Office for Nuclear Regulation

New Reactors Division

Step 4 Assessment of Probabilistic Safety Analysis for the UK Advanced Boiling Water Reactor

Assessment Report: ONR-NR-AR-17-014
Revision 0
December 2017

EXECUTIVE SUMMARY

Hitachi-GE Nuclear Energy Ltd. is the designer and Generic Design Assessment (GDA) requesting party (RP) for the United Kingdom Advanced Boiling Water Reactor (UK ABWR). Hitachi-GE commenced GDA in 2013 and completed Step 4 in 2017.

This assessment report is my Step 4 assessment of the Hitachi-GE UK ABWR reactor design in the area of probabilistic safety analysis (PSA).

The scope of the Step 4 assessment is to review the safety aspects of the UK ABWR in greater detail, by examining the evidence supporting the claims and arguments made in the safety documentation, building on the assessments already carried out for Step 3. In addition I have provided a judgement on the adequacy of the PSA information contained within the pre-construction safety report (PCSR) and supporting documentation.

Based upon the submissions made by Hitachi-GE during Steps 2 and 3 of the GDA for the UK ABWR, the Office for Nuclear Regulation (ONR) judged that there were serious regulatory shortfalls associated with the development of a modern standards full-scope PSA for the UK ABWR, suitable and sufficient for ONR to carry out a meaningful assessment within the project timescales. These had the potential to prevent provision of a Design Acceptance Confirmation (DAC). In line with the guidance to requesting parties, ONR therefore raised regulatory issue (RI) RI-ABWR-0002 (supported by a number of regulatory observations (ROs) and regulatory queries (RQs). The aim of RI-ABWR-0002 was to make regulatory expectations clear and to ensure that these shortfalls were addressed during GDA.

In response to RI-ABWR-0002, Hitachi-GE extended its PSA capability, improved the processes to support the development and use of the PSA and submitted a revised UK ABWR PSA. Following ONR assessment of Hitachi-GE submissions, RI-ABWR-0002 was closed during Step 4.

My assessment conclusions for Step 4 GDA in the area of PSA are:

- Based on my assessment, I have concluded that the UK ABWR PSA developed by Hitachi-GE, including the developments in response to regulatory issue (RI) RI-ABWR-0002 and the supporting regulatory observations (ROs) and regulatory queries (RQs), broadly meets the expectations of ONR's PSA Technical Assessment Guide (TAG) and is adequate to support the generic PCSR.

- The UK ABWR PSA has a credible and defensible basis and allows for comparison against the numerical risk targets contained within ONR's safety assessment principles (SAPs). Comparison of the results of the UK ABWR PSA against SAPs Target 9 shows that the estimated risk is well below (approximately an order of magnitude) the basic safety level. However, the risk remains above the basic safety objective for SAPs Target 9. Therefore, increased regulatory attention was given during my review to the demonstration by Hitachi-GE that the large release frequency was reduced to as low as reasonably practicable (ALARP).

- The PSA has been adequately used during GDA to ensure that risks are being managed towards an ALARP position as the design continues through GDA and into the site specific stage. The PSA has been used to identify ALARP improvements which have been incorporated into the GDA reference design and to identify potential ALARP improvements for further consideration beyond GDA. My assessment has not found any major areas of the plant design for which additional ALARP analysis was needed in GDA, from a PSA point of view, to consider alternative features. However, further work is needed early in the site specific stage to consider the potential ALARP options identified during GDA and identify any new ALARP insights resulting from the development of the site specific PSA.

- The scope and content of the PSA is adequate for GDA. However the PSA needs to be revised beyond GDA to reflect the final detailed design, address shortfalls identified by the GDA review, include site specific characteristics and operational matters (such

as procedures, testing and maintenance (T&M) schedule, refuelling outage strategy) and to allow for these aspects to be risk informed.

- The UK ABWR PSA is built on a number of assumptions based on the design documentation available at the time of the PSA development. It is important that adequate substantiation is provided when detailed information becomes available, or the PSA updated as appropriate.

My judgement is based upon the following factors:

- Detailed and in-depth technical assessment, on a sampling basis, of the full scope, level 1, 2 and 3 PSA developed in response to RI-ABWR-0002. The scope of my assessment encompassed all the technical areas of PSA following the guidance and structure established in Appendix 1 of ONR's PSA TAG.
- As well as the detailed review of all the technical areas of the PSA, I requested Hitachi-GE perform sensitivity analyses to evaluate the impact of my review findings on the risk. I have considered the adequacy of these analyses as part of my review. I have used some of the insights of these analyses in combination with qualitative arguments and quantitative information from the PSA to understand the potential risk significance of the findings of my review.
- Provision by Hitachi-GE of a PSA commitment log, detailing the shortfalls in the PSA model and documentation, identified as a result of my review, to be resolved beyond GDA.
- Review of the differences between the reference design reflected in the PSA and the final GDA reference design identified by Hitachi-GE, and their risk significance.
- Independent review of the PSA results to support the view that the UK ABWR risks are being managed ALARP as the UK ABWR design process continues through GDA and into the site specific phase. This included the results of an inspection towards the end of Step 4 at Hitachi-GE's offices to aid in drawing conclusions on use of the PSA to demonstrate that risks are being managed ALARP and the use of the PSA to identify ALARP design improvements.
- Detailed technical interactions on many occasions with the Hitachi-GE PSA team, along with my review of the responses to the RQs I raised during Step 4.

I consider that certain matters remain, which are for a future licensee to consider and take forward in its site specific safety submissions. These matters do not undermine the generic safety submission but require licensee input / decision at a specific site. These matters have been captured in eleven assessment findings.

To conclude, I am satisfied with the claims, arguments and evidence laid down within the PCSR and supporting documentation for PSA. I consider that, from a PSA view point, the Hitachi-GE UK ABWR is suitable for construction in the UK subject to future permissions and permits being secured.

LIST OF ABBREVIATIONS

ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ADS	Automatic Depressurisation System
ALARP	As Low As Reasonably Practicable
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without SCRAM
BOC	Break Outside Containment
BSL	Basic Safety Level
BSO	Basic Safety Objective
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
C&I	Control and Instrumentation
CCF	Common Cause Failure
CCS	Canister Cooling System
CDF	Core Damage Frequency
CET	Containment Event Tree
CO	Carbon Monoxide
COPS	Containment Overpressure Protection System
CRD	Control Rod Drive
CsI	Caesium Iodide
CST	Condensate Storage Tank
CUW	Reactor Water Clean-up System
DAC	Design Acceptance Confirmation
DAG	Diverse Alternative Generator
DBA	Design Basis Analysis
DC	Direct Current
DDI	Direct Debris Interaction
DF	Decontamination Factor
DRP	Design Reference Point
EA	The Environment Agency
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOC	Errors of Commission
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guideline
EPRI	Electrical Power Research Institute

EUR	European Utility Requirements
FCI	Fuel Coolant Interaction
FCVS	Filtered Containment Venting System
FDF	Fuel Damage Frequency
FDW	Feedwater System
FLSR	Flooding System of Reactor Building
FLSS	Flooding System of Specific Safety Facility
FMEA	Failure Modes and Effects Analysis
FPC	Fuel Pool Cooling and Clean-up System
FSF	Fundamental Safety Functions
FV	Fussell-Vesely
GDA	Generic Design Assessment
HBSC	Human-Based Safety Claim
HCLPF	High Confidence of Low Probability of Failure
HELB	High Energy Line Break
HEP	Human Error Probability
HF	Human Factors
HFE	Human Failure Event
HPCF	High Pressure Core Flooder System
HPIN	High Pressure Nitrogen Gas Supply System
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation and Air Conditioning System
HWBS	Hard Wired Back-up System
Hx/B	Heat Exchanger Building
IAEA	The International Atomic Energy Agency
IE	Initiating Event
IEAP	Internal Events At Power
IEF	Initiating Event Frequency
ISLOCA	Interfacing System LOCA
J-ABWR	Japanese ABWR
JNES	Japanese Nuclear Energy Safety Organisation
LDW	Lower Drywell
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-site Power
LPFL	Low Pressure Core Flooder System
LRF	Large Release Frequency
LUHS	Loss of Ultimate Heat Sink
MCCI	Molten Core Concrete Interaction

MCR	Main Control Room
MCS	Minimal Cutset
MDEP	Multi-national Design Evaluation Programme
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
MSQA	Management of Safety and Quality Assurance
MUWC	Makeup Water Condensate System
MUWP	Makeup Water Purified System
NPP	Nuclear Power Plant
NPSH	Net Positive Suction Head
NRW	Natural Resources Wales
NWL	Normal Water Level
OECD-NEA	Organisation for Economic Co-operation and Development - Nuclear Energy Agency
ONR	Office for Nuclear Regulation
PCS	Power Conversion System
PCSR	Pre-construction Safety Report
PCV	Primary Containment Vessel
Pd	Design Pressure
PDS	Plant Damage State
PGA	Peak Ground Acceleration
POS	Plant Operational State
PQC	Process Quality Control
PRA	Probabilistic Risk Analysis
PSA	Probabilistic Safety Analysis
PSF	Performance Shaping Factor
QA	Quality Assurance
R/A	Reactor Area
R/B	Reactor Building
RAW	Risk Achievement Worth
RBVS	Reactor Building Ventilation System
RC	Release Category
RCCV	Reinforced Concrete Containment Vessel
RCIC	Reactor Core Isolation Cooling System
RCS	Reactor Coolant System
RCW	Reactor Building Cooling Water System
RDCF	Reactor Depressurisation Control Facility

RGP	Relevant Good Practice
RHR	Residual Heat Removal System
RI	Regulatory Issue
RIP	Reactor Internal Pump
RO	Regulatory Observation
RP	Requesting Party
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
RSW	Reactor Building Service Water System
RUHS	Reserve Ultimate Heat Sink
RVI	Reactor Vessel Instrument System
S/P	Suppression Pool
SAA	Severe Accident Analysis
SAMG	Severe Accident Management Guideline
SAP	Safety Assessment Principle
SAuxP	Safety Auxiliary Panel
SBO	Station Blackout
SDC	Shutdown Cooling
SEL	Seismic Equipment List
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SLC	Standby Liquid Control System
SMA	Seismic Margins Analysis
SoDA	Statement of Design Acceptability
SPSA	Seismic PSA
SQEP	Suitably Qualified and Experienced Person
SRV	Safety Relief Valve
SSC	Structure, System, and Component
T&M	Testing and Maintenance
TAF	Top of Active Fuel
TAG	Technical Assessment Guide
TSC	Technical Support Contractor
U.S.	United States (of America)
UK ABWR	United Kingdom Advanced Boiling Water Reactor
V&V	Verification and Validation
V/B	Vacuum Breaker

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Annexes

1 INTRODUCTION

1. This assessment report details my Step 4 Generic Design Assessment (GDA) of Hitachi-GE's United Kingdom Advanced Boiling Water Reactor (UK ABWR) design in the area of probabilistic safety analysis (PSA).

1.1 GDA Background

2. Information on the GDA process is provided in a series of documents published on our website (www.onr.org.uk/new-reactors/index.htm). The outcome from the GDA process sought by Requesting Parties such as Hitachi-GE is a Design Acceptance Confirmation (DAC) for the Office for Nuclear Regulation (ONR) and a statement of design acceptability (SoDA) for the Environment Agency (EA) and Natural Resources Wales (NRW).

3. The GDA of the UK ABWR has followed a step-wise approach in a claims, arguments and evidence hierarchy; which commenced in 2013. Major technical interactions started in Step 2 with an examination of the main claims made by Hitachi-GE for the UK ABWR. In Step 3, the arguments which underpin those claims were examined. The reports in individual technical areas and accompanying summary reports are also published on ONR's website.

4. Hitachi-GE commenced GDA in 2013 and completed Step 4 in 2017. The Step 4 assessment is an in-depth assessment of the safety, security and environmental evidence. Through the review of information provided to ONR, the Step 4 process should confirm that Hitachi-GE:

   - has properly justified the higher level claims and arguments;
   - has progressed the resolution of issues identified during Step 3;
   - has provided sufficient detailed analysis to allow ONR to come to a judgment as to whether a DAC can be issued.

5. The full range of items that might form part of the assessment is provided in ONR's GDA Guidance to requesting parties (RPs) (www.onr.org.uk/new-reactors/ngn03.pdf). These include:

   - consideration of issues identified in Step 3;
   - judging the design against the Safety Assessment Principles (SAPs) (Ref. 1) and whether the proposed design reduces risks as low reasonably practicable (ALARP);
   - reviewing details of the Hitachi-GE design controls, procurement and quality control arrangements to secure compliance with the design intent;
   - establishing whether the system performance, safety classification, and reliability requirements are substantiated by the detailed engineering design;
   - assessing arrangements for ensuring and assuring that safety claims and assumptions are realised in the final as-built design;
   - resolution of identified nuclear safety and security issues, or identifying paths for resolution.

6. All of the regulatory issues (RIs) and regulatory observations (ROs) issued to Hitachi-GE during Steps 2 to 4 are also published on ONR's website, together with the corresponding Hitachi-GE resolution plan.

