determined by statistical analysis of OPEX. This is summarised in Table 4 below (noting that C^{15} and Na^{24} were added as a result of my RQs, hence are not included in Ref. 24 or 29).

Nuclide	Data Source	Number of Plants	Data Type	BE Method	DB Method
H^3	[redacted]	[redacted]	[H^3] in RW	[redacted]	[redacted]
N^{13}	[redacted]	[redacted]	[N^{13}] in RW	[redacted]	[redacted]
C^{14}	[redacted]	[redacted]	C^{14} annual discharge [C^{14}] in RW	[redacted]	[redacted]
C^{15}	[redacted]	[redacted]	Steam dose rate RW dose rate	[redacted]	[redacted]
N^{16}	[redacted]	[redacted]	Steam dose rate [N^{16}] in steam RW dose rate	[redacted]	[redacted]
F^{18}	[redacted]	[redacted]	[F^{18}] in RW	[redacted]	[redacted]
Na^{24}	[redacted]	[redacted]	[Na^+] in RW	[redacted]	[redacted]
Cl^{36}	[redacted]	[redacted]	[Cl^-] in RW	[redacted]	[redacted]
Ar^{41}	[redacted]	[redacted]	Ar^{41} discharge	[redacted]	[redacted]

Table 4: Overview of OPEX used to Derive Activation Products for UK ABWR

134. As can be seen from Table 4, there are four main sources of OPEX used by the RP for the AP derivation; nuclide measurements [redacted]), discharge data [redacted], dose rates [redacted] or inactive precursor concentrations [redacted]. Inevitably where the OPEX used in this approach is not a direct measure of the concentration of the desired nuclide some additional steps are necessary, such as ORIGEN calculations. These are similar to other parts of the PST methodologies so are not detailed here. In principle I consider the statistical analysis of OPEX to be reasonable. However, there are clearly differences between the application of this to the individual APs and I therefore asked a number of RQs mainly related to understanding a number of assumptions and clarifying the methodologies.

135. I specifically questioned the use of different statistical methodologies in RQ-ABWR-0683 query 1 (Ref. 34). The RP argues that the preferred approach of [redacted] for BE and [redacted] for DB is used, except where the underlying OPEX does not support it. I further queried the approach for Ar^{41} and F^{18} in queries 4 and 6 of the same RQ. While I am not convinced by these arguments in some instances, I have confirmed that any changes would not have a significant effect on the eventual defined concentration (as there is little scatter in the underlying OPEX).

136. N^{16} is the most safety significant AP. It dominates doses during operations. The definition of this nuclides is based on 7 plants for the steam concentration and only 1 plant for the reactor water. I questioned if additional substantiation could be provided for this nuclide in RQ-ABWR-0682 query 4 (Ref. 34). This is particularly important for UK ABWR given the operating chemistry and its impact on steam transfer of nitrogen species. The response provides data from one additional plant. The RP commits to provide additional data when it is available. While I am content that it is unlikely that N^{16} will change significantly, it is important that suitable and sufficient evidence is provided; **[Residual Matter 12]**.

137. RQ-ABWR-0682 query 2 (Ref. 34) asked why AP could not be calculated, instead of relying on potentially unreliable OPEX. The response argues why preference is given to OPEX, which is reasonable, but does not state why it is not reasonable to calculate those which could be determined via this route. I further questioned a number of AP in RQ-ABWR-0793 query 1 (Ref. 34). This argues that the large uncertainty in any calculation renders this route unreliable. While this uncertainty may be large, I would not judge this to be any different than the reliance on the OPEX used in some cases. For example, I probed the use of (independent) discharge data to both define and justify some AP. The response (to RQ-ABWR-0682 query 3 and RQ-ABWR-0793 query 2 (Ref. 34)) restates the approach but does not provide a rationale. Calculations would provide an additional degree of reassurance in such instances. Nevertheless I do not consider this to be fundamental to resolution of the RI, as I consider it very unlikely that changes to the values would result. I am content to accept the approach as fit for purpose for definition.

138. The concentration of H^3 calculated for UK ABWR is dependent on the water management arrangements that the plant operator puts in place. In effect it would be possible to operate within a wide range of H^3 concentrations depending on how much water is discharged. In response to my question in RQ-ABWR-0683 query 2 (Ref. 34) the RP developed a Tritium Management (TM) model which simulates how tritium moves around the plant. This is based on assumptions on factors such as generation rates and flows between different systems. The TM model provides concentrations for the primary coolant, SFP and suppression pool. Three cases are calculated, a base case with realistic assumptions and two sensitivity cases. The results of the base case compare favourably with both the UK ABWR BE value and OPEX from Japanese BWRs and demonstrates that an equilibrium concentration is reached after around 10

years of operation. When liquid discharges are set to zero the resultant equilibrium concentrations are around [redacted]% higher but lower than the DB PST values. As liquid discharges are the factor that the operator has most control over, this represents a bounding case. I am satisfied that this demonstrates that the derived PST values for UK ABWR are based on a reasonable set of assumptions over water management.

139. A number of the AP are volatile and therefore the assumptions used about their behaviour are potentially important. In particular:

- Under HWC conditions the amount of N^{16} transferred to the steam phase is increased by a factor of 1.3. The equivalent factor for N^{13} is 10. I asked for a justification and evidence for this different behaviour between nitrogen nuclides in RQ-ABWR-0683 query 5 (Ref. 34). The response explains that this is a function of the differing half-lives and transport around the primary circuit and cited OPEX from Japanese and US BWRs as evidence.
- Ref. 24 states that "*…a steam carryover factor of* [redacted] *% is assumed*" for F^{18}. This was not evident in the defined values (Ref. 31) where the reactor water concentration is not zero, but equal to the steam phase. The response to RQ-ABWR-0685 query 3 (Ref. 34) clarifies the scientific rationale for this behaviour (it is analogous to the equilibrium behaviour exhibited by H^3). The response to RQ-ABWR-0683 query 6 is also relevant. In effect because there is uncertainty in the mechanism behind this and no OPEX for steam systems is available the RP assumes that the water and steam phases are equal. I am not convinced this is reasonable, but it is clearly conservative.

I accept these simplifications as appropriate.

Cycle Average

140. As suggested the Cycle Average (CA) values represent a single value to account for all normal operating phases. It is in effect the average across a full fuel cycle. This is derived for uses in the safety case where longer term consideration of the radioactivity is more important, for example as part of the DST methodology. In its simplest form it is the average of the integral of the activity for each operating phase (i.e. start-up, pwer operation, shutdown and outage). For CA values to be representative with respect to plant operation, additional terms are also included to account for unplanned shutdown and restarts (for CP, FP and ActP) and for fuel failure (for FP and ActP) as an expected event. BE and DB CA values are derived by changing the relevant input conditions. These are shown below (Figure 6) for each nuclide category:

[redacted]

[redacted]

Figure 6: Schematic Cycle Average Profiles

KEY: From top to bottom - CP; Halogen, noble gas and volatile FP; ActP and non-volatile FP; AP. (1) start-up, (2) power operation, (3) shutdown, (4) outage, (5) / (6) unplanned shutdown and restart or fuel failure.

141. Note that these represent the final CA methodologies employed, which have evolved from Ref. 24. A number of these changes were in response to RQs. For example RQ-ABWR-0688 query 2 (Ref. 34) asked about the original assumption used of an instantaneous decrease in concentration during an outage and the approach was later amended to better reflect reality.

142. One of the important assumptions in the CA definition is the frequency of unplanned events such as reactor trips and fuel failures. The assumption for trips is once every [redacted] cycles for BE and [redacted] or DB; whereas the fuel failure frequency is given as one pin [redacted] (both BE and DB). RQ-ABWR-0688 queries 1 and 3 (Ref. 34) requested a justification for these. The response provided relent OPEX from JBWRs. I further queried the relevance of this based on worldwide BWR experience in RQ-ABWR-0795 query 1 (Ref. 34). This response extended the justification to include the entire BWR fleet. The data supplied broadly supports the assumptions used by Hitachi-GE (typically [redacted] unplanned trips and slightly over [redacted] fuel pin failure per cycle). I remain unclear why the fuel failure rate assumption is the same for BE and DB. However, while each additional failure will increase the BE CA value by over 40% (for example for I^{131}), this still remains an order of magnitude lower than the DB value. Similarly any increase in the unplanned shutdown frequency would impact the BE CA but have very limited impact on DB concentrations.

143. I queried how the CA values for ActP were calculated in RQ-ABWR-0693 query 3 (Ref. 34). The response provides an explanation for this, which also indicates that the BE CA value includes a [redacted] day period at the DB level due to a fuel failure. As the RP contends that there is no increase in ActP activity during fuel failures or transients I queried this further in RQ-ABWR-0824 (Ref. 34). I remain unconvinced about this approach as there does not seem to be a rationale to support it. Given the safety significance of ActP I am not inclined to push for any improvements, as the net effect is to marginally increase the CA compared to the BE power operation values.

144. One oddity in some CA values is that they can sometimes be smaller than the corresponding power operation values. I queried this in RQ-ABWR-0692 query 3 and RQ-ABWR-0799 query 1 (Ref. 34). The responses further clarified why this was the case, which appears to be a function of the methodology employed. However, while this appears technically incorrect, the scale of difference is relatively small and particularly affects the BE values. I therefore consider little benefit would be gained by changing this.

Justification

145. Hitachi-GE provides dedicated justification for the PST which varies according to the nuclide category. This is in addition to the justification that is an inherent part of using relevant OPEX for definition.

Corrosion Products

146. The RP recognise that as Co^{60} is the most safety significant radionuclide in the CP inventory, and is used as a scaling factor to calculate other unmeasured CP radionuclides, greater efforts are made to justify its concentration. This is done using OPEX associated with the JABWR, [redacted], to derive an independent estimate. Comparisons are also made to the dataset used for definition as a whole, which provides additional reassurance. While this is the closest plant to UK ABWR in terms of design, it does operate with a different water chemistry regime (Normal Water Chemistry (NWC)). As previously Hitachi-GE argues that the dominant effect is feedwater iron concentrations, which for [redacted] are in-line with that specified for UK ABWR. A similar comparison with respect to other measured radionuclides is also provided.

147. To justify the unmeasured radionuclides (albeit indirectly as OPEX is not available) the RP derives values for the other measured (Co^{58}, Fe^{59}, Mn^{54} and Cr^{51}) using the Co^{60} scaling factor, and compares these values to the corresponding OPEX data. These values do agree, within a degree of accuracy that would be expected for such a method.