1.2 Scope

7. The scope of my assessment is detailed in the assessment plan (Ref. 2).

8. My Step 4 assessment of the UK ABWR PSA has looked in detail at all of the areas reviewed at a high-level in Step 3. Using, as the basis, the revised PSA model and documentation submitted by Hitachi-GE in Step 4, and all the additional information received in response to regulatory issue RI-ABWR-0002 'Probabilistic Safety Analysis: Project Plan and Delivery' (Ref. 3) and any ROs and regulatory queries (RQs) raised.

9. The scope of my assessment encompassed all the technical areas of PSA following the guidance and structure established in Appendix 1 of ONR's PSA Technical Assessment Guide (TAG) (Ref. 4). Not each and every fault tree, event tree, supporting analysis or item of reliability data has been examined in detail. However, I consider that the sample selected for the detailed review is representative of the PSA. I further consider that the sample is sufficiently large to ensure that my review could confirm whether the implementation of the methods and techniques used have been adequate. This is required to ensure that the serious regulatory shortfalls identified in GDA Step 3 that led ONR to raise RI-ABWR-0002 have been adequately addressed. Further information about the sampling approach is presented in Section 2.4.

10. In addition to the detailed review of all the technical areas of the PSA, I requested Hitachi-GE to perform sensitivity analyses to evaluate the impact of my review findings on the risk. These sensitivity analyses were provided in Refs 5, 6, 7 and 8. I have considered the adequacy of these analyses as part of my review. I have used some of the insights of these analyses, in combination with qualitative arguments and quantitative information from the PSA, to understand the potential risk significance of the findings of my review.

11. The versions of the PSA models and documentation submitted by Hitachi-GE for review during Step 4 are presented on the PSA document map (Ref. 9). This document was prepared by Hitachi-GE in response to RI-ABWR-0002 (Ref. 3) to enhance the clarity of the PSA submission. Therefore, this document has been the key reference that delineates the 'top layer' of the PSA models, documentation and supporting analyses reviewed during Step 4. In Step 4 Hitachi-GE has also submitted, or referred to, additional documents in response to ROs and RQs that I have raised. These are not all included on the PSA document map (Ref. 9), but they are referred to on a 'case by case' basis, in the description of the assessment of the different technical aspects of the PSA which are presented in Section 4.

12. Therefore, based on the above, the key references used for the Step 4 review of the UK ABWR PSA, which have constituted the scope of my assessment, are:

- PSA strategy document (Ref. 10);
- PSA summary report providing a collated picture of the global risk calculated by the various elements of the UK ABWR PSA (Ref. 11);
- internal events at power (IEAP) level 1 (Ref. 12) and level 2 PSA (Ref. 13);
- internal events spent fuel pool (SFP) level 1 (Ref. 14) and level 2 PSA (Ref. 15);
- internal events shutdown level 1 (Ref. 16) and level 2 PSA (Ref. 17);
- fuel route and dropped loads level 1 and level 2 PSA (Ref. 18);
- seismic level 1 and level 2 PSA for the reactor at power and the SFP, including a qualitative assessment of shutdown states (Ref. 19);

- hazards prioritisation and hazard PSA studies (Refs 20, 21, 22, 23 and 24);
- internal fire level 1 and level 2 PSA (Refs 25 and 26);
- internal flood level 1 and level 2 PSA (Refs 27 and 28);
- containment performance analysis (Ref. 29);
- consequence analysis for non-reactor faults and PSA success paths leading to radioactive release (Ref. 30);
- level 3 PSA (Ref. 31);
- PSA assumptions list (Ref. 32);
- sensitivity analyses (Refs 5, 6, 7, 8);
- methodologies produced for all the technical areas of the PSA (Ref. 33);
- task procedures / plans produced for the internal fire and flooding PSAs (Ref. 34);
- topic report on use of PSA in ALARP assessment (Ref. 35);
- chapter 25 of the pre-construction safety report (PCSR) (Ref. 36) which acts as a summary of the above references alongside the PSA summary report.

13. Hitachi-GE has revised the PSA models and documents several times during Step 4 to take into account outstanding peer review comments (which are part of Hitachi-GE's quality assurance (QA) arrangements), regulatory review comments and additional design information that has become available during GDA. Revised internal events PSAs were submitted in January 2016, and further revisions were developed in June 2016. Further documentation updates to the internal events PSAs were submitted between January 2017 and June 2017. My review has considered the totality of the revisions of the internal events PSA submissions, but the main focus has been on the January 2016 and June 2016 revisions. Internal hazards PSAs were submitted in August 2016, with a further revision in March 2017 to address regulatory review comments and include analysis for the shutdown plant operational states (POSs) and SFP. In addition, prior to the end of Step 4, Hitachi-GE undertook further refinement of the internal hazard PSAs, removing conservatisms and taking credit for additional mitigating and protective measures. This is referred to in this assessment report as the 'internal hazards PSA refinement'.

1.3 Method

14. My assessment complies with internal guidance on the mechanics of assessment within ONR (Ref. 37).

2 ASSESSMENT STRATEGY

2.1 Standards and Criteria

15. The standards and criteria adopted within this assessment are principally the safety assessment principles (SAPs) (Ref. 1), internal TAGs (Refs 4, 38, 39, 40), relevant national standards, international standards, and relevant good practice (RGP) as described below.

2.1.1 Safety Assessment Principles

16. The key SAPs applied within the assessment are presented in Annex 1.

2.1.2 Technical Assessment Guides

17. The TAGs that have been used as part of this assessment are presented in Annex 2.

2.1.3 National and international standards and guidance

18. The international standards and guidance that have been used as part of this assessment are presented in Annex 3.

2.2 Use of Technical Support Contractors (TSCs)

19. It is usual in GDA for ONR to use technical support contractors (TSCs), for a variety of reasons; for example: to provide additional capacity, to enable access to independent advice and experience, analysis techniques and models, to enable ONR's inspectors to focus on regulatory decision making, etc.

20. To supplement ONR's internal capability, one contract was placed with Corporate Risk Associates (CRA) Ltd. for a part-time PSA specialist to work as an integral part of GDA Step 4 assessment team under my supervision.

21. Independent of this contract for embedded resource; Table 1 sets out the broad areas in which TSCs were used for the GDA Step 4 assessment of the UK ABWR PSA. This support was commissioned to share the detailed technical review workload, provide high quality expertise for the broad range of specialised and diverse technical subjects needed for a full scope PSA, assist in the production of RQs to Hitachi-GE and review the responses, and to provide support at technical meetings with Hitachi-GE. The TSCs were chosen based on a competitive tendering process.

Table 1: Technical Support Contractors

Technical Support Contractor	Scope of the Work
Jensen Hughes	In-depth technical reviews, on a sampling basis, of Hitachi-GE's UK ABWR PSA model, data and supporting analyses (except for review of the human reliability analysis (HRA) which has been led by ONR's human factors (HF) inspector). Evaluation of the risk importance of the findings in the various PSA technical areas. The review included: - Level 1 PSA for internal initiating events during

Table 1: Technical Support Contractors

Technical Support Contractor	Scope of the Work
	operation at power (IEAP). - Low power and shutdown PSA. - Spent fuel pool (SFP) and fuel route PSA. - Prioritisation of hazards. - Internal hazards PSA. - External hazards PSA. - Level 2 PSA. - Interface between level 2 PSA and the level 3 PSA. - Overall risk evaluation: quantification, sensitivity and uncertainty analyses, and interpretation of the PSA results.
ABS Consulting Ltd.	Detailed review of the seismic fragility methodology and results used in the seismic PSA (SPSA). Review of the characterisation of external hazard frequencies and hazard magnitudes in the prioritisation of external hazards and external hazards PSA.
AMEC	Review of the containment structural analysis for the level 2 PSA and the supporting structural analysis for dropped loads PSA. The review of the use of AUTODYN code to address fuel coolant interaction (FCI) impact on the containment.

2.3 Integration with Other Assessment Topics

22. GDA requires the submission of an adequate, coherent and holistic generic safety case. Regulatory assessment cannot therefore be carried out in isolation as there are often safety issues of a multi-topic or cross-cutting nature. The following cross-cutting issues have been considered within this assessment:

- Human factors (HF): HF provides input to the PSA's human reliability analysis (HRA). The assessment of HRA has been led by ONR's HF inspector. In addition, the PSA provides input to the identification of the human-based safety claims (HBSCs), human failure events (HFEs) and evaluation of their importance to overall risk.
- Fault studies: ONR's fault studies assessment has provided input to the PSA assessment in the following areas:
    - The assessment of the level 1 PSA success criteria. This work has been led by the PSA team in coordination with ONR's fault studies inspector.
    - The review of the adequacy of computer codes and input used to support the PSA success criteria analyses (for example code validations, experience of the code analysts etc). This work has been led by ONR's fault studies inspector, and is reported in Ref. 41.
    - The assessment of the non-reactor faults PSA, including the completeness of the list of initiating events and the assumptions made

for the consequence analyses. This review has been led by my PSA team.

- Severe accident analysis (SAA): this has provided input to the assessment of the level 2 PSA. This work has been led by ONR's severe accidents inspector in coordination with my PSA team and with input from ONR's reactor chemistry inspector (regarding for example composition of radioactive releases and behaviour of radioisotopes, aerosols). The work led by ONR's severe accidents inspector has included review of the adequacy of the computer codes used and confirmatory analyses of some severe accident scenarios. This has provided useful input into the PSA assessment, for example regarding the applicability of the codes and the nodalisation used for the SAA to support the level 2 PSA. Further details are provided in Ref. 42.
- Structural integrity: this provides input to the PSA assessment in the following areas:
  - The containment structural analysis (drywell head and flange) for the level 2 PSA. This piece of work has been undertaken by a TSC and led by ONR's civil engineering inspector and myself as part of the assessment of the integrity of the containment.
  - The external hazards PSA (regarding fragilities of metal components). This piece of work has been led by my PSA team in coordination with ONR's external hazards inspector.
  - Dropped loads PSA structural integrity supporting analyses. This piece of work has been undertaken by a TSC and led by my PSA team.
- Civil engineering / external hazards: this provides input to the PSA assessment in the following areas:
  - The external hazards PSA regarding definition of hazards' magnitudes and frequencies, and fragilities of structures. This assessment task has been led by ONR's external hazards inspector in coordination with my PSA team.
  - Dropped loads PSA, as noted above.
- Consequence analysis and radiological protection: this provided input to my assessment of the level 3 PSA. This work was led by ONR's level 3 PSA inspector.
- Control and instrumentation (C&I): PSA plays a key role in the design of these complex systems and their central role in the safety of the UK ABWR. The assessment of the C&I model in the PSA has been led by my PSA team with input from ONR's C&I inspector. My PSA team has also provided input to the C&I review regarding claims, failure modes and evaluation of their importance to overall risk.
- Management of safety and quality assurance (MSQA): ONR's MSQA inspector has supported some of the inspections of the PSA processes used to support the development of the UK ABWR PSA and PSA applications (including the use of the PSA as part of the design process, the process to capture PSA assumptions and to review the assumptions when further information becomes available, and Hitachi-GE's quality assurance (QA) processes as applied to the development of the PSA).
- Internal hazards: The review of the internal hazards PSA and prioritisation has been led by my PSA team with input from the ONR internal hazards and HF inspectors to ensure that the PSA assumptions are aligned with the design and operational procedures.

23. In addition to the above, there were continual two-way interactions between PSA and the rest of the technical areas throughout the Step 4 assessment.

2.4 Sampling Strategy

24. It is seldom possible, or necessary, to assess a safety case in its entirety, therefore sampling is used to limit the areas scrutinised, and to improve the overall efficiency of the assessment process. Sampling is done in a focused, targeted and structured manner with a view to revealing any topic-specific, or generic, weaknesses in the safety case.

25. I considered that a proportionate sampling strategy for this assessment would be achieved as long as it included a review of each of the main technical areas considered essential to produce a full scope PSA. A good understanding of each main technical area of the PSA would then enable an overall view to be established and an overall judgement regarding the adequacy of the PSA to be made. The technical areas I sampled were selected using ONR's PSA TAG (Ref. 4) for guidance.

26. For each technical area of the PSA, a representative sample of fault trees, event trees, supporting analysis and reliability data was identified to be examined in detail, including a consideration of the following:

- Shortfalls identified in Step 3. In particular regulatory observations RO-ABWR-0040 (Ref. 43), RO-ABWR-0041 (Ref. 44), RO-ABWR-0042 (Ref. 45), RO-ABWR-0046 (Ref. 46), RO-ABWR-0048 (Ref. 47) and RO-ABWR-0053 (Ref. 48) and RQs RQ-ABWR-0559 (Ref. 49) and RQ-ABWR-0560 (Ref. 50); which are all key references to RI-ABWR-0002 (Ref. 3);
- the UK ABWR risk profile and importance measures;
- coverage of all types of systems, structures and components (SSCs) and accident sequences.

Further information is provided in Section 4.

2.5 Out of Scope Items

27. No areas of the PSA have been left out of the scope of the PSA review in GDA.

28. However, as discussed in the relevant sections in this report, the detailed review of the HRA has been undertaken by ONR's human factors assessment team.

29. Similarly, my assessment of closure of RI-ABWR-0002 has been documented in a separate assessment report (Ref. 51).

30. In addition, my review did not cover, in detail, the following PSA-related technical aspects:

- Verification and validation (V&V) of the various computer codes used to support the PSA (Section 4.2.3 provides details of assessments performed by other inspection teams on these codes).
- Assessment against numerical target NT.2 'Time at risk' of the SAPs (Ref. 1) during operation at power. A supplementary PSA study to properly address compliance with this target has not been presented by Hitachi-GE, but the PSA results and supporting ALARP assessments provide confidence that, with good management of testing and maintenance (T&M), the future licensees will be able to meet NT.2.

2.6 Residual Matters

31. Throughout the Step 4 assessment of the PSA a number of shortfalls or residual matters have been identified (hereafter referred to solely as shortfalls). Shortfalls have been discussed with Hitachi-GE as appropriate throughout Step 4, and have often been resolved by PSA model and documentation updates during GDA. Shortfalls which remain outstanding at the end of GDA are recorded in this assessment report in the 'Findings' sections within Section 4.2.

32. Shortfalls which meet the criteria set out below and are seen as significant are recorded as assessment findings. Assessment findings are shortfalls which do not undermine the generic safety submission but must be addressed by any future licensee, with progress monitored by the regulator. Shortfalls recorded as assessment findings are generally the most significant and are not considered part of 'normal business'.

33. The criteria for an assessment finding, also discussed in Section 4.7, are if one of the following applies:

- site specific information is required to resolve this matter;
- resolving this matter depends on licensee design choices;
- the matter raised is related to operator specific features / aspects / choices;
- the resolution of this matter requires licensee choices on organisational matters; or
- to resolve this matter the plant needs to be at some stage of construction / commissioning.

34. Shortfalls which meet the criteria set out below are recorded as minor shortfalls. These shortfalls are generally related to the PSA documentation and processes and are expected to be addressed by any future licensee to ensure that the PSA documentation is fit for purpose. These shortfalls do not undermine the generic safety submission and are expected to be addressed by any future licensee without progress being directly monitored by the regulator.

35. The criteria for a minor shortfall, also discussed in Section 4.8, are if none of the following apply:

- undermine ONR's confidence in the safety of the generic design;
- impair ONR's ability to understand the risks associated with the generic design;
- require design modifications;
- require further substantiation to be undertaken.

36. Shortfalls and other future PSA development needs which do not meet the criteria for assessment findings, or are not significant enough to warrant the increased regulatory focus of assessment findings, and are not related to PSA documentation or processes or meet the criteria of a minor shortfall, but are still expected to be addressed by any future licensee to ensure that the PSA is able to be used as expected in the site specific phase are recorded throughout this report. As part of resolution of AF-UKABWR-PSA-001 any future licensee must review the shortfalls identified in the report and provide a programme for addressing them. Annex 7 is included in this assessment report as a summary to aid in resolution of AF-UKABWR-PSA-001. These shortfalls do not undermine the generic safety submission and are expected to be addressed by any future licensee without progress being directly monitored by the regulator. Some of the shortfalls identified may be considered part of 'normal business' for development of the PSA, however they are identified in this report to ensure that

any future licensee is aware of the development needs and produces a programme to address them.

3 REQUESTING PARTY'S SAFETY CASE

37. Hitachi-GE has produced a pre-construction safety report (PCSR) chapter on PSA (Ref. 36) which provides a summary and route map for the PSA and supporting information submitted during GDA.

38. Hitachi-GE has produced and submitted a PSA in response to RI-ABWR-0002 (Ref. 3) which, in general, covers level 1, level 2 and level 3 PSA with consequence analyses developed for core damage and non-core damage sequences leading to a release.

39. The Hitachi-GE document map (Ref. 9) provides the list of submissions which form the PSA documentation suite; the key references have been listed in Section 1.2. The PSA submission includes:

- PSA strategy document (Ref. 10) that includes a high-level description of the PSA objectives and applications and procedures related to the development and applications of the PSA, including the following:
    - Process quality control (PQC) procedure (Ref. 52) used to procure the design information that is used to initially develop the PSA and other inputs from the design team. Hitachi-GE states that design changes are subsequently communicated to the PSA team via the PQC procedure (and ultimately the PSA and assumptions are updated to reflect design changes).
    - PSA QA arrangements (Ref.52), including a peer review process conducted by external PSA experts (Ref. 53).
    - A process to capture, track and transfer PSA assumptions to the site specific phase has also been developed by Hitachi-GE (Ref. 32).
    - Hitachi-GE states that the PSA is used to support design changes categorised as requiring to enter the six step process for design change in Step 4. Hitachi-GE states that this was part of the 'Generic Design Development Control' process (Ref. 54); the PSA is used to support the decision making in design review meetings via the participation of expert(s) having experience or knowledge of PSA. The 'Generic Design Development Control' process has been updated in GDA to reflect that the Hitachi-GE PSA team can also identify design enhancements based on PSA results.

- A PSA summary report (Ref. 11). The PSA summary report presents the purpose and scope of the PSA, an overview of the results, methods and conclusions of the study. The PSA summary report also provides an overview of the contents and organisation of the documentation for each PSA task. The PSA summary report has been used as the basis of the PCSR chapter on PSA. In the PSA summary report Hitachi-GE concluded that the UK ABWR design in GDA has no further reasonably practicable measures which could be implemented to reduce risks. Hitachi-GE also notes and that the PSA has been instrumental in various UK ABWR design improvements.

- An internal events PSA for the reactor at power and shutdown operating modes, fuel route operations, spent fuel pool and consideration of other non-reactor facilities (Refs 12, 13, 14, 15, 16, 17, 18 and 30).
- A prioritisation of hazards (Refs 20 and 21) for the reactor and non-reactor facilities, including consideration of combined hazards. When hazards are considered important in terms of risk, more detailed studies are provided. On this basis, PSA studies for accidental aircraft impact, tornado missile and turbine missile events were conducted (Refs 22 and 24). Sensitivity analyses

were also undertaken to investigate the risk impact of external flooding and biological fouling events (Ref. 23).

- Internal hazards PSA, covering internal fire and internal flooding for the reactor at power (Refs 25 and 27). A simplified quantitative analysis was also developed to assess the risk for the shutdown POSs and SFP due to internal fire and internal flooding events.
- Seismic events PSA (SPSA) for the reactor at power and SFP (Ref. 19). A simplified quantitative analysis was also developed to assess the risk for shutdown POSs due to seismic events.

40. Hitachi-GE internal fire and flooding at power PSA submissions follow the American Society of Mechanical Engineers (ASME) / American Nuclear Society (ANS) standard tasks prescribed to develop the technical elements of an internal fire and flooding PSA. For the internal fire PSA this included the following tasks and documents:

- Tasks 1 to 4 (Refs 55, 56 and 57). The output of these tasks defines the scope of plant buildings for the generic design, the partitioning between fire compartments, the physical location of the safety equipment credited in the PSA and the cable routing within the plant.
- Task 5 (Ref. 58). This task develops the internal fire at power PSA from the internal events at power PSA.
- Tasks 6 and 7 (Refs 59 and 60). These two tasks develop the fire frequencies for the plant compartments and screen fire scenarios based on their risk contribution.
- Tasks 8, 9, 10 and 11 (Refs 61, 57, 62, 63, 64 and 65). These tasks utilise detailed fire modelling tools to determine the potential for fire growth, the response of safety equipment to fires, the failure modes of cables, the impact of fires in the main control room (MCR), the fire performance of barriers between compartments and the impact of fires on unprotected structural steel. The aim of these tasks was ultimately to define more realistic fire scenarios for quantification in the internal fire PSA model.
- Task 12 (Ref. 66). This task evaluates the response of the operators to fire scenarios.
- Task 13 (Ref. 67). The output of this task is an assessment of the potential for seismic events to initiate fires and to fail fire detection and suppression systems.
- Tasks 14 and 15 (Refs 25, 68). These tasks quantify the core damage frequency (CDF) and large release frequency (LRF) due to internal fires. Hitachi-GE also assessed uncertainties for the internal fire PSA and undertook sensitivity analyses.

41. For the internal flooding PSA this included the following tasks and documents:

- Tasks 1 to 4 (Ref. 69). The output of these tasks is the list of flood areas, flood sources, qualitative screening, and the corresponding flood scenarios that will be taken forward for detailed quantitative analysis.
- Tasks 5 to 7 (Ref. 70). The output of these tasks is characterisation of flood scenarios, flood initiating events analysis, flood consequence analysis, and the internal flood frequencies.
- Task 8 (Ref. 71). The output of this task is the human reliability analysis to the mitigation of flooding events.

- Tasks 9 and 10 (Ref. 27). The output of these tasks is the presentation of the PSA quantification. This includes the CDF, the (LRF), sensitivity and uncertainty analyses.

42. Prior to the end of Step 4 Hitachi-GE undertook further refinement of the internal hazards PSA; removing conservatisms and taking credit for additional mitigating and protective measures. This is referred in this assessment report as 'internal hazards PSA refinement' (Refs 26 and 28).

43. The PSA is modelled and quantified using the Electric Power Research Institute (EPRI) Risk and Reliability Workstation software suite for both the level 1 and level 2 PSA with the identified versions:

- CAFTA version 6.0b – logic model development program
- PRAQuant version 5.2 – event tree sequence quantification program
- FTREX version 1.8 – PSA model quantification engine
- UNCERT version 4.0 – uncertainty analysis program
- FRANX version 4.3 – hazard PSA analysis program

44. This suite of software has been developed by EPRI and is widely used for the construction and evaluation of PSAs.

45. Hitachi-GE states that the methods and data used in the PSA are well known and aligned with the latest international good practices. In response to RI-ABWR-0002, Hitachi-GE established a peer review process conducted by external PSA experts. The peer review process was a major activity within GDA. Hitachi-GE claims that the peer review follows the requirements of the US ANS/ASME probabilistic risk analysis (PRA) standard (Ref. 72) and included consideration of the PSA TAG (Ref. 4). The peer review process and objectives are explained in Ref. 52. Hitachi-GE provides visibility of the review's scope and outcomes through peer review documents for each technical area of the PSA (examples in Ref. 73)

46. Hitachi-GE also produced a topic report on 'Use of PSA in ALARP assessment' (Ref. 35) in response to RO-ABWR-0076. Hitachi-GE states that this report provides evidence from the PSA that the UK ABWR design follows the principles of ALARP and that the PSA has been used to identify any areas where further risk reduction may be practicable as Step 4 activities are completed or during the detailed design and plant operation which follow the completion of GDA.

47. Table 2 presents a summary of the PSA results, as reported in the PSA Summary Report (Ref. 11).

Table 2: UK ABWR PSA Results (Ref. 11)

Item	UK ABWR Results (/yr)		
	Core Damage Frequency	Large Release Frequency	Frequency of 100 fatalities*
Internal events at power	2.3×10^{-07}	4.6×10^{-08}	6.5×10^{-08}
Internal events during shutdown POS	8.7×10^{-08}	6.9×10^{-08}	7.0×10^{-08}
Internal events for spent fuel pool	4.2×10^{-07}	4.8×10^{-08}	4.8×10^{-08}

Table 2: UK ABWR PSA Results (Ref. 11)

Item	UK ABWR Results (/yr)		
	Core Damage Frequency	Large Release Frequency	Frequency of 100 fatalities*
Internal fire events at power	Initial:1.9×10^{-06} Refined:5.0×10^{-07}	Initial:1.6×10^{-06} Refined:2.7×10^{-07}	Initial:$2.4 \times 10^{-06\dagger}$ Refined:3.1×10^{-07}
Internal flood events at power	Initial:1.8×10^{-06} Refined:1.8×10^{-06}	Initial:7.8×10^{-07} Refined:1.8×10^{-07}	Initial:$7.9 \times 10^{-07\dagger}$ Refined:5.8×10^{-07}
Seismic events at power	7.3×10^{-07}	6.1×10^{-07}	6.5×10^{-07}
Seismic events during shutdown POS	4.2×10^{-08}	Not calculated	Not calculated
Seismic events for spent fuel pool	4.5×10^{-07}	3.9×10^{-07}	3.9×10^{-07}
Tornado missile events	5.2×10^{-10}	2.4×10^{-10}	Not calculated
Turbine missile events	7.1×10^{-10}	8.1×10^{-11}	Not calculated
Accidental aircraft impact	7.9×10^{-10}	4.6×10^{-10}	Not calculated
Total (including refined internal hazards)	4.3×10^{-06}	1.6×10^{-06}	2.1×10^{-06}
Off-site dose 0.1-1mSv	1.4×10^{-03}		
Off-site dose 1-10mSv	2.4×10^{-04}		
Off-site dose 10-100mSv	6.2×10^{-06}		
Off-site dose 100-1000mSv	1.5×10^{-06}		
Off-site dose >1000mSv	2.3×10^{-06}		
Individual Risk to people off the site	2.1×10^{-07}		

*Frequency of 100 fatalities is related to SAP Target 9. The difference in LRF and the frequency of 100 fatalities is mainly due to some release categories which result in greater than 100 fatalities not being categorised as a large release by Hitachi-GE.

†Taken from Ref. 74, noting that release category definitions were updated between the initial and refined internal hazard PSAs.

4 ONR STEP 4 ASSESSMENT

48. This assessment has been carried out in accordance with ONR internal guidance on the 'Purpose and Scope of Permissioning' (Ref. 75).