148. I noted that Hitachi-GE provide no justification for the values at shutdown in Ref. 24. Given the importance of this phase of operations I requested this in RQ-ABWR-0686 (Ref. 34). In their response the RP provides justification using BWR data and by further analysing the data set used in the derivation but for OPEX that was screened out due to too high feedwater Fe. I welcome these additions as providing further confidence. However the response specifically only considers Co^{60} because of its safety significance. The RQ did not specify this nuclide only and because other CP may behave differently, I followed this up further in RQ-ABWR-0794 (Ref. 34). In this RQ response the RP extends the analysis of existing OPEX to include other measured CP nuclides and provides additional data for [redacted] for 10 shutdowns. In analysing this new data Hitachi-GE notes a discrepancy when compared to worldwide OPEX for Co^{58}, with the JABWR values notably higher. While the reason for this is unknown, but is suspected to be a function of the ABWR plant feedwater heater design and materials, the RP decides to increase the UK ABWR values to account for this potential effect. I accept this as a reasonable decision.

Fission Products and Actinides

149. The PST derived for the different FP groups is justified by deriving a separate set of values based on independent and separate OPEX, which is taken from worldwide BWRs. The number of plants used in this justification differs for each group of FP but varies between [redacted] and [redacted]. In summary:

- Noble gases - BE concentrations are justified by comparing with graphical data and worldwide plants (numeric data). DB values are compared with JABWR data and theoretical release rate calculations. The main noble gas nuclides are included.
- Halogens – the same approach as noble gases is applied. The comparison is undertaken for the Iodine nuclides.
- Soluble FP - the same approach as noble gases is applied, except that the BE data is limited to a small number of worldwide plants. The graphical data considers Sr only but the latter includes a wider range of nuclides.
- Insoluble FP – the same approach and data as soluble FP is used. The comparisons are limited to Sr^{92} and Cs^{137} only.

150. Ref. 24 and 29 do not provide any explicit justification for the concentrations of ActP derived. This was requested in RQ-ABWR-0681 query 4 (Ref. 34). In response the RP provides additional OPEX for Np^{239} in 11 US BWRs and Pu^{241} from a single US plant. I also note the Swedish plant data used in the latter response to RQ-ABWR-0796 query 1 (Ref. 34). This data is included in the later updates to Ref. 24 and 28 as the justification.

151. The FP and ActP justification is therefore somewhat limited, but does concentrate on those nuclides of most importance. There is also a further degree of justification provided in the responses to many of my RQs, which provide additional data which can be compared against the UK ABWR values. Holistically, I judge the degree of justification provided to be sufficient (and also note that it would prove difficult for the RP to provide more).

Activation Products

152. As with the definition of APs the approach taken by Hitachi-GE to justify the values differs by nuclide, as shown in Table 5 below:

Nuclide	Data Source	Number of Plants	Data Type
H^3	[redacted]	[redacted]	$[H^3]$ in RW

	[redacted]		
N^{13}	[redacted]	[redacted]	$[N^{13}]$ in RW
C^{14}	[redacted]	Worldwide -	C^{14} annual discharge $[O^{17}]$ in RW
C^{15}	[redacted]	[redacted]	Steam dose rate
N^{16}	[redacted]	-	-
F^{18}	[redacted]	-	$[F^{18}]$ in RW
Na^{24}	-	-	-
Cl^{36}	[redacted]	[redacted]	$[Cl^{36}]$ in RW
Ar^{41}	[redacted]	[redacted]	Ar^{41} annual discharge

Table 5: Overview of OPEX used to Justify Activation Products for UK ABWR

153. Where OPEX is used the same statistical analysis is undertaken to derive separate BE and DB values to compare with the defined values. As identified earlier the amount of OPEX available to Hitachi-GE to define the AP was relatively limited, therefore this is further reflected in the restricted scope for justification that is provided. Given these restrictions I am content with the approach taken by the RP.

154. I note that the most safety significant AP, N^{16}, is justified via a calculation route. This is based on a fairly simple methodology, used extensively and should be reliable given the significance of this nuclide. RQ-ABWR-0682 query 2 (Ref. 34), as discussed in Para. 137, discusses the suitability of calculations for AP more generally. While I am content that their use for definition may be restricted, I consider that further comfort could be gained if they were to be applied for justification.

<u>Cycle Average</u>

155. There is no specific justification provided for the CA values. Given that these are a composite measure, and therefore take credit for the justifications given for each nuclide category above, I judge this to be proportionate and acceptable.

Defined Values

156. Ref. 31 contains the tabulated values for soluble, insoluble and total radionuclide concentrations in the reactor water and steam. This is provided for all four normal operating phases and the CA.

157. In my assessment of Ref. 31 I did raise further queries when I compared the cited values with the described methodologies:

- RQ-ABWR-0692 query 1 (Ref. 34) noted that the soluble:insoluble ratio varied for the CP between water and steam phases, which would not be possible in the described methodology. The RP noted this was an error and corrected this in subsequent revisions of Ref. 31.
- RQ-ABWR-0693 query 1 (Ref. 34) asked why the CP shutdown values were quoted as a peak value for reactor water, but an average for steam. To aid consistency the RP changed both to peak values in subsequent updates.
- RQ-ABWR-0692 query 2 (Ref. 34) queried why the shutdown phase values for FP were the same as the power operation phase in some instances. The response clarified that Ref. 24 and 29 were in error, and the spike methodology is applied only to the halogens, alkali metals and noble gases. The RP

contends that the power operation phase bounds any perturbations in the other FP nuclides due to the relatively short timescales for shutdown periods compared to power operation. Further information was provided in response to RQ-ABWR-0799 query 1 which led to revisions to the methodologies.

- RQ-ABWR-0693 query 2 (Ref. 34) asked why the outage values are often defined as zero. The response acknowledges that this simplification does not represent reality but provides evidence that it is a reasonable assumption to make. To clarify this additional information has been added to updates to Ref. 31.
- RQ-ABWR-0692 query 4 (Ref. 34) asked a number of detailed points regarding the values for F^{18}, N^{16}, N^{13} and some other AP where values were quoted as "*0.0E+0*" and others as "-". The RP provided satisfactory responses to my questions making a number of updates in later revisions of Ref. 31, including for example, the removal of the insoluble form of N^{13} originally defined (see also RQ-ABWR-0799 query 2, and RQ-ABWR-0773 query 1 relevant to the PrST).

158. It is notable in the RI-ABWR-0001 responses that Hitachi-GE does not provide data for a comparable BWR to UK ABWR. While there will inevitably be differences from any plant, this would provide comfort that the approach is leading to reasonable values. I asked for this data in RQ-ABWR-0684 query 3 (Ref. 34). The response gives average reactor water concentrations for a full suite of nuclides for [redacted] measured during multiple cycles over around a 15 year period. This data has periods of known fuel failures removed, hence is comparable to the UK ABWR BE values. The numeric values are contained in the RQ response, alongside the graphical data. The response also includes a comparison between the UK ABWR and the plant datasets for the total concentration of radionuclides in reactor water during the power operation phase. This is repeated below (Figure 7) as it demonstrates that the PST defined for the UK ABWR is mostly comparable or more conservative in all cases, with the exception of Cr^{51}. The rationale for this specific difference is understood and accounted for in the UK ABWR values, as discussed in Para. 108.

[redacted]

Figure 7: Comparison between UK ABWR (BE) and [redacted] PST Values

159. The response to this RQ, plus the additional OPEX and data contained within multiple RQ responses gives confidence that the BE PST values defined for UK ABWR are reasonable. This is further confirmed by comparing the data with Refs 12, 37 and 39.

160. While it is relatively simple to compare the BE values with corresponding OPEX, the same is not true for the DB values. This is because they are based on a large number of assumptions and are therefore subject to a much greater degree of uncertainty. A number of RQ responses provide additional OPEX data that can be used to determine if similar comparisons can be undertaken for the DB values. The main of these are RQ-ABWR-0684, 798 and 870 (Ref. 34), amongst others. Additional comparisons are also possible with data provided in Refs 12, 16, 36, 37, 38 and 39. When taken collectively this data is supportive of the values determined by Hitachi-GE, within the range of uncertainty that would be expected and when the approach used by the RP is considered. However, as the DB values are used as part of the assessment of a number of significant issues within the safety case, it is more relevant to determine if they are suitable for those purposes. In this regard reviewing the DB values showed that the difference between the BE and DB values was relatively small. This difference, termed the "operating margin", is typically one or two orders of magnitude. When comparing this to previous LWR safety cases (Sizewell B, UK EPR™ and AP1000®) it appears that the operating margin in UK ABWR is smaller both in absolute terms and in the apparent sensitivity to changes that may reduce this margin. Clearly how much margin is defined is a balance that needs to be struck between meeting deterministic criteria, demonstrating risks are ALARP and the practicalities of operating a nuclear plant. I had concerns with the approach adopted by the RP in this regard, so raised RQ-ABWR-0741, plus subsequent RQ-ABWR-0906 (Ref. 34).

161. The response considers the use of the DB values, in particular with regards to shielding, radiation zoning and faults analysis. Hitachi-GE argues that, as the JABWR design is based on a source term generally higher than that defined for UK ABWR (for example the f-value OR is set at 3.7E9), the plant design (i.e. shielding) is conservative. It is noted that there are some exceptions to this, such as the Co^{60} DB values which are higher in UK ABWR than JABWR. The RP commits to review these differences as part of the on-going development of the safety case. While this aspect is not within the scope of this assessment, it is important that these differences are resolved. I therefore consider that this should be considered further; **[Residual Matter 13]**.

162. For the faults studies aspect the RP agrees that less operating margin is specified for UK ABWR, with a OR f-value of [redacted] Evidence is provided in response to this RQ, and elsewhere, that JABWR plants are able to operate within this limit (See Figure 5). While it is commendable that the RP expects the eventual licensee to operate the UK ABWR to stringent limits, I remained concerned that this was potentially impacting on the plant design. For example the justification for venting would be strongly influenced by the amount of radioactivity released. However, in response to RQ-ABWR-0870 (Ref. 34) the RP commits to change the DB f-value from [redacted] to [redacted] for the RI-ABWR-0001 responses (however the OR is still expected to be maintained at the lower level). This order of magnitude change directly increases the operating margin. I am satisfied that this removes my concerns.

Summary for PST

163. While my assessment, and the RPs responses, considers definition and justification separately (as indicated in the RI) my judgement is based on the overall adequacy of what Hitachi-GE have provided in order to meet the intent behind the RI. Namely, to provide a suitably robust set of source term information that is appropriate for use in the UK ABWR safety case. On this basis I judge that:

- The RP has provided a suitable and sufficient definition for the PST (based on its definition of PST);
- An adequate and proportionate degree of supporting evidence has been provided, which is focussed on those nuclides of highest safety significance;
- I am content that the BE values derived represent a reasonable estimate, for safety case purposes, of the likely performance of UK ABWR;
- I am content that the RP has derived a set of conservative DB values which should be suitable for use in the safety case (noting that it was outside the scope of my assessment to assess the application to individual end user topics)
- For those matters where I am less content with the methodologies, assumptions or approach I am satisfied that any changes would not have a significant impact on the derived values;
- While I have identified a number of Residual Matters for follow up, I do not consider any of these to be material to resolving the RI.