4.1 Scope of Assessment Undertaken

49. The scope of the assessment carried out in Step 4 has followed the strategy described in Section 2 of this report and has been undertaken with the assistance of TSCs who have carried out their work under my direction and supervision.

50. For each of the relevant 'assessment expectations' in the tables presented in Appendix 1 of ONR's PSA TAG (Ref. 4), a view on the adequacy, or otherwise, of the submitted documentation, including any appropriate RQ and RO responses, has been taken. In cases where limitations and/or potential findings have emerged there has been dialogue with Hitachi-GE in an effort to resolve the shortfall or identifying if further information could be provided within the GDA timeframe.

51. My Step 3 PSA assessment identified shortfalls in all the technical areas of the PSA. The outcomes of my assessment were captured in a series of related ROs (Refs 43, 44, 45, 46, 47 and 48) and Regulatory Queries (RQs) (Refs 49 and 50). These ROs and RQs highlighted that the arguments supporting the PSA safety claims did not meet the relevant expectations in ONR's PSA TAG (Ref. 4), which captures ONR's PSA SAPs (Ref. 1) and international good practice. On this basis, ONR considered that the PSA submission did not meet the expectations defined in the Step 2 PSA assessment report (Ref. 76), did not provide a clear understanding of the UK ABWR risk and was not adequate to support the demonstration that the level of risk was ALARP.

52. The review also identified issues with Hitachi-GE's PSA capability and the PSA quality assurance (QA). Furthermore, the documentation provided in Step 3 was incomplete and not coherently structured. On the basis of the review outcomes, I did not have confidence that Hitachi-GE, without significant improvements, would be able to deliver a modern standards full-scope PSA for the UK ABWR, which was suitable and sufficient for ONR to carry out a meaningful assessment in Step 4, and be able to judge during GDA whether the overall risks from the UK ABWR are acceptable. The general position was that the PSA assessment was not ready to move to Step 4 with an examination of the evidence which supports the claims and arguments presented to date. This was considered to be a serious regulatory shortfall and escalated to a regulatory issue (RI) (Ref. 3) in July 2015 (RI-ABWR-0002).

53. In response to RI-ABWR-0002 (Ref. 3), Hitachi-GE established a significantly revised programme, extended their PSA capability and improved the processes to support the development and use of the PSA. The first milestone of the programme was the delivery of the updated level 1 IEAP PSA during the extended Step 3 period at the end of September 2015.

54. At the end of Step 3 I reviewed the updated level 1 IEAP PSA to determine whether it was suitable for ONR to commence Step 4 detailed assessment. My review is documented in Ref. 77. My review highlighted that Hitachi-GE's improved PSA arrangements and PSA capability had established a basis to develop and deliver the PSA information that I required for a meaningful Step 4 assessment. In Step 4, I reviewed the totality of Hitachi-GE's response to RI-ABWR-0002 and confirmed its adequacy for ONR to carry out a meaningful assessment in GDA. The outcome of my assessment, which supported closure of RI-ABWR-0002 is documented in Ref. 51.

55. As indicated previously, the scope of my Step 4 assessment has considered whether the shortfalls identified in Step 3 (Refs 77 and 78) have been addressed. A summary of

the key shortfalls identified in Step 3 is therefore provided for each of the technical areas discussed in Section 4.2.

56. My review has also included several inspections of the implementation of Hitachi-GE's key processes as outlined in their PSA strategy (Ref. 10) and use of PSA to support ALARP (Ref. 35) documents, including QA arrangements, the process followed to ensure the PSA reflects the UK ABWR design reference point (DRP), the process followed to use the PSA to support the design development, the process followed to track and transfer PSA assumptions and commitments to the site specific phase. The outcomes of these interactions are recorded in Ref. 79 and are referred to in several sections of this report.

57. Details of my assessment of the revised UK ABWR PSA submitted to ONR in Step 4, including my conclusions and findings are presented in the following sections.

58. To help traceability, the section number (A1.i.i) of the table of assessment expectations in Appendix 1 of ONR's PSA TAG (Ref. 4) is included, where applicable.

4.2 Assessment

4.2.1 General Expectations – Approaches and Methodologies (A1-1.1), PSA Scope (A1-1.2), Freeze Date (A1-1.3)

4.2.1.1 ASSESSMENT

59. In response to RI-ABWR-0002, Hitachi-GE updated the PSA to reflect more modern approaches. Task procedures were produced for the internal events and internal fire and flooding PSAs (Ref. 10). Overall, the methods and data used in the UK ABWR PSA submitted to ONR in Step 4 are well known and, in general, aligned with international good practice. This is discussed in more detail for each individual technical area of the PSA throughout this report.

60. In response to RI-ABWR-0002, Hitachi-GE also extended the scope of the PSA. My Step 3 review (Ref. 77) identified that the scope of the PSA was insufficient to understand the risk associated with the UK ABWR. In particular the following were highlighted:

- the treatment of fuel route handling operations was not clear;
- there were missing initiating events in the shutdown and SFP PSAs;
- the scope of the internal fire and flooding PSAs was limited to the reactor at power;
- a seismic margin analysis (SMA) methodology was constructed around reactor at power events (Hitachi-GE extended this analysis in Step 4 to a seismic PSA (SPSA) for all reactor states and SFP);
- non-reactor faults had not been considered. During Step 3, the ONR fault studies inspector issued RO-ABWR-0037 (Ref. 80) requesting deterministic and probabilistic assessments of non-reactor faults for the UK ABWR; and
- the internal fire and flooding PSA methodologies did not provide specific information on how the level 2 PSA for fire and flood events was going to be developed.

61. As part of my assessment for closure of RI-ABWR-0002, I reviewed the following:

- Hitachi-GE's QA plan and process; including: Hitachi-GE's peer review process (Ref. 53), implementation against ONR's PSA TAG (Ref. 4) and relevant good practice (in particular the International Atomic Energy Agency (IAEA) guidance (Ref. 81)).

- The completeness of the scope of the overall UK ABWR PSA submission against ONR's expectations identified in SAP FA.12 (Ref. 1) and ONR's PSA TAG (Ref.4.).
- Whether the PSA adequately reflects the design reference and this is clearly documented in line with the expectations of ONR's PSA TAG (Ref.4) and the relevant SAPs (Ref. 1).

62. This review, documented in Ref. 51, identified areas that required regulatory follow-up in GDA. I undertook an inspection in Japan in May 2017 to gather sufficient information to complete the assessment of these topics (Ref. 79). The outcomes from this inspection are reported in the following sub-sections.

63. No further assessment has been undertaken in terms of appraisal of the PSA scope in general. Specific comments are reported in some of the technical areas of the PSA in the following report sub-sections. Overall, the strengths and findings from Ref. 51 remain valid. These are summarised below.

64. My assessment was performed using the latest available models and documentation provided by Hitachi-GE, which are described in Sections 2 and 3 of this report. My assessment was carried out against the expectations in ONR's PSA TAG (Ref. 4).

4.2.1.2 STRENGTHS

65. The scope of Hitachi-GE's PSA submission in response to RI-ABWR-0002 is comprehensive; it includes the UK ABWR internal events PSA for the reactor at power and shutdown operating modes, fuel route operations, spent fuel pool and consideration of other non-reactor facilities. The PSA also covers internal fire and internal flooding events for the reactor at power, seismic events for the reactor and the spent fuel pool; simplified quantitative analyses have also been developed to assess the risk of the reactor shutdown operating states and SFP due to internal fire, internal flooding and seismic events.

66. A prioritisation of hazards has been developed for the reactor and non-reactor facilities, including consideration of combinations of hazards; when hazards are considered important in terms of risk, more detailed studies were provided. Sensitivity analyses were also undertaken to investigate the risk impact of external flooding and biological fouling events.

67. The PSA has, in general, covered level 1, level 2 and level 3. Consequence analyses were also developed for non-core damage sequences which led to a release.

4.2.1.3 FINDINGS

68. My review in Ref. 51 identified that Hitachi-GE's QA plan and processes would need improvements, if used by future licensees, to take into account lessons learned during GDA and Hitachi-GE's peer review process (follow-up item 4 in Ref. 51). This has been captured as a minor shortfall to ensure visibility to any future licensee.

69. The PSA submission did not provide sufficient visibility regarding outstanding peer review comments and how they will be addressed in the future (follow-up item 5 in Ref. 51). Hitachi-GE clarified that most of the peer review comments have been addressed in GDA. Any remaining comments are judged by Hitachi-GE to have a small impact on the conclusions of the PSA. These are captured together with other shortfalls and some PSA developments needed for the site specific PSA in Hitachi-GE's commitment log (Ref. 82). My inspection of the commitment log in May 2017 (Ref. 79) provided me with confidence that Hitachi-GE has put in place a robust process to capture the areas that need further work beyond GDA. Furthermore, for most of the cases, and upon ONR request, Hitachi-GE has provided sensitivity

analyses to assess the risk impact of the shortfalls and a clear link to these analyses is provided in the commitment log. The commitment log is part of the GDA PSA submission and should be transferred to the future licensee and the commitments addressed in future PSA revisions.

70. Some evidence of the implementation of Hitachi-GE's process to maintain and update the PSA to reflect design changes was shared with ONR in 2016. However, my review identified that the information provided was high-level and incomplete. Further information was needed to justify the criteria used to categorise modifications and decide whether the PSA needed to be updated in GDA. There was also a lack of information regarding how the cumulative impact on the risk of several modifications was considered. I captured these shortfalls in Ref. 51 (follow-up item 1). Upon request, Hitachi-GE provided further information regarding the criteria used in response to RQ-ABWR-1161 and made available to ONR the records of how the modifications had been categorised during the May 2017 inspection. This information is captured in a database (Ref. 83), including the assessment of the aggregated impact on the risk of minor modifications. On the basis of this information and the outcomes of the May 2017 inspection, I am confident that, although the PSA does not exactly reflect the UK ABWR GDA design reference, Hitachi-GE has identified the gap and recorded the design changes that will need to be included in future revisions of the UK ABWR PSA model. On the basis of the information provided by Hitachi-GE, I expect that the impact of these design changes on the risk profile will not be significant. The list of design modifications that are not reflected in the PSA is part of the GDA PSA submission and should be transferred to a future licensee and considered in future revisions of the PSA.

71. My inspection of Hitachi-GE's processes also noted that there is no systematic process for identifying updates to supporting analysis or other supporting information to the PSA in the same way as for design modifications. As there are 'gaps' between the PSA and the final GDA position in a number of areas (eg HRA) it is considered important that these updates are identified and scheduled for inclusion into the PSA in a similar manner to design changes. It is noted that, in some cases, these have been captured as part of the assumptions list (see Section 4.2.2) or the commitment log (Ref. 82). However, it is important that the future licensee undertakes a holistic and systematic review of the status of the PSA supporting references and updates the PSA as required.

Assessment Finding AF-UKABWR-PSA-001 (Part 1)*: The licensee shall:*

1. *Develop processes and procedures to ensure that the PSA is kept living and is aligned with the design reference. Implementation of this process should ensure that differences between the PSA and the final GDA design reference are adequately addressed.*

72. Throughout this assessment report a number of shortfalls and future development needs are identified in each section, as discussed in Section 2.6. Some specific aspects are identified as assessment findings and minor shortfalls; however it is expected that any future licensee should review all of the shortfalls and development needs identified in this assessment report and address them as appropriate.

Assessment Finding AF-UKABWR-PSA-001 (Part 2): *The licensee shall:*

2. *Develop an overall programme which ensures that the shortfalls and future PSA development needs presented in this assessment report (summarised in Annex 7) are included in the plans for the site specific PSA, such that risk insights are able to be identified and utilised to inform associated design and operational decision making.*

4.2.1.4 CONCLUSION

73. The overall scope of the UK ABWR PSA is sufficient to support the UK ABWR 'generic' PCSR and to reflect the design reference; differences between the UK ABWR GDA design reference and the PSA design reference are clear and are not considered significant in terms of risk.

74. Some limitations have been identified and are discussed in the following sections. However, their impact on the risk profile is understood on the basis of a number of sensitivity analyses provided by Hitachi-GE upon ONR request. These will need to be addressed as part of the development of the site specific PSA.

4.2.2 General Expectations – Assumptions in the PSA (A1-1.5)

4.2.2.1 ASSESSMENT

75. My review has considered Hitachi-GE's database that captures PSA assumptions including assumptions related to the design, procedures, limits and conditions, etc.

76. My Step 3 review identified a lack of completeness in the assumptions explicitly captured in the PSA documentation and in general a lack of visibility regarding what assumptions were captured and why.

77. In response to my review comments, Hitachi-GE provided, in Step 4, a document that collates key assumptions in the PSA (Ref. 32). I have reviewed this document against the expectations in ONR's PSA TAG (Ref.4).

78. As part of the inspection I undertook in May 2017, I assessed Hitachi-GE's process to review its assumptions against latest available design and operational information and how the relevant assumptions, documentation and models are updated accordingly.

79. In addition, my review of the different technical areas of the PSA has also identified shortfalls related to the use of assumptions. These are reported in the relevant sections of this assessment report.

80. My assessment was performed using the latest available models and documentation provided by Hitachi-GE, which are described in Sections 2 and 3 of this report. My assessment was carried out against the expectations in ONR's PSA TAG (Ref. 4).

4.2.2.2 STRENGTHS

81. Hitachi-GE has developed a process to capture PSA assumptions in a single location. This process has enabled a review of design (and other) assumptions during GDA and will enable their transfer to the site specific stage in cases where there is insufficient information in GDA.

4.2.2.3 FINDINGS

82. Hitachi-GE has collated an extensive list of assumptions used in the PSA (Ref. 32). My review identified that some of the PSA modelling makes assumptions based on the assumed 'as-built' plant design and operation. I questioned the basis of a number of PSA assumptions and specific shortfalls, details of which are reported in the relevant sections in this assessment report. Some examples are provided below to illustrate the type of shortfalls identified:

- The internal fire PSA and the internal fire PSA refinement, due to a lack of information available during GDA, rely on many assumptions such as cable routing and back-up building barriers (see Section 4.2.11.3). These assumptions have resulted in a reduction in the risk and therefore it is important that the assumed design features are substantiated and reflected in the detailed design.

- The sensitivity to the PSA assumption of failure of emergency core cooling system (ECCS) due to containment failure shows that the large release frequency (LRF) could be significantly reduced if ECCS survivability can be justified. Analysis of the survivability and/or operating limits versus the expected conditions inside the reactor building is needed (see Section 4.2.9.3.2) and therefore is it important this analysis is performed as part of the development of the detailed design.
- The use of basaltic concrete is assumed in the analysis of containment response to molten core concrete interaction (MCCI). Confirmation of this key assumption will be needed beyond GDA. The use of basaltic concrete also has an impact on other aspects of the design and this should be reflected in the documentation. For example, the UK ABWR does not have a carbon monoxide (CO) detector and Hitachi-GE argues that the use of basaltic concrete will ensure that the CO generation from MCCI is small, even if the erosion of concrete progresses.
- Low pressure injection valves are assumed to close against full reactor coolant system (RCS) pressure. It is understood that this assumption is currently justified on the basis of the purchase specification to the valve vendor for the Japanese ABWR (J-ABWR) and is included in the UK ABWR assumption list for future resolution. However, no specific confirmation that this will be part of the UK ABWR detailed design specification has been provided in GDA.
- The modelling of the hard-wired back-up system (HWBS) excludes some failure modes which are assumed to be detected by the daily surveillances.

83. The inspection conducted in May 2017 confirmed that the assumptions list was reviewed by Hitachi-GE in September 2016 against design changes, and any assumptions which were no longer necessary were ‘closed’. However, the closure of these assumptions has not been propagated to the rest of the PSA documentation. This has been captured as a minor shortfall.

84. It is noted that the assumptions list has been used by Hitachi-GE as a source for selection of sensitivity analyses. The sensitivity analyses undertaken show that some assumptions can have a significant impact on the risk profile. Cases of particular importance are summarised in Section 4.2.20.

85. It is important that the licensee develops a process to effectively enable the PSA assumptions to be captured in future design, construction and procedure development. This process should also ensure that the PSA model and documentation is updated to reflect any changes as information becomes available, and that the adequacy of the PSA assumptions is confirmed.

***Assessment Finding AF-UKABWR-PSA-001 (Part 3)**: The licensee shall:*

3. Develop processes and procedures to ensure the PSA assumptions are captured in future design, construction and procedure development. This process should also ensure that the PSA model and documentation is updated to reflect any changes to assumptions as more detailed information becomes available.

4.2.2.4 CONCLUSION

86. Assumptions in the PSA have been captured in a single location and overall are judged to be reasonable for the ‘generic’ PCSR.

87. My review has found that these assumptions have primarily been made either to supplement a lack of design or procedural information, or due to simplifications in the analysis. Hitachi-GE has undertaken sensitivity analyses that have enabled me to understand the potential impact on the risk profile of certain key assumptions.

88. The PSA assumptions will need to be reviewed beyond GDA when further information becomes available. Some of the PSA assumptions should also be used to support the detailed design development as they capture important insights that can help to reduce the UK ABWR risk ALARP.

4.2.3 General Expectations – Computer Codes and Inputs (A1-1.4)

4.2.3.1 ASSESSMENT

89. During the review of the different technical areas of the PSA, my team reviewed the adequacy of the various codes and how they have been used by Hitachi-GE. My review has focused on the two main codes used in the UK ABWR PSA; the CAFTA software developed by EPRI and MAAP (Version 4) code used for the PSA supporting calculations for containment heat removal success criteria and severe accident analysis. Other codes used by Hitachi-GE include:

- SAFER (majority of the core cooling success criteria), ODYN/TASC (success criteria for reactivity control functions in case of failure of the reactor protection system (RPS)) and TRAC-G (reactivity control, core cooling and pressure boundary protection success criteria). These codes have also been used by Hitachi-GE to undertake design basis analysis (DBA); the review of the validation and verification (V&V) of Hitachi-GE's models and documentation status have been undertaken by ONR's fault studies inspector (Ref. 41). The review of the adequacy of the success criteria analysis for the PSA is reported in Section 4.2.5.
- SHEX is used in the station blackout (SBO) analysis (Ref. 84), but has not been explicitly referenced in the PSA documentation; as this code does not underwrite any analysis in the PSA. The use of this code has been reviewed by ONR's fault studies inspector.
- GOTHIC has been used for heat up calculations supporting the level 1 PSA for selected rooms. I have considered these analyses as part of my review of the level 1 PSA systems analysis. The findings of my review are reported in Section 4.2.5. ONR's severe accident inspector has also looked at the adequacy of this code as part of their review of Hitachi-GE's hydrogen management safety case (Ref. 42).
- JASMINE code has been used to determine the intensity of a steam explosion due to ex-vessel fuel coolant interaction (FCI). The review of the adequacy of this code has been undertaken by ONR's severe accident and fuel and core inspectors and is reported in Ref. 85.
- For the structural response to FCI steam explosion pressure waves, AUTODYN models have been applied. The review of the adequacy of this code and its application was undertaken by my review team and ONR's severe accidents inspector supported by a specialist TSC.
- STAR-CCM+ (Version 7.06.012) is used for computational fluid dynamic analysis of corium ejection into the lower drywell (LDW) in support of the containment performance analysis (Ref. 29, Appendix D). In conjunction with ONR's severe accidents inspector, it was decided not to sample this code for the review of the V&V of their models and documentation status. This was based on the judgment of myself and ONR's severe accident inspector that

there was a low risk that deficiencies in these codes would impact on the validity of Hitachi-GE's severe accident safety case and PSA models.

- The review of the codes used for consequence analyses and level 3 PSA is reported in Section 4.2.19.

90. My judgement regarding the adequacy of the codes and their use by Hitachi-GE for the PSA is based on the following:

- My review team members' knowledge of the codes and their international status.
- An inspection of UK ABWR V&V testing performed for the following selected PSA software: CAFTA (Version 6.0b), PRAQuant (Version 5.2), and FTREX (Version 1.8) and FRANX (Version 4.2).
- Independent quantification of the PSA performed by my review team.
- The review of the adequacy of the MAAP code input deck, parameter file, graphical outputs and other results (Ref. 86). This review also considered Hitachi-GE's demonstration in Ref. 87 that the use of the most recent version of MAAP (Version 5) would not have had an impact on the outcomes of these analyses. Sensitivity analyses were also provided by Hitachi-GE to evaluate the impact of code limitations such as the modelling of ex-vessel corium coolability.
- Hitachi-GE's examples for the validation of MAAP for severe accident phenomena and other characteristics of boiling water reactor (BWR) type reactors. This included a comparison (Ref. 88) of selected unmitigated sequences between Hitachi-GE's MAAP results for the UK ABWR with results obtained with MELCOR and the J-ABWR by the Japanese Nuclear Energy Safety Organisation (JNES).
- As part the review of the level 1 PSA success criteria analysis my review team compared the inputs and the outputs of Hitachi-GE's deterministic calculations supporting the level 1 and level 2 PSA with other BWR and ABWR published analyses.
- As part of closure of RI-ABWR-0002, I considered Hitachi-GE's overall allocation of suitably qualified and experienced persons (SQEPs) in the field of PSA to develop the UK ABWR PSA. Although I did not specifically review the records of the qualifications and experience of the code analysts, this was evaluated through the interactions that my review team had with Hitachi-GE during GDA (Ref. 79).
- Input from ONR's fault studies and severe accidents inspectors related to the adequacy of MAAP, SAFER, ODYN, TRAC-G and JASMINE (Refs 41 and 42).
- The outcomes of confirmatory analyses with MELCOR of some severe accident scenarios that were undertaken by ONR's fault studies TSC. Further information is reported in Ref. 42.
- Input from ONR's civil engineering inspector, supported by ONR's structural integrity inspector, regarding the adequacy of ABAQUS for containment performance analysis under severe accident loads (Ref. 89). Further information is provided in Section 4.2.18.3.3.

91. My assessment was performed using the latest available models and documentation provided by Hitachi-GE, which are described in Sections 2 and 3 of this report. My assessment was carried out against the expectations in ONR's PSA TAG (Ref. 4).

4.2.3.2 STRENGTHS

92. My review of the verification and validation (V&V) methodology, testing, and results of the CAFTA software has found it to be adequate. A few shortfalls were identified (see below), but they do not invalidate the conclusions of the V&V tests and results.

93. The review of level 1 PSA thermal-hydraulic analyses confirms that the selection and application of the thermal-hydraulic codes is representative of the UK ABWR. Generally, these analyses are traceable and well documented. However, some exceptions are noted in Section 4.2.5.

94. MAAP 4 is a well-recognised and widely used code, which was originally developed for severe accident analysis (SAA). The targeted review of key input parameters undertaken by my review team has confirmed that the initial and boundary conditions for the sequences were correctly modelled with no errors or inconsistencies noted.

4.2.3.3 FINDINGS

95. The detailed technical reviews that support the findings discussed in the following paragraphs are documented in Refs 90 and 91.

96. Hitachi-GE's use of the modern PSA software CAFTA is considered to be overall adequate. However, the description of some of the gates in the PSA needs to be improved. In particular, higher-level gates do not always identify specific systems, which impair the understanding of the model. The documentation of fault tree gates also needs improvement as several inconsistencies and gaps were found (see Section 4.2.7). This has been captured as a minor shortfall.

97. RQ-ABWR-0817 summarises a number of findings that my review team identified during the inspection of the V&V of the CAFTA software. These findings do not impact the conclusions of the assessment or the calculations performed by Hitachi-GE. However, any findings should be addressed to ensure the completeness of the PSA documentation. This has been captured as a minor shortfall.

98. Review of the codes used to support the PSA by the ONR fault studies inspector confirmed that they are well established and appropriately applied. However, SHEX, SAFER, ODYN and the spreadsheet tool (used for shutdown and SFP faults) are not considered to be 'best estimate' codes (Ref. 41). Using more 'best estimate' codes to support the PSA may have an impact on the success criteria resulting in larger grace times available for operator actions. In particular, this can have an impact on the PSA risk profile for faults affecting a shutdown reactor or the SFP, where grace times are already large, as discussed in Sections 4.2.15 and 4.2.16. Future updates of the PSA, should consider whether an update of the success criteria analyses to achieve more 'best estimate' results is needed. This has been captured as a minor shortfall.

99. For SBO scenarios in the PSA, the reactor thermal-hydraulic transient water level inside reactor vessel and core heat up are analysed by SAFER, with the containment pressure and temperature being analysed by SHEX and MAAP. The evaluation of the adequacy of SHEX code has been considered by ONR's fault studies inspector (Ref. 41). My review team has noted that the results from SHEX show significantly different plant conditions than those calculated using SAFER when compared with medium term loss of off-site power (LOOP) with common cause failure (CCF) of emergency diesel generators (EDGs), but diverse additional generator (DAG) success. Hitachi-GE argues that SHEX has a simplified model inside the reactor pressure vessel (RPV) compared with SAFER and the timing of any water level signal would be more accurately and realistically modelled by SAFER. A more detailed justification of the difference in results that are relevant to the PSA should be developed and included in the PSA documentation. This has been captured as a minor shortfall.

100. In view of the outcomes of my review, the use of MAAP (Version 4) was considered acceptable for GDA. However, future revisions of the PSA should include consideration of relevant 'state of the art' codes. For example, MAAP (Version 5) presents a better treatment of some severe accident phenomena and has the capability to model severe accident scenarios in the SFP. This has been captured as a minor shortfall.

101. A number of shortfalls regarding the severe accident calculations with MAAP have been identified as part of my review of the level 2 PSA and are reported in Section 4.2.18. In line with these findings, the outcomes of the confirmatory analysis, reported in Ref. 42, highlighted major areas of uncertainty regarding the modelling of some of the severe accident phenomena. ONR's severe accidents assessment report (Ref. 42) has identified the need to update the severe accident safety case to reflect new data and insights that become available as a result of ongoing and future investigations into the accident progression of Fukushima Dai-ichi. This information should also be used to improve the modelling of the severe accident phenomena in the PSA and reduce uncertainties.