164. I have reviewed the latest revisions of Refs 24, 28 and 31 and am content that these reflect the response to most of the RQs raised as part of my assessment, plus any additional changes identified by Hitachi-GE during development of the RI-ABWR-0001 response. Importantly this includes the significant amendments made to the DB FP methodology in response to RQ-ABWR-0870. I am content that sufficient has been documented to capture and understand the basis of the UK ABWR source terms should this need to be revisited in the future.

165. However, when reviewing the PST suite of documents together, I do not think they are as clear as I would expect over the uncertainties in and reliabilities of the various defined values. This is implicit within the detailed methodologies, but it would be useful to describe this upfront. For example, I have limited confidence in the defined soluble/insoluble splits for many nuclides. I judge this should be improved; **[Residual Matter 14]**.

4.2.5 Process Source Terms

166. The Process Source Term (PrST) is the next stage in the source terms methodology developed by Hitachi-GE in responding to RI-ABWR-0001. The PrST determines the concentration of the various nuclides in the plant systems for UK ABWR, taking the PST developed earlier as the starting point. The approach is essentially one based on deriving a mass balance to account for how radioactivity moves around the various water and steam systems.

167. The PrST documentation consists of a methodology report (Ref. 25), supporting report (Ref. 29) and values (Ref. 32). It is relevant to note that these revisions (Rev. 0) are in some instances unclear and poorly written and many of my RQs focussed on clarifying the content of these.

168. My assessment of the PrST is described below.

Methodology

169. The same reporting structure as for the PST is retained for the PrST, namely the methodology is described in Ref. 25, and further detailed in the supporting report, Ref. 29. Similarly the same approach to definition is used, with separate BE, DB and CA values derived. The difference in these comes about mainly because of the use of the different PST values as inputs (e.g. BE or DB), although there are some additional factors within the PrST calculations that differ between BE and DB. Unlike the PST, the PrST in Ref. 29 only defines values for the power operation and CA. The RP argues that this is driven by the requirements of the end users (i.e. it is not necessary to define the shutdown values for instance).

170. I specifically queried this point in RQ-ABWR-0771 query 3 (Ref. 34). In response the RP identifies that it is, in some instances, necessary to define other PrST values. RQ-ABWR-0773 query 3 provides further details and works through the end user requirements to identify what these are. The latest revision of Ref. 32 now contains values for other operation phases such as shutdown and outage for a number of systems.

171. I also questioned how this approach worked for systems which are operated only on an intermittent basis in RQ-ABWR-0771 query 2 (Ref. 34), in particular in relation to the Residual Heat Removal (RHR) system. Hitachi-GE clarified that "power operation" in this instance actually refers to using the shutdown PST values as an input. This is reasonable, but is potentially an error trap within the documentation. The RP provided clarity on this (and others) in later updates to Refs 25, 29 and 32. I asked for further information on the use of CA values for intermittent systems more generally in query 4 of the same RQ. The RP details where some of these are used, but in effect they are defined because they are needed to derive the DST values for those systems.

172. While the PST is determined using OPEX data, the approach for the water and steam systems downstream of the reactor differs. The RP argues that this is because a wider range of processes and plant-specific operational practices are involved, and more relevantly insufficient OPEX is available to allow a comprehensive and meaningful assessment for most systems or that allows a PrST to be established at all points required. On the other hand, these systems represent process plant with well-defined design parameters so are amenable to process modelling. Hitachi-GE note that such modelling has been routinely applied to assessment of source terms for JABWR's (but do not cite any further details or results). A small number of systems do not use this approach and the source term is taken directly from OPEX due to the large uncertainty on any calculated values for that system. In effect the RP is choosing the method which it feels to be the most reliable (modelling or OPEX) for a given system. I am satisfied that this is a sensible tactic for the PrST.

173. I questioned the use of OPEX for the SFP, suppression pool and CST in query 5 of RQ-ABWR-0771 (Ref. 34). In response the RP provides a rational justification for the retention of the OPEX based approach for the suppression pool, which is based on many measurements from [redacted]. While the OPEX used for the CST is retained the processing methodology is revised to account for the operations within UK ABWR (in effect to account for the higher proportion of Low Conductivity Waste [water] system (LCW) input expected). Hitachi-GE recognises that the SFP OPEX is of limited values and therefore reverts to a calculation based approach, similar to other PrST determinations. I welcome these changes, especially for the SFP.

174. On a related matter RQ-ABWR-0771 query 6 (Ref. 34) notes that the PrST methodology report states that "*sludge or crud at the bottom of the spent fuel pool is not in scope of the PrST but covered in the scope of the deposit source term*", which is not correct, and asks if and where this is considered. In response the RP commits to develop a method for inclusion in future updates to the report. It also notes that pond sludge makes a very small overall contribution to the total SFP inventory. The latest revision of Ref. 29 contains this analysis, which while simplistic and readily challenged does support the overall contention that it is of little practical importance within RI-ABWR-0001. However, this may need reconsidering for other specific uses, for example, in radiological consequence assessment of boiling within the SFP.

175. The precise methodology for determining the respective mass balance within each system is conceptually straightforward, with the RP considering the relevant processes that change the concentration of individual nuclides as they move through the various systems. These include factors such as decay (and subsequent daughter nuclide production), deposition, removal, partitioning and accumulation. Inevitably there are

many assumptions in these processes, such as decontamination factors within ion exchangers or partitioning factors between phases.

176. To try and understand the sensitivity inherent within the PrST methodology I asked the RP to identify the operational practices which could have a significant impact on the PrST in RQ-ABWR-0771 query 1 (Ref. 34). In response Hitachi-GE provide an overview of the factors which they consider to be most important. The details of many of these are outside the scope of RI-ABWR-0001, falling within RO-ABWR-0006 (Ref. 9). However the response does highlight factors such as feedwater iron control, operation of clean-up systems (i.e. DFs, flows etc) and adequate chemistry control of importance. Operation of the Condensate Purification system (CPS) is identified as being the key, as this potentially impacts on many other connected area of plant. Arguments are provided as to why the RP considers their approach to definition to be suitable which are centred on inherent conservatisms in the OPEX or assumptions used. I am satisfied with this response in the context of this RI.

177. While the RP considers all of the nuclides defined with the PST, the nuclides are divided in a manner that reflects their expected behaviour around the plant. Thus a number of different assumptions are used for each group. It is here where most of the important assumptions that could affect the PrST values are defined. The groups are:

- [redacted]
- [redacted]
- [redacted]
- [redacted]
- [redacted] and
- [redacted]

178. The RP also chooses to further sub-divide the PrST into systems classes; water systems, steam systems, air off take and off-gas systems, liquid radwaste systems and Heating, Ventilation and Air Conditioning (HVAC) system. This split is primarily based on the different behaviours of the nuclides in these classes of systems, which reflects into differences in the assumptions used. Importantly the RP changes the assumptions based upon the specific end use; for example assumptions are made which maximise flow and partitioning when discharges are being considered. For each system a number of PrST values are defined at identified points within the system, for example before and after an ion-exchanger. The location of these points has been considered in terms of the use that might be made of that information within the safety case; for example in defining the loadings in radwaste. It is welcome that Hitachi-GE is specifically tailoring the work in resolving RI-ABWR-0001 to meet the needs of the safety case.

179. I queried a number of important assumptions in more detail. In particular as part of RQ-ABWR-0772 (Ref. 34):

- Query 1 asks for a demonstration of why Hitachi-GE state that the CUW demineraliser performance is not affected by operations under HWC, noble metals and zinc addition. In response the RP provides evidence from US BWRs from pre- and post-HWC and noble metal addition. I note that the response does not consider zinc. The response refers to the design justification topic report (Ref. 44) for further evidence. In the context of defining the PrST I am content that this response is reasonable as only a small DF is claimed in the calculations compared to the actual performance likely.
- Query 2 clarified that the particulate form of C^{14} was based on an erroneous assumption in the PST, which was subsequently removed leaving only the soluble form.
- Query 3 asked about the behaviour of nitrogen species within the PrST. Hitachi-GE assume that [redacted]% is transferred to the steam in the moisture

separator reheater and feedwater heaters, but [redacted] % is transferred in the condenser (although an additional [redacted] % is assumed to stay in the condensate). Theoretical and the limited OPEX that is available is provided to support these assumptions. In addition, the RP undertakes a sensitivity analysis to determine the effect of changing the transfer percentage to [redacted] % in the feedwater heaters (as these are the most likely source of personnel exposure). While this increases the dose associated with the drains, they remain below the expected (R0) area limits.

- Query 5 asked about the effects of assumed DFs in water treatment systems on the PrST values. In response the methodology is changed such that two DFs are defined; one for accumulation within resins (set as ∞) and one for liquid abatement (set at the defined value, which is taken as the minimum for any period of operation). The calculations can therefore be considered conservative.
- Query 6 was raised to confirm why the PrST for the off-gas system is calculated with different flow rates. Simply put, these conditions maximise the accumulation in the charcoal beds or in the process gas (similar to the treatment of DFs in water systems above).

On balance, given the uncertainties elsewhere and the sensitivity to these and other assumptions used in the PrST I am content to accept these responses.

Justification

180. In Refs 25 and 29, the results of the PrST have, where possible, been underpinned using available OPEX data from JBWRs and elsewhere. The extent of this justification varies both by system and nuclide mainly due to constraints on that availability of reliable data. In summary this justification comprises comparison with plant OPEX (Japanese) for the following:

- CUW – resin accumulation (Co^{60}, Cs^{137}, Fe^{55} and Sr^{90})
- Condensate – iodine and C^{14} concentrations in condensate water
- CPS – resin accumulation (Co^{60}, Cs^{137}, Fe^{55} and Sr^{90})
- SFP – water concentrations (Cr^{51}, Mn^{54}, Co^{58}, Co^{60}, Cs^{134} and Cs^{137})
- Main steam – N^{16} and C^{15} (from dose rates)
- Extract steam - N^{16} and C^{15} (from dose rates)
- Off-gas – SJAE outlet concentrations (iodine nuclides) and condenser outlet (noble gases and C^{14})
- LCW – resin accumulation (Co^{60}, Cs^{137}, Fe^{55} and C^{14})

181. The OPEX justification largely underpins the results of the PrST modelling for the UK ABWR. Where results are less consistent there are reasons for this, including differences in design or operational practices between the plants (that are frequently only poorly known from the published information) or uncertainties in the reported OPEX data. Thus differences can be rationalised with a reasonable degree of confidence. The plant OPEX does demonstrate that the modelling approach adopted provides a more wide ranging and extensive PrST than that which would be possible using the plan data alone.

Values

182. The derived PrST values are given within Ref. 32. The approach adopted by Hitachi-GE is to define four sets of values (BE and DB for power operation, BE and DB for CA) for all of the defined points around the various systems (140 points) plus an additional 8 datasets for shutdown and outage conditions. Each dataset provides concentrations for all 117 nuclides (some split into soluble and insoluble). This leads to over 70,000 individual values being defined in the latest revision of Ref. 32.

183. Defining BE and DB values for power operation and CA for all of these systems results in a huge amount of data. I queried if all of these were actually necessary in RQ-ABWR-0771 query 4 (Ref. 34). The response states that many of these are in fact not used by the end users, but another portion of these are used specifically in the derivation of the DST. There still remains a large portion of the defined PrST values that, at present, are not required as part of the safety case.

184. Given this large amount of data I have not assessed all of the PrST values. Instead I chose to sample the outputs for various key nuclides across a range of systems. In general I am content that the values are consistent with the approaches described previously and theoretically make sense with what might be expected. The exception to this was the zero concentration of iodine nuclides in the charcoal absorbers of the off-gas system. This was queried in RQ-ABWR-0773 query 2 (Ref. 34), because the previous response provide by the RP to RQ-ABWR-0083 argues that the beds are very effective at removing iodine and this may be important in fault analysis. The response is contradictory to other information in the safety case and argues that the approach is reasonable. I require further evidence for this, and therefore consider that this should be resolved as part of RO-ABWR-0066 (Ref. 10); **[Residual Matter 15]**.

Summary for PrST

185. Based on my assessment of the definition, justification and values for the PrST produced for UK ABWR I judge that:

- The RP has provided a definition for the PrST which appears reasonable for use within the UK ABWR safety case;
- The PrST values are perhaps the source term category where there is the highest sensitivity to input assumptions (as the PST and DST are scientifically constrained, whereas the PrST is more operationally based) but the methodologies used are conceptually simple, and do lead to trends and behaviours that would be expected;
- The quality and quantity of evidence that has been provided to justify that the derived values are reasonable is variable, but this is limited in scope and the RP has concentrated on the most safety significant aspects;
- For those matters where I am less content with the methodologies, assumptions or approach I am satisfied that any changes would not have a significant impact on the derived values;
- While I have identified a number of Residual Matters for follow up, I do not consider any of these to be material to resolving the RI.

186. I have reviewed the latest revisions of Refs 25, 29 and 32 and am content that these reflect the response to most of the RQs raised as part of my assessment, plus any additional changes identified by Hitachi-GE during development of the RI-ABWR-0001 response. These are significantly improved from the first revisions in terms of clarity and content.

4.2.6 Deposit Source Terms

187. The Deposit Source Term (DST) is the third and last part of the source terms methodology developed by Hitachi-GE in responding to RI-ABWR-0001. The DST considers the concentration of nuclides that accumulate on both the internal pipework surfaces within various systems and the fuel cladding. The DST is therefore an important element of the source term derivation, in particular Co^{60} which is the largest dose contributor during outages. The DST relies on both the PST and PrST values derived earlier as inputs.

188. The DST documentation consists of a methodology report (Ref. 26), supporting report (Ref. 30) and values (Ref. 33) and my assessment of these is described further below.

189. The first revisions of the DST documentations that were submitted were very difficult to read. They were often unclear and contained no overall narrative that linked the various aspects together, to enable an understanding of the logic that was being developed. Because of this, many of my RQs asked for further clarity on the approach and methodologies that were being used (RQ-ABWR-0765 queries 1 to 5, RQ-ABWR-0766 queries 3 and 8 and RQ-ABWR-0767 query 2 all refer (Ref. 34)). The following descriptions are consistent with these responses.

Methodology

190. The DST is derived for both the piping surfaces exposed to reactor coolant and the fuel cladding. As such the methodology (Ref. 26) is different for each of these. As with other aspects of the UK ABWR source terms the approach is to undertake a statistical analysis of relevant OPEX, where this can be justified.

191. In RQ-ABWR-0767 query 1 (Ref. 34) I asked why Hitachi-GE considered it unnecessary to derive DST values for other components where radioactivity may deposit, such as the steam dryer, reactor internals or RIPs. The response argues that it is both unnecessary, due to the contribution of deposited radioactivity compared to activation, and difficult, due to a lack of detailed OPEX, to calculate DST values for these components. The RP states that the safety case associated with these will mainly rely on measured dose rates. I accept these arguments.

<u>Piping DST</u>

192. Both BE and DB values for the concentration of radioactivity deposited on piping surfaces are calculated, based upon 60 years of operation of the plant (i.e. the values derived represent the established nuclide composition).The piping DST is defined for 66 nuclides, covered CP, FP, ActP and AP. Ref. 26 provides a rationale for which nuclides are considered.

193. The piping DST covers a range of systems across the plant. The significance of the DST varies across these, ranging from the CUW where deposited radioactivity has a major safety impact to other, lower tier systems where the DST is relatively unimportant. This is reflected in the quality and quantity of OPEX that is available to the RP. To account for this variability three modifications on the piping DST methodology are used, which try to adopt a proportionate approach to the rigour of derivation. An important part of the methodology (both definition and justification) is a Hitachi-GE derived model (HGE-DST), based upon an effective deposition rate coefficient, which quantifies deposition, spalling and dissolution behaviour. The RP claims that this has been used successfully on JBWRs under NWC conditions for many years. This model predicts radioactivity deposited on piping surfaces exposed to reactor water. Further details of the model are presented in Ref. 26, but it is a simplified version of the Hitachi-GE CP model (see also Ref. 45 for a TSC review of this model).

194. The piping DST is derived for 14 systems. The RP identifies four of these as having a significant DST burden, namely the [redacted], [redacted] [redacted] and [redacted] systems. The approach taken by the RP is logical and well structured, considering factors such as what are the most important DST nuclides, what are the important factors that affect the DST and the impact of system design and operation. Where relevant these considerations are supported by OPEX, for example from JBWRs.

195. The three variations on the methodology, and their application amongst the various UK ABWR systems are summarised as:

- For the CUW system the DST is evaluated by direct translation of OPEX (in terms of deriving dose rate and hence radionuclide deposition) and also by application of the HGE-DST deposition model (for unmeasured nuclides). In this case there is sufficient OPEX data from US plants and other sources which quantify radiation dose rates (under standardised conditions) for reactor water systems at high temperature. The application of the HGE-DST model for this system provides a comparison with the OPEX derived values, derives unmeasured nuclides and also serves to justify the use of the model to other systems with less OPEX.
- Although the LCW, RHR and FPC systems are important in terms of DST, there is little OPEX data of relevance available so derivation uses the HGE-DST model alone. OPEX from JABWRs is available for the LCW system as a comparison.
- For all the other systems (other than CUW, RHR, FPC, LCW), the radioactive concentrations will be relatively low and the accumulation of deposited radioactivity similarly slow. Therefore a simple calculation is made for these systems, based upon factoring the PrST with the ratio of radionuclide concentrations in the reactor water.

196. The details of the methodology are complex and are not repeated here (totalling 17 steps, each of which contains several sub-steps). They are described in full in Ref. 26. However, a number of salient points are:

- The UK ABWR DST for the CUW is based on OPEX data from a number of US BWRs which operated with HWC, noble metals and zinc addition. This is for many cycles of operation (> 10 years). Analysing this OPEX gives the key outputs of the average measured dose rates (BRAC data) and average Co^{60}:Zn (soluble) ratio. This latter ratio is the key determining factor in the entire methodology.
- These outputs are used, along with a number of assumptions on the UK ABWR operating chemistry and behaviour, to determine the corresponding ratio for Co^{60}:Zn (soluble) in UK ABWR, which is then correlated back to a dose rate. Using calculations this dose rate is converted into the DST value. The BE and DB values are determined by using the BE or DB PrST Co^{60} value as an input.
- The HGE-DST model makes use of “effective deposition rate coefficients”. These are based on [redacted] nuclides from JBWRs and are based on measurements for stainless steel piping under NWC conditions. The DST can be calculated by inputting the appropriate PrST BE or DB value and accounting for growth and decay in intermittently used systems.

197. I queried a number of aspects of the piping DST methodology during my assessment.

198. In RQ-ABWR-0766 (Ref. 34) query 1 I asked how the CUW unmeasured nuclide values derived by the HGE-DST model are aligned with the OPEX measured nuclides. In particular it is noted that where the two methods overlap, the model results are higher. Hitachi-GE confirmed that no standardisation process takes place (i.e. the OPEX and calculated Co^{60} values are not ratioed) and the model derived values are used for the unmeasured nuclides, with recognition that they have further conservatism.

199. While the US OPEX used to derive the DST is based upon plants which operate under the UK ABWR operating chemistry they have not always done so, starting life under NWC conditions. I queried why this factor was not important in query 2 of RQ-ABWR-0766 (Ref. 34). The response provides BRAC data for periods where plants have transitioned operating chemistry regimes which shows that there is great variability in the measured dose rate which does not align directly with chemistry changes. In addition arguments are made regarding the impact of any chemical decontamination and the removal of oxide layers meaning that subsequent DST behaviour should be

comparable to a new plant. The RP notes that some of the data used in the UK ABWR analysis includes plants which have undergone decontamination and because there are no significant differences in behaviour the effect can be neglected. I can accept this as a reasonable argument but the response provides no data to support this argument. I do not consider this to be fundamental to resolving the RI, but think this is an important claim that needs to be followed through in the safety case, particularly when considering commissioning and early start up operations (RO-ABWR-0072 (Ref. 47) is relevant here); **[Residual Matter 16]**.

200. The BRAC dose rate data is processed to derive an average of all individual plant averages. I queried what would be the impact of taking a simple average of all the data in RQ-ABWR-0766 query 4 (Ref. 34). The response provides information on the results for the original and queried methods on the dose rates, which change relatively little except for the standard deviation which increases. The response further goes on to explain why the approach adopted was reasonable. On the basis of the evidence supplied I would consider the existing simplification sufficient.