102. The review of the use of AUTODYN code to address FCI impact on the containment (Ref. 92) found that the basic approach, analysis methodology and the use of the computer codes were adequate (Ref. 93). However, the review raised concerns regarding the scope and assumptions used in the analysis. These concerns were raised by ONR's severe accidents inspector in RQ-ABWR-1236.The review of this RQ confirmed that, overall, the information provided was judged adequate to support the model in the PSA. The outcomes of this review are reported in Ref. 42.

4.2.3.4 CONCLUSION

103. My evaluation of the shortfalls in this particular area, on the basis of the assessment performed by the PSA, fault studies and severe accident review teams, concludes that the codes supporting the PSA are appropriate and have been adequately applied to support the UK ABWR 'generic' PCSR PSA.

104. A number of shortfalls have been identified in the findings section above, however the scope of analysis is large and ONR review has been extensive. The number of shortfalls identified does not necessarily compromise the integrity of the analysis or results. Many of the shortfalls are related to: discrepancies between two independent codes (ie SHEX and SAFER), uncertainty in the severe accident analysis or identifying where more recent codes could have been used, with areas of further work identified. Differences between codes and uncertainties in complex analyses are expected, however it is important that the impact of the differences and uncertainties are understood and future analyses consider new information or codes which can help to reduce uncertainty.

4.2.4 Level 1 PSA: Identification and Grouping of Initiating Events (A1-2.1)

4.2.4.1 ASSESSMENT

105. A detailed review was conducted during Step 3 to confirm whether the basis of the PSA is robust and to gain confidence on its completeness (Ref. 77).

106. My Step 3 review concluded that a significant number of initiating events (IEs) were missing or not explicitly considered in the PSA. My review team also identified the need for Hitachi-GE to enhance the documentation of the 'Identification and Grouping of Initiating Events' so that the traceability and completeness are evident. A summary of the shortfalls is provided in Ref. 77.

107. My Step 3 review also identified a significant number of cases where the process for grouping initiating events was not clear, ie the grouping criteria and the mapping to

derive the final initiating event groups were not transparent. The most significant shortfall was Hitachi-GE's approach to define generalised support system initiating event groups in a way that masked the true nature of the initiator. This was judged to likely prevent the uncovering of specific vulnerabilities.

108. My review team also identified issues with the modelling of locations of loss of coolant accidents (LOCA), modelling of interfacing system LOCAs (ISLOCAs) definition of LOCA sizes and conditional LOOP.

109. The shortfalls identified by my Step 3 review were captured in RO-ABWR-0042.

110. During Step 4 my review team evaluated, in detail, Hitachi-GE's responses to RO-ABWR-0042. This included a revised level 1 IEAP PSA which was submitted to ONR at the end of Step 3 (and updated several times during Step 4).

111. The detailed review of the 'Identification and Grouping of Initiating Events' discussed in this section of the report focuses on internal initiating events that can occur in the full power operating mode. The review of the initiating events in low power and shutdown, those related to the fuel ponds and of the initiating events that can occur as a consequence of external and internal hazards are documented in their respective sections of this assessment report.

112. As part of my review in Step 4, I raised several RQs (Ref. 94) that Hitachi-GE mostly addressed through PSA model and documentation updates in January 2016 and June 2016 or a documentation update in March 2017.

113. For the remaining shortfalls, I requested Hitachi-GE perform sensitivity analyses of my review findings to evaluate the impact on the risk. These sensitivity analyses were provided in Ref. 5. I have considered the adequacy of these analyses as part of my review. Additionally I have used some of the insights of these analyses, in combination with qualitative arguments and quantitative information from the PSA, to understand the potential risk significance of the findings in this area.

114. My assessment was performed using the latest available models and documentation provided by Hitachi-GE, which are described in Sections 2 and 3 of this report. My assessment was carried out against the expectations in ONR's PSA TAG (Ref. 4).

4.2.4.2 STRENGTHS

115. In response to RO-ABWR-0042, Hitachi-GE has performed a systematic identification of initiating events. The documentation provides a clear justification including records of the analysis undertaken for the identified initiating events, which demonstrates the completeness of the initiating events list. This supporting analysis includes the failure modes and effects analysis (FMEA) developed to support the UK ABWR fault schedule (Ref. 95) in response to RO-ABWR-0007 (Ref. 96), RO-ABWR-0008 (Ref. 97) and RO-ABWR-0010 (Ref. 98). Assumptions are also explicitly captured in the documentation.

116. Loss of support system initiating event fault trees have been developed which facilitates the modelling of dependencies.

4.2.4.3 FINDINGS

117. The detailed technical review that supports the findings discussed in the following paragraphs is documented in Ref. 90.

118. Hitachi-GE's PSA submitted at the end of Step 3, represented an improvement, but I identified that additional work was required. Therefore, I raised a number of RQs (Ref. 94) and held technical workshops with Hitachi-GE in March and October

2016 (Ref. 79). As a result, Hitachi-GE updated the PSA in January 2016 and June 2016 to address the majority of the shortfalls identified by my review team, extended the list of initiating events and provided sensitivity analyses to evaluate the impact of some of the remaining shortfalls that were not considered in these updates.

119. My review in Step 4 has concluded that the following IEs are missing from the PSA and should be included:

- Potential IEs triggered by software errors that would create conditions that spuriously induce actuation of systems and also simultaneously 'freeze' the actuation logic for safety systems. These potential IEs have been identified by Hitachi-GE in response to RO-ABWR-0007 (Ref. 41), but have not been included in the PSA.
- The potential for a reactivity excursion following slow insertion of control rods (Ref. 41).
- The loss of electrical system due to CCF of components such as transformers, batteries and chargers, uninterrupted power supply for C&I, interlocks etc. The initiating events modelled in the PSA only consider CCFs of circuit breakers and protection relays.
- The spectrum of break sizes outside containment may need to be extended to consider small or medium breaks (only large LOCAs are currently considered). Deterministic analyses have been provided and reviewed by ONR's fault studies inspector (Ref. 41), however the PSA has not been updated to take them into account.
- Break outside containment (BOC) events should consider the failure of the high pressure core flooder system (HPCF) pipe segment as a possible contributor. This fault could occur as a result of HPCF pump start or internal leakage of the normally closed motor operated valve (MOV). While such failures are low likelihood they are included as failure modes within the scope of NUREG/CR-6928 (Ref. 99) and have frequencies within the range of other BOC scenarios explicitly included in the model.

120. In addition, the PSA does not consider a loss of ultimate heat sink (LUHS) that could lead to the loss of all external water sources such as, for example, blockage of the intake. My assessment of the impact of this missing IE on the risk profile is presented in Section 4.2.13.

121. Some of the missing initiating events identified above have been assessed deterministically and included in Hitachi-GE's fault studies submission during GDA. A review of the fault schedule should be undertaken to confirm that it is aligned with the PSA and differences are justified.

122. My review team identified that there were cases in which either the plant response would not be the same for all the IEs grouped together, or the success criteria used in the event tree for the IE group was not applicable to all the IEs included in the group. In particular, there is a lack of clarity on whether the loss of support systems IEs, including loss of instrument air, reactor building cooling water system (RCW) and reactor building service water system (RSW), were incorrectly modelled as a manual shutdown instead of reactor trip. This is potentially optimistic as a reactor trip could have a more onerous impact on the plant.

123. Upon request, Hitachi-GE clarified that in the J-ABWR there are cues and procedures that would allow operators to perform administrative shutdown before automatic or manual scram occurs. However, my view is that the UK ABWR PSA model is optimistic as it does not include a failure of operator response to manually shutdown the reactor given the loss these support system initiators. Furthermore, CCFs of several or all the trains of the support system are not always considered. The probability that a manual shutdown is completed prior to a reactor trip should be explicitly modelled in the PSA,

including consideration of available operator's cues, the failure of multiple trains and their impact on the time window.

124. The risk impact of the above missing initiating events was considered by my review on the basis of Hitachi-GE's sensitivity analyses, the PSA results and associated importance measures. In the absence of detailed design information during GDA, I have conservatively evaluated the impact on the risk of the shortfalls identified in my review. I have identified that the following shortfalls could have the highest impact on the risk profile of the internal events at power PSA. The evaluation of the impact is reported below.

- Potential IEs triggered by software errors that could also 'freeze' actuation logic for safety systems. Hitachi-GE has provided sensitivity analyses that assumed the class 1 C&I system could spuriously activate the automatic depressurisation system (ADS) opening or main steam isolation valves (MSIVs) closure due to occurrence of a software CCF which then 'freezes' and prevents the actuation of the RPS or ECCS. The impact on the results in the sensitivity study was small; however, this was primarily due to low software CCF probability assumed by the study. No justification of the software CCF data used was presented by Hitachi-GE and no clarity was provided regarding the impact of this assumption on the results. In view of the high importance of the digital C&I system, these scenarios could potentially have an impact on the risk profile of the internal events at power (IEAP) PSA. This impact will be limited by the reliability and detailed design of the UK ABWR hard wired back-up system (HWBS) and the digital C&I system. The analyses of these spurious initiating events should be included in the PSA and used to inform the development of the detailed design of the UK ABWR HWBS and digital C&I system. Detailed update of the PSA modelling for C&I initiating events is expected when the design is further developed.
- Loss of support systems incorrectly modelled as a manual shutdown instead of reactor trip. As Hitachi-GE identified, it is expected that there are cues and procedures that would enable the operators to perform an administrative shutdown before automatic or manual scram occurs for most of the loss of support systems. During GDA there was a lack of information to confirm which initiating events would need to be modelled as a reactor trip. Hitachi-GE undertook a conservative sensitivity analysis, which assumes that loss of support systems would result in a reactor trip which resulted in an increase of 6% to the large release frequency (LRF) for the IEAP PSA. A 'best estimate' evaluation of the risk should consider the probability of the operator to administrative shutdown the reactor and the time available. It should also be noted that scenarios for which the time available can be more limited would usually be due to CCFs; however, these would have a lower likelihood of occurrence and therefore expected to limit their contribution to the risk profile.

4.2.4.4 CONCLUSION

125. My evaluation of the shortfalls in this particular area on the basis of Hitachi-GE sensitivity analysis has shown that none of the shortfalls identified would lead to a significant increase in the risk results. If the assumptions are not supported, the PSA will require update and there may be a more significant risk impact. A number of shortfalls have been identified, however the size and scope of the analysis is large and ONR review has been extensive. The number of shortfalls does not necessarily compromise the integrity of the analysis or results.

126. Based on the outcome of this assessment, I have concluded that the list of initiating events (IEs) together with the results of the sensitivity analyses undertaken by Hitachi-GE, are sufficient for a reasonable understanding of the UK ABWR risk associated with internal events at power and to close RO-ABWR-0042 (Ref. 100).

127. Further work is required in the site specific stage. The list and grouping of initiating events included in the PSA should be reviewed and completed in the site specific stage to allow the PSA to support further stages of the nuclear power plant (NPP) development. This is expected to take place as part of resolution of AF-UKABWR-PSA-001, to address any shortfalls identified in this report, or as part of normal business to update the PSA to reflect site specific aspects or design development.

4.2.5 Level 1 PSA: Accident Sequence Development – Determination of Success Criteria (A1-2.2)

4.2.5.1 ASSESSMENT

128. A review of the UK ABWR PSA task on 'Success Criteria Analysis' methodology and examples of its implementation against the expectations in the ONR's PSA TAG (Ref. 4) was conducted during Step 3 (Ref. 77). This review raised general concerns in the following areas:

- There was lack of clarity regarding the mapping between the accident sequences depicted in the event trees and the success criteria analyses, including the limiting conditions defined for success and failure.
- The limiting conditions defined for the success criteria and failure were not always provided for reactor pressure vessel, containment integrity and other safety functions considered critical for plant operation.
- Power flow oscillations and reactivity excursions were not addressed in the core damage success criteria.
- There were sequences for which the success criteria had not been entirely defined or for which supporting analyses to demonstrate the success path had not been provided.
- There was a lack of clarity regarding the justification of time windows for operator actions.
- There was also a lack of clarity regarding the sequence assumptions adopted in the analyses (eg, LOCA break location). The review identified examples of assumptions that did not appear to be appropriately chosen and justified to be bounding for the sequences depicted in the event trees.
- The influence of the physical conditions that arise during the evolution of the sequences on the functionality and operability of the systems and the functions did not appear to be always taken into consideration in the evaluation of the success criteria.
- Conservatisms were in general not identified and there was a lack of justification to demonstrate that there were no excessive conservatisms.

129. The shortfalls identified by my review were captured in RQ-ABWR-0559 which is a reference to RI-ABWR-0002.

130. During Step 4 my review team evaluated in detail Hitachi-GE's response to RQ-ABWR-0559, which primarily consisted of an updated level 1 IEAP PSA and was submitted to ONR at the end of Step 3 (which was subsequently updated several times during Step 4).

131. The objective of the Step 4 PSA assessment was to undertake a detailed review (on a sampling basis) of this technical area in order to consider if the concerns raised during my Step 3 assessment had been satisfactorily addressed. For the assessment of the UK ABWR PSA success criteria this review considered the following:

- The adequacy of the technical basis for plant response to prevent core damage, RPV failure and containment failure considered in the PSA.
- A detailed review of analyses supporting the success criteria for key functions and systems for a number of accident sequences for the following IE groups:
  - transients (including turbine trip, loss of feedwater, MSIV closure);
  - LOCAs (small, medium, large LOCA inside containment; LOCA outside containment; interfacing systems LOCA);
  - special initiators (supporting system initiating events including: loss of a single alternating current (AC) bus, loss of a direct current (DC) bus, failure of RCW);
  - transient with failure to SCRAM (or anticipated transient without SCRAM (ATWS)).