201. One of the main assumptions implicit in the Hitachi-GE methodology is that the development of deposited radioactivity is derived from soluble species. In effect the approach takes no account of insoluble (particulate) matter. I wished to understand the sensitivity of this assumption in RQ-ABWR-0766 query 5 (Ref. 34). The RP acknowledges that two processes are taking place simultaneously in growing the deposit films, particular deposition and absorption of soluble species. The technical arguments provided are sound, involving the absorption of Co^{60} into the inner oxide which must mainly be from soluble species not particulate, but in effect the main reason for the correlation adopted is that it shows the best fit for OPEX. I do not think that the response explores the potential impact of this choice completely. However, given the uncertainties elsewhere in the methodology I see little benefit to be gained in terms of refining the UK ABWR DST estimates.

202. I queried a number of assumptions used for the HGE-DST model in query 6 of RQ-ABWR-0766 (Ref. 34). This clarified a number of aspects, but the main outcome of this was the clarification that the effective deposition coefficients derived using JBWR data for NWC plants are considered conservative for HWC conditions. This is based upon comparisons of HGE-DST model results to US OPEX from BWRs operated under HWC. This analysis (of Co^{60}) is contained in Ref. 30 and shows that the HGE-DST model indeed predicts the trends observed, but with larger overall concentrations. The TSC review in Ref. 45 is also relevant when considering the HGE-DST model. While this review raises questions over the suitability of this model in predicting behaviour under HWC conditions I consider this to be a function of the difficulty in making such predictions and take comfort from the OPEX based comparisons provided in the RQ response, which are conservative.

203. Once a dose rate is calculated for UK ABWR the RP proportions this to individual nuclides in accordance to the HGE-DST model predictions. The BRAC data used in the analysis already contains some analysis of key nuclide (CP) ratios. Both these data sources differ. I asked the RP why they chose the former method (RQ-ABWR-0766 query 7 (Ref. 34)). In response, this is due to the model predicting the saturated ratio for 60 years operation. OPEX data is not necessarily representative of this. I consider this to be a reasonable approach for most systems, with the exception of those which are only operated intermittently (such as the RHR or RCIC that are used only at shutdown). The reason for this difference is because the longer lived nuclides may reach equilibrium levels, but the shorter lived nuclides will never do so because they decay during the off periods. The original approach to DST definition did not account for this, but this was reflected in later revisions. Similar modifications were also made to the methods to factor matters such as flow rates and temperature (all of which reduced conservatisms, particularly for the less significant systems).

Fuel Crud DST

204. The fuel crud DST covers the amounts of “loose” (outer) and “adherent” (inner) deposits. A BE and DB value is defined for each. The values calculated represent the activity present of the fuel after permanent discharge from the reactor (i.e. after four 18 month fuel cycles, RQ-ABWR-0763 query 1 refers (Ref. 34)). As with the piping deposit DST a range of nuclides (44) are considered for the fuel estimates, including CP, FP and ActP.

205. The basic methodology applied to derive the fuel crud DST is based on statistical analysis of OPEX from the GEH fuels database. This database contains information for fuel crud analysis conducted on fuel discharged from operational BWRs. The fuel crud scraping process provides information on both the “loose”, outer crud layer and the “adherent”, inner crud layer. These layers are notably different due to the growth processes. Relevant information is mined from this database for a number of plants which operate under the UK ABWR operating chemistry (a smaller number of plants are also reserved for justifying the Hitachi-GE CP model described later). The input data includes parameters such as deposit loadings, specific activity of the crud scrapings and metal concentrations. Corresponding water chemistry data is also sourced and by correlating the two it is possible to derive a DST for UK ABWR. A second, validation method employs the Hitachi-GE CP model that assesses corrosion product transport to the reactor vessel based upon plant performance parameters (feedwater flow rate, CUW flow rate and clean-up efficiency etc.) with assumptions on the corrosion rate of feedwater piping. A third method involves a simple model based upon the mass transport of corrosion product to the reactor vessel, with the assumption that the entire input of corrosion product adheres to fuel bundles. This simplified method employs aspects of the CP model and is used to ensure that the primary and verification methods are bounded within rational limits. A comparison of the ouputs of these three methods is shown below:

[redacted]

Figure 8: Output from the different models for Fuel Crud DST

206. The OPEX method is the preferred method, and the one upon which the UK ABWR DST is based, because it makes direct use of plant data in terms of actual fuel crud build-up and therefore is less prone to uncertainty and sensitivity to assumptions. For nuclides where OPEX isn’t available the RP uses the PST ratio to a known analogous nuclide. Clearly there are many assumptions inherent here, but this is an area of particular uncertainty in any operating BWR (or LWR) with no standard practices

available to determine fuel crud deposition. The approach taken by Hitachi-GE in responding to this aspect of the RI could therefore be considered as state of the art.

207. RQ-ABWR-0765 query 3 (Ref. 34) provided further clarity on this overall methodology. It also clarified that the fundamental assumption is that the BE value is based on a feedwater Fe concentration of [redacted] ppb, whereas the DB values are based on [redacted] ppb. As there is an order of magnitude or greater difference between these two values for most nuclides, the impact of Fe control is again highlighted. Residual Matter 6 raised earlier is relevant here.

Justification

208. A degree of additional justification is provided for both the piping and fuel crud DST values and methodologies in Ref. 30.

<u>Piping DST</u>

209. The justification for the piping DST varies by the system. For the CUW this consists of:

- Comparing the OPEX derived DST with the HGE-DST model values;
- Calculation of 2 plants DST behaviour (Co^{60}) with the HGE-DST model;
- Comparison of the BE and DB values (Co^{60}) for UK ABWR to OPEX from Japanese BWRs under NWC conditions; and
- Comparison of the HGE-DST model results to the Studsvik BWRCrud model.

210. The piping DST for the RHR and FPC are justified by comparing the derived DST values for the UK ABWR to independent OPEX data. Where the RHR is concerned, OPEX data (nuclide data for Co^{60}, Co^{58}, Fe^{59} and Mn^{54}) from both [redacted] and [redacted] is used, and OPEX data (dose rates) from [redacted] is used with respect to the FPC. The LCW DST is compared to [redacted] (does rate) data. For all other systems the justification is a comparison of the predicted Co^{60} to Japanese OPEX (summary data only).

211. As described this justification therefore concentrates on the nuclides (especially Co^{60}) and systems of most significance. It is heavily influenced by the availability of suitable data for comparison. The justification does generally support the UK ABWR values, particularly for Co^{60}, but some obvious differences do exist.

212. I also queried some aspects of the justification for the piping DST. In query 10 of RQ-ABWR-0766 (Ref. 34) I asked the RP what dose rates (BRAC) were measured in plants which were screened out of the OPEX. In response it is noted that a further 16 plants have similar data, but for plants with differing operating chemistry. The response does not provide any values for comparison. I consider that this information would be useful to provide an additional level of justification for the derived UK ABWR values; **[Residual Matter 17]**.

213. Similarly in RQ-ABWR-0767 query 3 (Ref. 34) I asked the RP for as complete a dataset as possible for a comparable plant to UK ABWR. The response included data (dose rate and nuclide composition) for 2 JBWRs for a range of systems. There is no history provided for these plants, but it can be safely assumed that they are older plants which operate under NWC. This data is of limited value, but it does show that the DST trends, certainly for Co^{60}, mirror that predicted for UK ABWR. The absolute concentrations predicted also compare favourably, within the large uncertainties that might be expected.

<u>Fuel Crud DST</u>

214. The justification approach for the fuel crud DST follows the same methodology as the derivation, but uses completely independent data. As described previously the main tool used to justify the fuel crud DST is the Hitachi-GE CP model. This is specifically

used to model fuel crud from plants not included within the DST derivation dataset. Again, this model is reviewed by my TSC in Ref. 45, and the conclusions are not favourable. However, I consider its use here reasonable. In a similar manner the simple mass transfer model is employed for the separate BWR data. The outputs are compared with fuel scrape data within the GEH database. These comparisons show reasonable agreement.

Values

215. Ref. 33 contains the tabular DST values derived from the above methods. A single set of values is presented for each system considered.

216. As with the other aspects of the UK ABWR source terms I chose to sample a number of nuclides across various systems and the fuel crud DST, to confirm that I could trace their derivation through. I was satisfied in most instances, but this led to a number of detailed queries on the DST values in RQ-ABWR-0763 (Ref. 34). These were centred on understanding how the values defined align with the methodologies. Most of these RQs were answered by providing additional clarification on the methodologies. However, it is worth noting query 4 and its response. When comparing the respective BE and DB values for the fuel crud DST it is notable that the amount of deposit increases generally, as expected, but not uniformly across the different nuclides and loose or adherent forms. I requested an explanation for this behaviour. The response explains the complexity and uncertainties in development of fuel crud and helps to put the calculated UK ABWR DST values into context.

217. In RQ-ABWR-0767 query 3 (Ref. 34) I asked Hitachi-GE to provide as complete a dataset as possible for a comparable plant. In response the RP provided DST values for two JBWRs. The data is somewhat limited, but does show similar trends to that predicted for UK ABWR. The most meaningful comparison is for Co^{60} which is within the same order of magnitude (for BE values). It is notable that the comparisons tend to diverge more for those systems with lower DST values, perhaps due to conservatisms in the derivation method.

218. As described above, the majority of definition and justification is centred on Co^{60} and a small number of other important CPs. There is therefore very little information on a number of other nuclides defined within the DST. From a purely theoretical point of view I would question the values for a number of these other nuclides. Their concentrations seem large and are not what I might expect. However, this has to be balanced against their safety significance which is lower. I also consider the DST values to be rather conservative, certainly the DB values and especially in some of the systems where the DST is low anyway. All things considered, I see little benefit in further refinements in this regard.

Summary for DST

219. The DST is the part of the source terms with the greatest uncertainty. It is also defined using the most complex methodology. Based on my assessment I judge that:

- When taken collectively, the RP has defined a set of DST values for piping and fuel crud deposits which appear suitable for further use within the UK ABWR safety case;
- While some of the values may be questionable, I am content that those which are most safety significant have been justified to a proportionate degree;
- For those matters where I am less content with the methodologies, assumptions or approach I am satisfied that any changes would not have a significant impact on the derived values;
- While I have identified a number of Residual Matters for follow up, I do not consider any of these to be material to resolving the RI.

220. As noted earlier the original versions of Refs 26, 30 and 33 were poor, but the issues with these have been largely resolved and they now present an adequate description of the definition and justification process for the DST.

4.2.7 Management of Source Terms

221. While RI-ABWR-0001 does not have any Actions associated with the management of source terms, these do form part of RO-ABWR-0006 (Ref. 10); in particular Actions 3, 7 and 8. As part of assessing the RI responses some aspects relating to the management of source term data became apparent. These are documented briefly in this section of my assessment, for completeness.

222. As described throughout my assessment various changes were necessary to the source terms for UK ABWR, most notably the change in FP methodology that resulted from RQ-ABWR-0870 (Ref. 34). In effect, this was a demonstration that the RP had control over the UK ABWR source term data and could reflect any changes as necessary in an accurate manner. I have sampled various aspects of these updates and am content that they show that the UK ABWR source terms are managed adequately.

223. As is evident from the assessment above, the response to RI-ABWR-0001 contains a huge amount of OPEX data, which is manipulated, filtered and used as part of calculations. None of the main submissions (Refs 22 to 33) include, nor make reference to this data, in either raw or manipulated form. I queried how this data was recorded, maintained and stored in RQ-ABWR-0764 (Ref. 34). The response provided information for the raw (OPEX) data, which is maintained within a database, but did not consider the intermediate stages where it is manipulated. The response to RQ-ABWR-0907 (Ref. 34) considers this and commits to provide an additional report to document this, in addition to updates to other RO-ABWR-0006 submissions. I am content that these revised arrangements are appropriate.

224. As an important part of the safety case for UK ABWR it is essential that any assumptions are appropriately reflected within ORs. For example, this could include limits on radioactivity itself or on the concentration of any precursors. RQ-ABWR-0683 query 3 and RQ-ABWR-0822 (Ref. 34) were raised to understand how this process would be managed. The initial response to the first RQ is inadequate, indicating that the RP does not consider there to be any such links necessary. The latter response is improved, at least acknowledging that such a link is necessary, but there are many aspects which I would not consider to be aligned with ONRs expectations and guidance related to ORs. This is not important to resolving the RI (as it is in fact an output from the RI submissions) but does need to be an area for continuing regulatory interactions with Hitachi-GE; **[Residual Matter 18]**.

225. I am therefore content that Hitachi-GE have demonstrated that they are managing the UK ABWR source terms appropriately as part of resolving RI-ABWR-0001.

4.3 Comparison with Standards, Guidance and Relevant Good Practice

226. As described in Section 2.3 of my assessment, there are no standards and guidance which directly relate to the definition or justification for a source term. Those applicable are more generic and relate to production of an adequate safety case. I am content that the responses provided by Hitachi-GE to this RI meet the expectations contained within these.

227. I have considered specific aspects of Relevant Good Practice (RGP) throughout my assessment and cite this throughout the previous sections. In a more general sense, RGP is considered within the regulatory expectations for RI-ABWR-0001. This encompasses the expectations defined in the RI (Annex 1), the regulatory expectations

of RO-ABWR-0006 (Ref. 9) and the feedback given in letter REG-HGNE-0077R (Ref. 19). Further regulatory guidance on source terms is contained in Ref. 20. In summary, an adequate definition of the UK ABWR source terms should:

- Cover all significant radionuclides;
- Cover all systems which are expected to contain radioactivity;
- Cover all operational states;
- Cover all appropriate sources of radioactivity within the plant, including mobile and fixed sources;
- Consider how the nature and quantity of radioactivity within the plant may change over time;
- Cover all aspects of the safety or environmental case for UK ABWR;
- Be consistent with how the defined source terms are used by, and support, these cases; and
- Be consistent with the design and operations of UK ABWR.

228. Similarly an adequate justification should:

- Provide an appropriate degree of robust supporting evidence for the defined source terms;
- Cover the full scope of the definition, but be targeted towards those radionuclides, systems or operations which have the highest safety or environmental impact; and
- Be demonstrated to be appropriate for the UK ABWR and consistent with the extant safety and environmental cases.

229. I am content that the responses to RI-ABWR-0001 meet these expectations and therefore are consistent with RGP.

4.4 Residual Matters

230. On the basis of my assessment I have identified 18 residual matters. These are given in Annex 2, which also details the ONR technical disciplines that may be impacted by these.

231. However, it is not appropriate for this report to consider these further in order to identify if they should be Assessment Findings or Minor Shortfalls (as per Ref. 8) as this assessment report does not represent the full judgement on all aspects of this topic for GDA of UK ABWR. This will be reported in the assessment reports produced on a discipline basis at the end of Step 4. The identified Residual Matters from this report should be addressed during the remainder of GDA Step 4, or beyond, as judged appropriate within the relevant ONR technical disciplines.

232. I do not consider that any of these Residual Matters prevents resolution of RI-ABWR-0001.

4.5 ONR Assessment Rating

233. In line with the ONR Assessment Rating guidance (Ref. 48) I judge this assessment to have an AMBER rating.

234. The rationale for this rating is that the original submissions made to resolve the RI were insufficient, required significant changes to both methodology and resultant values and resulted in a large number of RQs to clarify and provide additional evidence.

5 CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions

235. This report presents the findings of the my assessment of submissions provided by Hitachi-GE in response to Regulatory Issue RI-ABWR-0001 (Definition and Justification for the Radioactive Source Terms in UK ABWR during Normal Operations (Ref. 1)).

236. The purpose of this assessment was three-fold;

- To document the assessment which underpins the recommendation made in closing RI-ABWR-0001, or otherwise;
- To serve as a record of the scope of the assessment undertaken for RI-ABWR-0001, and therefore the boundaries of the judgements made; and
- To identify any associated residual matters which may need to be satisfactorily addressed during the remainder of GDA Step 4, or beyond, as appropriate.

237. In response to RI-ABWR-0001 Hitachi-GE have provided a suite of documentation (Refs 22 to 33) which defines and justifies the concentration of radionuclides around the UK ABWR plant during all modes of normal operations. This includes radioactivity with the reactor, water and gaseous auxiliary systems as well as deposited on piping surfaces and fuel cladding. In addition the Requesting Party provided responses to my Regulatory Queries, providing additional clarification and evidence to support the main submissions.

238. The main conclusions of my assessment are:

- I consider that the scope and approach adopted by Hitachi-GE in responding to the RI is adequate;
- The use of relevant OPEX, utilising the broadest data set that is considered pertinent, gives confidence in the defined values. Where suitable OPEX does not exists recourse is made to other methods to provide the data in an satisfactory manner;
- Throughout the development of the source term I am content that suitable and sufficient consideration has been given to safety, including consideration of all significant radionuclides that exist in the systems expected to contain radioactivity throughout the envisaged operational states;
- The defined UK ABWR source terms now includes all appropriate sources of radioactivity within the plant, including mobile and fixed sources, and considers how the nature and quantities of radioactivity within the plant may change over time;
- Variations in radioactivity due to the different operational phases of the plant both in the short term and long term are appropriately considered, covering the entire fuel cycle;
- Both Best Estimate and Design Basis values are defined, representing an expected and more conservative estimate for the likely levels of radioactivity within UK ABWR. I am content that the BE values derived represent a reasonable estimate, for safety case purposes, of the likely performance of UK ABWR. I am content that the RP has derived a set of conservative DB values which should be suitable for use in the safety case (noting that it was outside the scope of my assessment to assess the application to individual end user topics);
- For those matters where I am less content with the methodologies, assumptions or approach I am satisfied that any changes would not have a significant impact on the derived values;
- The derived values are further justified using additional OPEX, calculation, literature and sensitivity analysis. An adequate and proportionate degree of

supporting evidence has been provided, which is focussed on those nuclides of highest safety significance;

- While the responses have been updated several times throughout my assessment I am content that sufficient has been documented to capture and understand the basis of the UK ABWR source terms should this need to be revisited in the future.

239. My overall view on the UK ABWR source terms is that it is now fit for purpose in making the UK ABWR safety case. While further changes may still occur I am confident that these should only be minor in nature.

240. While I have identified a number of Residual Matters through the course of my assessment I do not consider any of these to be significant enough to prevent closure of the RI.

241. To conclude based on my assessment, I am content that Hitachi-GE have provided sufficient to meet the intent of RI-ABWR-0001 and have addressed the issues which led to it being raised. I am therefore content that the RI has been resolved.

5.2 Recommendations

242. My recommendations are as follows.

- <u>Recommendation 1</u>: RI-ABWR-0001 should be closed.

- <u>Recommendation 2</u>: The Residual Matters identified in this report should be considered by the relevant UK ABWR discipline inspectors and actioned as considered appropriate.

- <u>Recommendation 3</u>: The evidence provided as part of the resolution of RI-ABWR-0001 regarding management of the source terms should be considered as part of RO-ABWR-0006 Actions 3, 7 and 8.

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18. *Topic Report 2: Demonstration and Justification of the Source Term for the UK ABWR*, GA91-9201-0001-00108, HE-GD-5089, Revision 0, 15 January 2015. TRIM Ref. 2015/20346.

19. *Letter to Hitachi-GE from ONR*, REG-HGNE-0077R, 6 March 2015. TRIM Ref. 2015/79410.

20. *Regulators' Feedback on RO-ABWR-0006 (Source Terms)*, 24 February 2015, ONR. TRIM Ref. 2015/62364

21. *Source Terms Workshop – Approach in UK Reactor Safety Cases*, 19 May 2015, ONR. TRIM Ref. 2015/183790

22. *Source Term Strategy Report*, GA91-9201-0003-00864, HE-GD-0107, Revision 1, 1 October 2015. TRIM Ref. 2015/296470.

23. *Source Term Manual General Report*, GA91-9201-0003-00942, HE-GD-0117, Revision 1, 27 November 2015. TRIM Ref. 2015/449298.

24. *Primary Source Term Methodology Report*, GA91-9201-0003-00863, WPE-GD-0184, Revision 1, 1 October 2015. TRIM Ref. 2015/366540. [Latest version is Revision 2, 29 July 2016, TRIM Ref. 2016/303858]

25. *Process Source Term Methodology Report*, GA91-9201-0003-00946, HE-GD-5135, Revision 0, 7 October 2015. TRIM Ref. 2015/374272. [Latest version is Revision 3, 29 July 2016, TRIM Ref. 2016/304017]

26. *Deposit Source Term Methodology Report*, GA91-9201-0003-00960, WPE-GD-0201, Revision 0, 23 October 2015. TRIM Ref. 2015/397356. [Latest version is Revision 2, 29 July 2016, TRIM Ref. 2016/304047]

27. *Nuclide Selection by End User Requirement*, GA91-9201-0003-00941, HE-GD-0116, Revision 0, 1 October 2015. TRIM Ref. 2015/366588.