132. The above selection provided a good representation of all the types of initiating events that can occur in the UK ABWR and ensured that my review addressed the thermal-hydraulic behaviour of the reactor in a comprehensive manner.

133. The PSA submitted at the end of Step 3 represented an improvement, but I identified that additional work was required, raised a number of RQs (Ref. 94) and held technical workshops with Hitachi-GE in March and October 2016 (Ref. 79). As a result, Hitachi-GE updated the PSA in January 2016 and June 2016 to address most of the shortfalls identified by my review team in Step 3 and Step 4. A further documentation update was provided in March 2017.

134. In addition, Hitachi-GE provided sensitivity analyses (Ref. 5) to evaluate the impact of some of the remaining shortfalls that were not considered in these updates. I have considered the adequacy of these analyses as part of my review. I have used some of the insights of these analyses, in combination with qualitative arguments and quantitative information from the PSA, to understand the potential risk significance of the findings in this area.

135. My assessment was performed using the latest available models and documentation provided by Hitachi-GE, which are described in Sections 2 and 3 of this report. My assessment was carried out against the expectations in ONR's PSA TAG (Ref. 4).

4.2.5.2 STRENGTHS

136. The updated PSA has included improvements in the documentation such as a route-map from each success sequence in the level 1 IEAP PSA to the supporting analyses that underpin its success criteria. This update also included an extension of sequence and success criteria analyses to cover most of the new initiating events identified in response to RO-ABWR-0042 and addressed many of the shortfalls identified in Step 3.

137. The revised UK ABWR PSA is supported by representative thermal-hydraulic analyses performed to demonstrate that each of the success paths (and systems) claimed in the event trees lead to successful outcomes (eg, non-core damage). These analyses were clearly identified and traceable in the revised PSA documentation.

4.2.5.3 FINDINGS

138. The detailed technical review that supports the findings discussed in the following paragraphs is documented in Ref. 90.

139. My review in Step 4 has concluded that there are a number of concerns still outstanding which should be addressed beyond GDA. A summary is presented below.

140. Overall my review has found that the analyses performed by Hitachi-GE are representative of the UK ABWR and sufficient to demonstrate that the success paths

depicted in the event trees lead to successful outcomes. However, the review has identified the following concerns related to specific scenarios:

- There is a lack of analysis to assess the influence of the physical conditions that arise during the evolution of LOCAs, BOCs and ISLOCAs on the functionality and operability of the systems (ie HPCF). Therefore, it has not been fully demonstrated that the functions assumed in the success criteria analyses and modelled in the event trees are realistic for these scenarios. I consider that the environmental conditions in the reactor building following these events should be established and used to justify these implicit assumptions. Hitachi-GE indicated that the reactor building design is not sufficiently mature within GDA to enable such an analysis. Hitachi-GE expects that potential design improvements identified to mitigate impacts from fires and high-energy line breaks (HELBs) would also reduce the impact of such environmental conditions (see Section 4.2.11).
- The PSA currently assumes that containment heat removal is not required for LOCAs outside containment. However, there is no analysis supporting this modelling assumption. In particular, for cases such as the reactor water clean-up system (CUW) sample line BOC, it is not clear that this line alone can convey enough energy to prevent heat up of the suppression pool (S/P). However, it is noted that the contribution of these scenarios to the overall risk is small.
- The PSA initially assumed that class 2 DC batteries in the back-up building would survive and be available for 14 hours. Upon request, and as additional design information became available, Hitachi-GE has explained that the real capacity of the batteries will likely be smaller than that initially assumed in the PSA. Hitachi-GE has indicated that alternative options to manually close the necessary circuit breakers without DC batteries need to be considered. The PSA assumption in Ref. 101 has been updated, however the PSA model has not been. The PSA should be modified to adequately reflect the battery life and other alternative measures to ensure core cooling even if DC battery supplies have been depleted.

141. The PSA documentation does not provide an explicit demonstration that the success criteria for initiating events groups bounds all potential actuations that could contribute to more severe conditions, including potential system malfunctions. Ref. 101 identifies a small number of assumptions dealing with expected system actuation that could contribute to more severe conditions. Although these are considered by my review team as reasonable and no errors or omissions have been noted, there is a lack of justification that they represent a complete treatment of this issue. The PSA documentation should be revised to include a justification that success criteria reasonably bound all potential actuations.

142. The documentation of the analyses used for derivation of success criteria has been identified by my review as an area that could benefit from enhancement. It would be beneficial to improve traceability, to include all the analysis cases in a single document with a clear identification of inputs, systems available, actuation times, and the resulting RPV and containment conditions. This has been captured as a minor shortfall.

143. In addition, the documentation does not provide a clear identification of the minimum equipment requirements and performance for success for each success criterion. Furthermore, my review has identified the following cases for which the analyses are not based on 'best estimate' considerations and may result in a conservative bias on core damage frequency (CDF) and importance measures; these should be addressed in future development of the PSA. Examples include:

- There are a number of sequences resulting from the failure of low pressure and high pressure injection for which the residual heat removal system (RHR) is available but not claimed by the event tree logic.
- For some scenarios the RPV depressurisation success criteria applied may be conservative. For example, for scenarios where RPT and RPS fail, it is assumed that only 14 SRVs are available to open and therefore RPV failure cannot be avoided. Additional analysis (Ref. 5) shows that if all16 SRVs were considered to be available, then RPV failure could be avoided. Sensitivity analysis (Ref. 5) performed by Hitachi-GE shows that the impact on risk of this particular conservative assumption is small.
- The low pressure core flooder system (LPFL) is not claimed following failure to depressurise the RPV via ADS. In cases of ADS failure, RPV depressurisation via the reactor depressurisation control facility (RDCF) is claimed to allow for provision of flooding system of specific safety facility (FLSS). LPFL could be claimed following ADS failure and RDCF success.
- Further consideration is needed to determine whether crediting feedwater in some small LOCA scenarios is possible. When crediting feedwater is not possible, it should be documented.
- The analysis should clearly establish the number of SRV tail pipe failures and open SRVs that can compromise the containment overpressure protection system (COPS). Currently the analysis appears to be conservative.
- Failure to open both of the tailpipe check valves is assumed to lead to a break in the wetwell airspace. However, deterministic structural analyses have not been provided to justify this treatment.
- The FLSS system success criteria require two out of four pumps for success. However, if some other systems such as the reactor core isolation cooling system (RCIC) operate for a period of time, one out of four FLSS trains could potentially support some success paths.
- ATWS scenario credits one out of two standby liquid control system (SLC) pumps given all ten reactor internal pumps (RIPs) trip. The PSA should consider cases with fewer than ten RIPs tripping.
- The control rod drive (CRD) system is not credited as a form of RPV injection. The timescales of some of the CDF dominant scenarios are long and credit for CRD injection may be beneficial. It is acknowledged that this benefit may be limited by CRD dependencies on support systems.
- The EDGs, DAG, and back-up building generator have simplistically assumed mission times of 24 hours for LOOP independently of shorter AC recovery times.
- Loss of condensate storage tank (CST) inventory is simplistically assumed to cause both unavailability of the CST and prevent automatic switch over to the suppression pool.

144. Other potential additional success paths not included in the PSA (resulting in conservative analysis) may include: venting success for ATWS sequences, off-site AC recovery after 14 hours, potential credit for makeup water condensate system (MUWC) and CUW as accident mitigating systems for RPV injection and heat removal system respectively, potential credit for RCIC operation with a single SRV to depressurise the RPV and the possibility of crediting repairs (if supported by analyses).

145. Overall, the review has found that most of the shortfalls identified are related to conservatisms in the PSA model. Regarding the gaps identified by my review, Hitachi-GE has provided sensitivity analyses to evaluate their impact on the risk (Ref. 5). On the basis of this information, I have conservatively evaluated the impact on

the risk of the shortfalls identified in this particular area of the PSA. I have identified that the following shortfalls could have the highest impact on the risk profile for the IEAP PSA:

- The lack of substantiation of HPCF claims following BOC / ISLOCA events. A sensitivity analysis was performed by Hitachi-GE limiting claims for injection systems inside the reactor building. The results show an increase to LRF of 20%, which provides an upper bound to the risk increase if the claim on equipment survival cannot be substantiated. As mentioned previously, Hitachi-GE expects that potential design improvements identified to mitigate impacts from fires and HELBs would also reduce the impact of environmental conditions. I expect substantiation, or further ALARP justification, to be provided in the site specific stage.
- The potential for the battery life modelled into the PSA to be optimistic and not aligned with the design reference. Hitachi-GE provided a sensitivity analysis to evaluate the impact on the risk of not crediting the class 2 DC batteries, the sensitivity study showed a small impact on LRF (3%).

4.2.5.4 CONCLUSION

146. The underlying analysis supporting the success criteria for the sequences examined by my review was judged to be adequate, although there are some gaps in the PSA models and documentation. The shortfalls with the greatest potential to impact the UK ABWR risk profile are related to the substantiation of the design. Further design substantiation is expected beyond GDA. This is expected to take place as part of resolution of AF-UKABWR-PSA-001 to address any shortfalls identified in this report, or as part of normal business to update the PSA to reflect site specific aspects or design development. The sensitivity analyses performed by Hitachi-GE provide an upper bound of their potential impact on the risk and show that the risks remain within the same order of magnitude as the current results and well below the ONR SAPs numerical targets basic safety levels (BSLs).

147. My review has also identified that the PSA documentation needs enhancement and the PSA should be developed further to remove undue conservatisms that could distort the risk profile and importance measures.

4.2.6 Level 1 PSA: Accident Sequence Development – Event Sequence Modelling (A1-2.3)

4.2.6.1 ASSESSMENT

148. A review of the UK ABWR PSA ‘Event Sequence Modelling’ (event trees) methodology and examples of implementation against the expectations in ONR’s PSA TAG (Ref.4) (Table A1-2.3) was conducted during Step 3. My review (Ref. 77) noted:

- The documentation of the accident sequence analyses needed improvement to meet regulatory expectations in ONR’s PSA TAG (Ref. 4). For example, the functional description of each event tree node and its applicability to each event tree branch was not provided in the documentation. In addition, the links to the supporting thermal-hydraulic analyses were not always identified.
- There was a lack of detailed discussion of key scenarios such as those involving total loss of AC power (referred to as station blackout (SBO)). Justification of the accident sequence duration assumed or systems mission times were not explicitly provided.
- Reference to emergency operating procedures (EOPs) / surrogate EOPs was lacking for the UK ABWR PSA accident sequence analyses.

- The general assumptions relating to all event tree development were not always defined 'up-front' and properly justified.
- The PSA had a number of sequence end states that did not result in 'success' nor 'core damage'. However, these end states were not clearly identified and defined in the PSA documentation, nor their rationale for their use to support the PSA safety claims explained. In particular, it was not clear what the overall contribution to the PSA results was and how they compare against ONR SAPs numerical targets such as Target 8.
- The PSA documentation did not identify all the dependencies or provide explanation of the way in which such dependencies were treated and included in the accident sequences.
- There was a lack of consideration of consequential initiators (other than consequential LOOP).
- The review identified examples in which the event trees did not appear to have been constructed correctly to provide adequate representation of the progression of the accident sequences for all initiating events (IEs) under specific IE groups.
- The UK ABWR level 1 and level 2 IEAP PSA were not linked in an integrated model and therefore it was not possible to automatically calculate point estimates of the overall risk, importance measures, and parametric uncertainties or provide a merged minimal cutset (MCS) list.

149. The issues identified by my review were captured in RQ-ABWR-0559 which is a reference to RI-ABWR-0002.

150. During Step 4 my review team evaluated, in detail, Hitachi-GE's response to RQ-ABWR-0559 and RI-ABWR-0002, and an updated level 1 IEAP PSA submitted to ONR at the end of Step 3 and updated several times during Step 4.

151. The objective of the Step 4 PSA assessment was to undertake a detailed review (on a sampling basis) of all PSA technical areas in order to consider if the concerns raised during my Step 3 assessment had been satisfactorily addressed. For the assessment of the UK ABWR PSA event trees my review team selected ATWS, LOCAs, long term LOOP and loss of RPV injection scenarios (represented by end-state TQUV). This was judged to be a good representation of all the types of initiating events and the various aspects related to the evolution of accident sequences.

152. The updated versions of the event sequence analysis for internal events at power (IEAP), SFP and shutdown PSAs included consideration of lower dose band end states. The results for the lower dose band sequences are quantified and fed into Hitachi-GE's assessment against ONR SAPs numerical targets 7 and 8 (Ref. 30). My review of this part of the event sequence analysis was performed on a sampling basis and covered multiple areas of the PSA, specifically the IEAP, SFP and shutdown PSAs (see Section 4.2.16 for SFP PSA and 4.2.15 for shutdown PSA). Assessment of the results when compared to Targets 7 and 8 is considered in Section 4.2.19.