28. *Primary Source Term Supporting Report*, GA91-9201-0003-00929, WPE-GD-0195, Revision 0, 1 October 2015. TRIM Ref. 2015/366561. [Latest version is Revision 1, 29 July 2016, TRIM Ref. 2016/304006]

29. *Process Source Term Supporting Report,* GA91-9201-0003-00945, HE-GD-5136, Revision 0, 7 October 2015. TRIM Ref. 2015/374246. [Latest version is Revision 3, 29 July 2016, TRIM Ref. 2016/304024]

30. *Deposit Source Term Supporting Report*, GA91-9201-0003-00959, WPE-GD-0202, Revision 0, 20 November 2015. TRIM Ref. 2015/440257. [Latest version is Revision 2, 29 July 2016, TRIM Ref. 2016/304041]

31. *Calculation of Primary Source Term Value*, GA91-9201-0003-00928, WPE-GD-0196, Revision 0, 1 October 2015. TRIM Ref. 2015/366551. [Latest version is Revision 3, 20 June 2016, TRIM Ref. 2016/247706]

32. *Calculation of Process Source Term Value*, GA91-9201-0003-00944, HE-GD-5137, Revision 0, 9 October 2015. TRIM Ref. 2015/378529. [Latest version is Revision 4, 24 June 2016, TRIM Ref. 2016/255177]

33. *Calculation of Deposit Source Term Value*, GA91-9201-0003-00961, WPE-GD-0203, Revision 0, 24 November 2015. TRIM Ref. 2015/443220. [Latest version is Revision 3, 24 June 2016, TRIM Ref. 2016/255227]

34. *List of Source Term related Regulatory Query Responses*. TRIM Ref. 2016/422154.

35. *Nuclide Selection by End User Requirement*, GA91-9201-0003-00941, HE-GD-0116, Revision 2, 29 June 2016. TRIM Ref. 2016/260640.

36. *The Radiochemistry of Nuclear Power Plants with Light Water Reactors*, Karl-Heinz Neeb, Walter de Gruyter, 1997. ISBN 3110132427.

37. *Step 3 Supporting Report - Compilation of Publicly Available Source Term Information for BWRs under Normal Operations and Initial Comparisons to UK ABWR*. TRIM Ref. 2015/238259.

38. *BWRVIP-190: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines – 2008 Revision*. Report 1016579. EPRI. October 2008. www.epri.com

39. *LCC6 Special Topic Report - radioactivity in LWRs*. ANT International. November 2010. TRIM Ref. 2014/440115.

40. *Review of Fuel Failures in Water Cooled Reactors*, NF-T-2.1, 2010. IAEA. www.iaea.org.

41. *Management of damaged fuel*, GA91-9201-0003-01019, UE-GD-0473, Revision 0, April 2016. TRIM Ref. 2016/178093.

42. *Topic Report on Noble Metal Chemical Addition*, GA91-9201-0001-00088, WPE-GD-0088, Revision 1, April 2016. TRIM Ref. 2016/175696.

43. *Topic Report on Zinc Injection*, GA91-9201-0001-00089, WPE-GD-0089, Revision 1, April 2016. TRIM Ref. 2016/177709.

44. *Topic Report on Design Justification in Chemistry Aspect for Primary Water System*, GA91-9201-0001-00199, WPE-GD-0232, Revision 1, August 2016. TRIM Ref. 2016/344015.

45. *Review of Hitachi-GE Modelling of Activity Transport in the UK-ABWR*, SPR07473/06/10/06, NNL. TRIM Ref. 2016/422218.

46. *Process and Information Document – Generic Design Assessment*. Version 2. Environment Agency. March 2013. http://cdn.environment-agency.gov.uk/LIT_7998_3e266c.pdf

47. *RO-ABWR-0072 – Commissioning Chemistry*, Rev. 0, 15 August 2016. TRIM Ref. 2016/322866. www.onr.org.uk/new-reactors/uk-abwr/reports/ro-abwr-0072.pdf

48. *ONR Assessment Ratings Guide Table*. TRIM Ref. 2016/118638.

Table 1

Regulatory Queries (RQs) Raised During the Assessment

RQ Number	RQ Title	Number of queries	Source Term Area	Response TRIM Ref
RQ-ABWR-0681	Primary source term – actinides	4	PST	2016/42446
RQ-ABWR-0682	Primary source term – activation products (1)	3	PST	2016/46196
RQ-ABWR-0683	Primary source term – activation products (2)	6	PST	2016/86493
RQ-ABWR-0684	Primary source term – general approach and assumptions (1)	3	PST	2016/86267
RQ-ABWR-0685	Primary source term – general approach and assumptions (2)	5	PST	2016/86461
RQ-ABWR-0686	Primary source term – corrosion products (1)	1	PST	2016/42942
RQ-ABWR-0687	Primary source term – corrosion products (2)	6	PST	2016/86499
RQ-ABWR-0688	Primary source term – cycle average	3	PST	2016/42998
RQ-ABWR-0689	Primary source term – fission products (1)	5	PST	2016/42487
RQ-ABWR-0690	Primary source term – fission products (2)	5	PST	2016/86158
RQ-ABWR-0691	Primary source term – nuclide selection	1	PST	2016/46219
RQ-ABWR-0692	Primary source term – values (1)	4	PST	2016/46215
RQ-ABWR-0693	Primary source term – values (2)	3	PST	2016/86331
RQ-ABWR-0740	Application of Nuclide Selection Report	3	PST	2016/86451
RQ-ABWR-0741	Demonstration of adequate operating margin in the UK ABWR source terms	1	Margin	2016/160172
RQ-ABWR-0742	Iodine behaviour during normal operations	1	Iodine	2016/141954
RQ-ABWR-0763	Deposit Source Terms – values	4	DST	2016/138761
RQ-ABWR-0764	Source Terms – use of source term values	3	Management	2016/159978
RQ-ABWR-0765	Deposit Source Terms – fuel crud deposits	5	DST	2016/160046
RQ-ABWR-0766	Deposit Source Terms – piping deposits	10	DST	2016/160082
RQ-ABWR-0767	Deposit Source Terms – scope and approach	3	DST	2016/160110
RQ-ABWR-0771	Process Source Terms – scope and approach	6	PrST	2016/159893
RQ-ABWR-0772	Process Source Terms – detailed queries	6	PrST	2016/159932
RQ-ABWR-0773	Process Source Terms – values	3	PrST	2016/159950
RQ-ABWR-0793	Follow up on RQ-ABWR-0682	2	PST	2016/177491
RQ-ABWR-0794	Follow up on RQ-ABWR-0686	1	PST	2016/177809
RQ-ABWR-0795	Follow up on RQ-ABWR-0688	1	PST	2016/177835
RQ-ABWR-0796	Follow up on RQ-ABWR-0681	2	PST	2016/194079
RQ-ABWR-0797	Follow up on RQ-ABWR-0691	1	PST	2016/177511
RQ-ABWR-0798	Follow up on RQ-ABWR-0689	4	PST	2016/194078
RQ-ABWR-0799	Follow up on RQ-ABWR-0692	2	PST	2016/177625
RQ-ABWR-0822	Follow up on RQ-ABWR-0683	1	PST	2016/241886
RQ-ABWR-0823	Follow up on RQ-ABWR-0690	1	PST	2016/200175
RQ-ABWR-0824	Follow up on RQ-ABWR-0693	1	PST	2016/200124

RQ Number	RQ Title	Number of queries	Source Term Area	Response TRIM Ref
RQ-ABWR-0870	Methodology to derive the source term associated with fuel failure events	3	PST	2016/245435
RQ-ABWR-0904	Follow-up on RQ-ABWR-0742 (iodine in normal operations)	1	Iodine	2016/267253
RQ-ABWR-0906	Source term used for determining the shield design of UK ABWR	1	Margin	2016/260619
RQ-ABWR-0907	Management of source term data (Follow-up on RQ-ABWR-0764)	1	Management	2016/284410

Table 2

Relevant Safety Assessment Principles Considered During the Assessment

SAP No	SAP Title	Description
FP.4	Fundamental principles – Safety assessment	Dutyholders must demonstrate effective understanding and control of the hazards posed by a site or facility through a comprehensive and systematic process of safety assessment.
SC.4	The regulatory assessment of safety cases – Safety case characteristics	A safety case should be accurate, objective and demonstrably complete for its intended purpose.
SC.5	The regulatory assessment of safety cases – Optimism, uncertainty and conservatism	Safety cases should identify areas of optimism and uncertainty, together with their significance, in addition to strengths and any claimed conservatism.
SC.6	The regulatory assessment of safety cases – Safety case content and implementation	The safety case for a facility or site should identify the important aspects of operation and management required for maintaining safety and how these will be implemented.

Table 3

Relevant Technical Assessment Guides Considered During the Assessment

Reference	Revision	Title
NS-TAST-GD-004	5	Fundamental principles
NS-TAST-GD-051	4	The purpose, scope and content of nuclear safety cases

Annex 1

RI-ABWR-0001 - Definition and Justification for the Radioactive Source Terms in UK ABWR during Normal Operations

REGULATORY ISSUE	
REGULATOR TO COMPLETE	
RI unique no.:	RI-ABWR-0001
Date sent:	2 June 2015
Acknowledgement required by:	8 June 2015
Agreement of Resolution Plan Required by:	15 June 2015
Resolution of Regulatory Issue required by:	*To Be Determined By The Hitachi-GE Resolution Plan.*
TRIM Ref.:	2015/202107
Related RQ / RO No. and TRIM Ref. (if any):	RO-ABWR-0006 (TRIM Ref. 2014/463098)
Issue title:	Definition and Justification for the Radioactive Source Terms in UK ABWR during Normal Operations
Technical area(s) 9. Reactor Chemistry 21. Generic Environmental Permitting	**Related technical area(s)** 4. PSA 5. Fault Studies 8. Fuel Design 9. Reactor Chemistry 10. Radiation Protection & (Level 3 PSA) 12. Structural Integrity 13. Human Factors 15. Radwaste & Decommissioning

Regulatory Issue

SUMMARY

The objective of this Regulatory Issue (RI) is to state the regulators (ONR and the Environment Agency) expectations with respect to Hitachi-GE providing a suitable and sufficient definition and justification for the radioactive source terms in UK ABWR during normal operations.

The definition of the radioactive source term; the nature and amount of radioactivity, is a fundamental part in understanding and therefore being able to control the hazards associated with any nuclear facility. Once defined, it is important that the Requesting Party (RP) is able to demonstrate and justify that this source term is appropriate to be used as the basis for the safety and environmental cases. Failure to adequately define or justify the source term could ultimately mean that the design, operations or controls specified for the UK ABWR may not be soundly based. It is therefore important that Hitachi-GE submit a resolution plan which provides sufficient regulatory confidence that the source terms can be defined and justified.

During Step 2 of GDA, the regulators jointly raised RO-ABWR-0006 in April 2014 requesting Hitachi-GE to define and justify the UK ABWR source terms, amongst other related matters. Hitachi-GE responded with their definition and justification in January 2015 in accordance with its schedule as defined in its resolution plan for RO-ABWR-0006. Overall, the regulators judge that the responses do not meet our expectations as they do not provide a complete or suitably robust definition and justification for the source terms expected in UK ABWR during normal operations. This is considered to be a serious regulatory shortfall which the regulators, in line with our Guidance to Requesting Parties and our Process and Information Document, are now escalating to a Regulatory Issue.

BACKGROUND

The definition and appropriate use of the "source term" is important in understanding, and therefore controlling, the hazards posed by any nuclear facility. In this context, the regulators defined source terms as:

The types, quantities, and physical and chemical forms of the radionuclides present in a nuclear *facility that have the potential to give rise to exposure to ionising radiation, radioactive waste or discharges.*

During Step 2 the regulators noted that there was a lack of information on the radiological source terms for the UK ABWR. This information would form a key part of justifying the design going forward, both from a safety and environmental perspective. Three main areas were identified where further justification and evidence would be required from Hitachi-GE, namely:

- To define and justify the source terms for UK ABWR, including how these are used;
- To demonstrate the impact of the material choices, operating chemistry and operating practices on radioactivity in the plant and to show that these reduce radioactivity So Far As Is Reasonably Practicable (SFAIRP); and
- To show that the source term information is adequately managed and controlled throughout the safety and environmental cases.

To address these aspects the regulators (ONR and the Environment Agency) jointly raised a Regulatory Observation (RO) related to the source terms in the UK ABWR, RO-ABWR-0006 [1] in April 2014. This RO was associated with all of these aspects, including the definition (Action 1) and supporting evidence that was considered necessary to justify (Action 2) the source terms for the UK ABWR design during "operational states" [2] and "expected events" [3] (see also the glossary for these terms). Other actions under RO-ABWR-0006 deal with management and justification that radioactivity is reduced SFAIRP, but these are not within the scope of this Regulatory Issue. Responses to Actions 1 and 2 were received during January 2015 [4, 5].

The regulators provided detailed feedback to Hitachi-GE on these responses during technical meetings in January and February 2015, and followed this up with letter REG-HGNE-0077R [6]. The regulators also provided additional technical advice to Hitachi-GE during March and April 2015. It is clear that some uncertainty still remains, as insufficient progress has been made to build regulatory confidence in the approach proposed by Hitachi-GE to address the shortfalls identified with RO Actions 1 and 2.

The regulators have judged Hitachi-GE's responses are not adequate to resolve RO Actions 1 and 2 because:

- The approach taken, of calculation of the source terms, means that there are inherently many assumptions, some of which would appear to impose a significant sensitivity on the results. These have not been appropriately justified;
- The definition of an "average" source term does not cover all potential transients, operational occurrences or operations expected at the plant, as requested in RO-ABWR-0006;
- The amount of fixed radioactivity (contamination) is inadequately defined and substantiated, with no supporting evidence;
- The scope of the defined source terms is incomplete with some significant aspects missing;
- The corrective factor applied when the source terms are used for specific purposes does not appear to be conservative;
- There is no link between the defined source terms and the extant UK ABWR safety and environmental cases; and
- A suitably robust demonstration and justification for the adequacy of the defined source terms has not been provided.

The regulators consider a robust source term to be a crucial aspect of the UK ABWR safety and environmental cases. The impact of the source term is significant for the GDA of UK ABWR, due to the large number of topics and areas which rely on this information. Overall, the responses received do not provide a complete or suitably robust definition and justification for the source terms expected in the UK ABWR during normal operations. This is considered to be a serious regulatory shortfall which the regulators, in line with our Guidance to Requesting Parties [7] (paras. 159 and 160), are now escalating to a Regulatory Issue.

REGULATORY EXPECTATIONS

The regulatory expectations are the same as those defined under RO-ABWR-0006 Actions 1 and 2 [1]. Overall, the regulators expect Hitachi-GE to provide a suitable and sufficient definition and justification for the source terms for the UK ABWR.

The definition should:

- Cover all significant radionuclides;

- Cover all systems which are expected to contain radioactivity;
- Cover all operational states;
- Cover all appropriate sources of radioactivity within the plant, including mobile and fixed sources;
- Consider how the nature and quantity of radioactivity within the pant may change over time;
- Cover all aspects of the safety or environmental case for UK ABWR;
- Be consistent with how the defined source terms are used by, and support, these cases; and
- Be consistent with the design and operations of UK ABWR.

The justification should:
- Provide an appropriate degree of robust supporting evidence for the defined source terms;
- Cover the full scope of the definition, but be targeted towards those radionuclides, systems or operations which have the highest safety or environmental impact; and
- Be demonstrated to be appropriate for the UK ABWR and consistent with the extant safety and environmental cases.

References:

[1] Regulatory Observation – Source Terms, RO-ABWR-0006. www.onr.org.uk/new-reactors/uk-abwr/reports/ro-abwr-0006.pdf
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Glossary: Terminology Used in Nuclear Safety and Radiation Protection, 2007 Edition, IAEA, Vienna (2007). www.iaea.org
[3] Process and Information Document – Generic Design Assessment. Version 2. Environment Agency. March 2013. http://cdn.environment-agency.gov.uk/LIT_7998_3e266c.pdf
[4] Topic Report 1: Definition of the UK ABWR Design Source Term, GA91-9201-0001-00107, HE-GD-5088, Revision 0, 15 January 2015.
[5] Topic Report 2: Demonstration and Justification of the Source Term for the UK ABWR, GA91-9201-0001-00108, HE-GD-5089, Revision 0, 15 January 2015.
[6] Letter to Hitachi-GE from ONR, REG-HGNE-0077R, 6 March 2015.
[7] New nuclear reactors: Generic Design Assessment Guidance to Requesting Parties, ONR-GDA-GD-001 Revision 1, August 2014. www.onr.org.uk/new-reactors/ngn03.pdf

Glossary:

Expected event – events that are expected to occur over the lifetime of the plant. This does not include events that are inconsistent with the use of best available techniques such as accidents, inadequate maintenance and inadequate operation.
Operational States – Including "normal operations" and "anticipated operational occurrences". For a nuclear power plant, this includes start-up, power operation, shutting down, shutdown, maintenance, testing and refuelling.
Source term – The types, quantities, and physical and chemical forms of the radionuclides present in a nuclear facility that have the potential to give rise to exposure to radiation, radioactive waste or discharges.

Regulatory Issue Actions

RI-ABWR-0001.A1 – *Hitachi-GE is required to provide a suitable and sufficient definition for the radioactive source terms for UK ABWR during normal operations.*

The scope of this Action is the same as that defined under RO-ABWR-0006 Action 1.

The response to this Action should:
- *Meet the regulatory expectations defined in this RI;*
- *Address the regulatory expectations of RO-ABWR-0006 Action 1 [1]; and*
- *Address the feedback given in letter REG-HGNE-0077R [6].*

RESOLUTION REQUIRED BY: *To Be Determined By The Hitachi-GE Resolution Plan.*

RI-ABWR-0001.A2 – *Hitachi-GE is required to provide a suitable and sufficient justification for the radioactive source terms for UK ABWR during normal operations.*

The scope of this Action is the same as that defined under RO-ABWR-0006 Action 2.

The response to this Action should:

- *Meet the regulatory expectations defined in this RI;*
- *Address the regulatory expectations of RO-ABWR-0006 Action 2 [1]; and*
- *Address the feedback given in letter REG-HGNE-0077R [6].*

RESOLUTION REQUIRED BY*: To Be Determined By The Hitachi-GE Resolution Plan.*

REQUESTING PARTY TO COMPLETE	
Actual Acknowledgement date:	
RP stated Resolution Plan agreement date:	

Annex 2

Residual Matters Identified During the Assessment

Number	Description	Paragraph	Affected ONR Technical Discipline(s)
1	Assessment is required on how the derived source terms are used or applied within particular technical aspects of the safety case	16	Radiation Protection Radioactive Waste Decommissioning Fault Studies Radiological Consequences PSA
2	Assessment is required of updates to the PCSR to reflect responses to RI-ABWR-0001	18	All
3	Assessment is required of updates to the Master Document Submission List to reflect responses to RI-ABWR-0001	18	Reactor Chemistry
4	Each “end user” is required to assess the adequacy of the approach to down selection of the consolidated list of nuclides defined in the RI-ABWR-0001 response to the relevant EUST subset	89	Radiation Protection Radioactive Waste Decommissioning Fault Studies Radiological Consequences PSA
5	Further evidence is required on the impact on NMCA and Zn on radioactivity, specifically in terms of loss of dosing, timings or transients (RQ-ABWR-0685 query 5 refers)	96	Reactor Chemistry
6	Assessment is required to ensure that the impact of feedwater Iron concentration control is appropriately reflected in the safety case for UK ABWR	104	Reactor Chemistry
7	Evidence is required on to what extent moisture carry over at the end of cycle is expected for UK ABWR, and the impact of this on any activity transport	107	Reactor Chemistry
8	Further evidence is required on the actual performance expected in the UK ABWR clean-up systems (CUW, SFP, CD)	110	Reactor Chemistry
9	Evidence is required to show that Power Suppression Testing stops fuel failures from escalating, that suitable methods are available to detect further degradation before it become significant and that such operations reduce risks SFAIRP	125	Reactor Chemistry Fuel and Core
10	Further evidence is required to demonstration that the I^{131} shutdown relationship is valid for plants outside Japan and for the UK ABWR operating chemistry regime	128	Reactor Chemistry
11	Further evidence is required the characterise the release of fission products during a shutdown transient (RQ-ABWR-0798 query 4 refers)	129	Reactor Chemistry
12	Further evidence is required to justify the defined N^{16} concentrations in both reactor water and steam	136	Reactor Chemistry
13	Assessment is required for where the UK ABWR defined source terms reflect into the plant design, particularly in areas where the UK ABWR values are higher	161	Radiation Protection Radioactive Waste Decommissioning
14	Further clarification is required within the source term documentation regarding the uncertainties and reliabilities of the various derived values, particularly for the PST.	165	Reactor Chemistry

Number	Description	Paragraph	Affected ONR Technical Discipline(s)
15	Further evidence is required on iodine behaviour in the off-gas system and effects as part of RO-ABWR-0066	184	Reactor Chemistry Fault Studies Radiological Consequences
16	Further evidence is required for the growth of DST in UK ABWR given the intention to operate with HWC, OLNC and DZO from first operations	199	Reactor Chemistry
17	Further evidence is required for BRAC dose rate data for plants screened out of the UK ABWR analysis	212	Reactor Chemistry
18	Assessment is required on the impact of RI-ABWR-0001 methodologies and approach on ORs necessary for UK ABWR (both radioactive species or precursors)	224	Reactor Chemistry