153. As part of my review in Step 4, I raised several RQs (Ref. 94) that Hitachi-GE have mostly addressed through a PSA model and documentation update in January 2016 and June 2016 or a documentation update in March 2017.

154. I also requested Hitachi-GE to perform sensitivity analyses to evaluate the impact of my review findings on the risk. These sensitivity analyses were provided in Ref. 5. I have considered the adequacy of these analyses as part of my review. I have used some of the insights of these analyses, in combination with qualitative arguments and quantitative information from the PSA, to understand the potential risk significance of the findings in this area.

155. My assessment was performed using the latest available models and documentation provided by Hitachi-GE, which are described in Sections 2 and 3 of this report. My assessment was carried out against the expectations in ONR's PSA TAG (Ref. 4).

4.2.6.2 STRENGTHS

156. In response to RI-ABWR-0002 and RQ-ABWR-0559, Hitachi-GE has significantly expanded the scope of the event sequence analysis to address many of the shortfalls identified in Step 3. This has included a review of the documentation to ensure event trees success sequences are explicitly linked to a defined end state and transient analysis cases. Additional notes are provided where exceptions are made, and the end state assigned may not be the obvious one. The documentation also identifies when a sequence will be 'subsumed' by another sequence.

4.2.6.3 FINDINGS

157. The detailed technical review that supports the findings discussed in the following paragraphs is documented in Ref. 90.

158. The review in Step 4 has concluded that there are a number of shortfalls that are still outstanding and a summary is provided below.

159. Table A1-2.3.2 of ONR's PSA TAG (Ref. 4) identifies the expectation that dependencies (human actions, equipment, environmental, spatial, common mode failure, fluid medium, subtle dependencies) should be identified and treated correctly. Specific findings in this regard are:

- There is a lack of clarity on how containment failure or the catastrophic failure of the containment diaphragm impacts ECCS suction from the suppression pool (S/P) and from CST. Hitachi-GE clarified that the PSA assumes a number of systems to be inoperable before containment overpressure failure, due to possible boiling in the S/P that leads to the loss of net positive suction head (NPSH). This may result in conservatism in the PSA. The PSA documentation should be extended to adequately explain and justify the approach adopted.
- The PSA has assumed that some of this equipment control is backed up by the safety auxiliary panel (SAuxP) but the design was not completed at the time of the PSA development. Further work will be needed to align the PSA with the system design and document how the dependencies are modelled.
- The PSA consideration of the operation of the flooding system of reactor building (FLSR) is based on a number of assumptions regarding access, environment, status of the injection line, indications, and timing for operator actions. These need to be justified and potentially revised once information regarding the detailed design and procedures become available.
- The injection valves to the RPV could be overstressed in scenarios that result from SRVs failing to open and the RPV exceeding service level C such that they are no longer reliable for opening to allow injection flow. This comment is also applicable to consequential LOCA sequences where ECCS and FLSS are credited. The PSA does not explicitly consider the integrity of these valves, but Hitachi-GE has included an assumption to cover this issue. Information from manufacturers is needed to support this assumption and a review of the PSA modelling will be required if this assumption cannot be substantiated.
- Shortfalls identified related to human action dependencies are reported in Section 4.2.8.

160. Emergency operating procedures (EOPs) were not available in GDA; therefore, the link between the event trees and EOPs was found to not always be clearly represented. When EOPs are available, it should be confirmed that they are adequately represented

in the model or the model should be updated as needed. Some examples of shortfalls identified by the review are below:

- The PSA currently assumes the operator would use an external water source injection (FLSS/FLSR) with higher priority than HPCF or LPFL in some circumstances. Hitachi-GE acknowledges that the priorities currently assumed in the PSA should be revised. The current model is judged to potentially inflate the importance of FLSS and FLSR at the expense of HPCF and LPFL. This shortfall is identified in the Hitachi-GE PSA commitment log (Ref. 82), which records PSA shortfalls identified in GDA which require addressing following GDA.
- The PSA models that LPFL through the RHR heat exchanger is adequate for containment heat removal by suppression pool cooling for transient conditions, simultaneous with RPV injection by LPFL. The EOPs need to identify whether there is a need for the operators to split the flow or swap between RPV injection and suppression pool cooling, and any operator actions should be included in the PSA.
- Additional examples have been reported below and in Sections 4.2.5 and 4.2.8 regarding the consideration of operator intervention for a number of EOP guided actions, including: control of RPV water level below level 8 in the RPV, RPV water level control when injecting from an external source during a LOCA and operator intervention during ATWS for ADS inhibit.

161. A number of event trees will need to be modified to adequately represent the progression of the accident sequences for the following scenarios:

- The PSA does not consider whether the wetwell to drywell vacuum breakers (V/Bs) may cycle multiple times during a large LOCA scenario. Multiple cycles could lead to V/Bs sticking open. Further consideration of this shortfall and the evaluation of its potential impact in the risk profile are reported in Section 4.2.18.
- Hitachi-GE considered the impact of failure of containment isolation for BOCs in the level 1 IEAP PSA; in general in the level 2 IEAP PSA and in the evaluation of success sequences contributing to ONR SAPs numerical targets 7 and 8 (see Section 4.2.19.3). However, the secondary or consequential effects of containment isolation failure during LOCAs and transients have not been considered in the level 1 PSA (for example via flooding or impact of high humidity in reactor building (R/B) rooms). Upon request, Hitachi-GE has provided a qualitative evaluation of the secondary or consequential effects and their potential impact on SSCs. Hitachi-GE argues that the most likely scenario would be that any steam will be directed to the main condensers with the impact on the SSCs in the turbine building limited by an engineered pathway for steam release from the condensers. I expect this claim to be substantiated when detailed design information becomes available following GDA, and the PSA event trees modified as appropriate. I expect that the contribution to the risk profile of these scenarios will be limited as they will have low frequencies, once the probability of failure of the containment isolation is taken into consideration.
- The PSA does not explicitly address the loss of RPV level instrumentation. Hitachi-GE has indicated that if the RPV level instrumentation is lost, operators would flood the RPV. However, the consequences of this action should be explored and the limitations on containment water level conditions that would force termination of external water injection should be identified and included in the PSA event trees. A similar shortfall has been raised in the assessment of the level 2 IEAP PSA and the containment performance analysis.

- Hitachi-GE's accident sequence report (Ref. 101) assigns a consequential LOCA due to RPV overpressure as a large LOCA in feedwater system line A (FDW-A). For this analysis Hitachi-GE made two significant assumptions, which I requested it justify:

  - The location of the consequential LOCA. Breaks in various locations could impact mitigation equipment differently. For example, since FLSS injects via the FDW-A piping line, other locations may not impact FLSS injection as significantly.
  - The size of the consequential LOCA. There is a lack of analysis to substantiate that the consequential LOCA would have a break size consistent with a large LOCA. This assumption was considered optimistic as a smaller breach may not depressurise the RPV sufficiently for RPV injection by high or low pressure systems. In addition this assumption conflicted with the one made in the shutdown PSA (see Section 4.2.15), which assumed the primary circuit pressure did not reduce sufficiently following an SRV CCF to open to enable high pressure feedwater systems to be engaged.

  Hitachi-GE's commitment log captures the need to undertake further work to justify these modelling assumptions; sensitivity analyses have been performed by Hitachi-GE in GDA to evaluate their risk impact (see below).

- ATWS scenarios are potentially conservative. It is unclear why the ATWS power control by the SLC is not credited in sequences with high pressure injection unavailability and subsequent RPV level control with low pressure systems. In addition, RPV failure prevention by operation of all SRVs 'as-designed' has been omitted. Furthermore, the ATWS event trees do not credit containment venting after successful reactor shutdown using SLC injection. Hitachi-GE has clarified that these strategies have been omitted due to a reduced time window being available, but agreed that further analysis regarding the feasibility of these actions is needed.

162. My review has identified the PSA documentation of the accident sequence analysis should be improved to provide more detailed information and justification of the event tree logic adopted, including the following:

- Sufficiently detailed explanation of the gate structures to provide assurance that the flag files are free from errors and reasonably represent the intended sequence logic. In addition, it may be more beneficial in the long term to consider replacing the flag structures with actual event tree and/or fault tree logic.
- The functional description of the event tree nodes and its applicability to each event tree branch. It is noted that improvements have been made in Step 4 in Ref. 101, however further work is required to identify applicability to each event tree branch and to increase usability of the documentation. Hitachi-GE has identified, in response to RQ-ABWR-1070, that further improvements will be considered for the site specific PSA documentation.
- Detailed description of sequence dependencies.
- In some cases, more detailed justification for the accident sequences should be provided in the documentation. The documentation of LOOP / SBO event tree modelling remains an example of lack of clarity regarding the justification for the fault tree modelling.

This has been captured as a minor shortfall.

163. Upon request, Hitachi-GE provided additional evidence to underpin the assumed mission times and event tree end states identified as a safe, stable condition for each

success sequence in the PSA. However, the following shortfalls have been identified by my review:

- There are scenarios for which it is unclear what systems have adequate inventory to maintain RPV level long enough to guarantee a sustainable safe and stable state and what system(s) can be used for decay heat removal. In particular, following a LOCA initiator when using an external water source for mitigation, there is insufficient clarity on which sources would be considered viable to reach a safe and stable condition when RPV water level is not controlled. Furthermore, the review has identified inconsistencies in PSA claims on water sources for transients and LOCAs inside the primary containment vessel (PCV). Cases of particular concern are LOCA below top of active fuel (TAF) and BOC / ISLOCA / LOCA, as the contents of the suppression pool may be discharged and therefore unavailable for RPV feed before core damage occurs. In response to my RQs (Ref. 94), Hitachi-GE have identified that manual RPV level control operator actions may be required in the PSA for some of these scenarios (Ref. 82). However, the supporting information to demonstrate that these actions are feasible has not been provided.
- There is a lack of clarity regarding the room heat up acceptance criteria (see Section 4.2.7 for further detail and evaluation of the associated impact on the risk profile) that need to be addressed as part of the development of the site specific PSA to ensure that the PSA model of the heating, ventilation and air conditioning system (HVAC) adequately reflects the detailed design of the UK ABWR.
- It is not clear how the PSA addresses late failures of the containment. Late failures of the containment may impact in the availability of modelled systems in the long term.

164. My evaluation of the risk significance of the findings in this particular area has considered the following sensitivity analyses provided by Hitachi-GE in response to the outcomes of my review (Ref. 5):

- evaluation of the risk impact of a change in the chronological position of the external water source injection (FLSS) in the event trees;
- evaluation of the risk impact if automatic RPV water level control by HPCF is not credited during BOC/ISLOCA scenarios; for LOCAs inside containment only the suppression pool is credited as a long term water source;
- evaluation of the risk impact when the 'level 8 signal' for automatic level control is unavailable for LOCAs with break position between level 8 and TAF. These scenarios may lead to submergence of the vacuum breakers and unavailability of wetwell vent lines due to water accumulation in the PCV during a LOCA. The sensitivity performed assumes a human failure event (HFE) for termination of injection systems prior to R/B flooding;
- evaluation of the risk impact of a different consequential LOCA break location in case of overpressure (including main steam line, SRV inlets, feedwater system line A (FDW-A), RHR suction and CUW mid-vessel suction) and size (assuming RPV remains at high pressure given no injection is possible).

165. My evaluation of the findings has also considered the PSA results and importance measures presented by Hitachi-GE. This is indicated in the text where relevant.

166. For the shortfalls identified in the review of this technical area, I have conservatively evaluated the impact on the risk. I have identified that the following shortfalls could have the highest impact on the risk profile of the IEAP PSA:

- Changes to the order in which RPV injection systems are claimed in the event trees are shown in a Hitachi-GE sensitivity study to increase the large release frequency (LRF) by up to 20%. This sensitivity has not credited FLSR and therefore can be considered conservative. The operating and emergency procedures will be developed beyond GDA. This sensitivity analysis highlights the importance of the PSA's role in risk informing operating procedures.
- The location, size and timing of a potential consequential LOCA in case of RPV over-pressurisation could have a significant impact on the risk profile and the importance of the SRVs. In particular, a smaller breach size that requires RPV depressurisation via the SRVs before RPV injection is possible. The risk significance of this scenario is limited by the CCF probability of the SRVs failing to open, as the SRVs are only partially diverse. The data used by Hitachi-GE's sensitivity analysis is consistent with Ref. 102, which I consider adequate for GDA. However, substantiation will need to be provided in the site specific phase. ONR's fault studies inspector reviewed the diversity of the SRVs from a deterministic point of view and considered the information provided was adequate for GDA (Ref. 41), however further work is expected beyond GDA to ensure the detailed design and test and maintenance (T&M) procedures minimise the likelihood of an SRV CCF. I expect that, subsequently, the PSA will be updated to reflect the outcomes of this work.
- Reflection of the on-site inventories in a number of scenarios and the potential need to include additional operator actions in the PSA to ensure sufficient inventory and flow rate is available (including level 8 RPV level control action). The results of Hitachi-GE sensitivity studies show a small impact on LRF (6%).

4.2.6.4 CONCLUSION

167. My evaluation of the shortfalls in this particular area on the basis of Hitachi-GE sensitivity analysis has shown that none of the shortfalls identified would lead to a significant increase in the risk results.

168. Based on the outcome of this assessment, I have concluded that, the current event trees are sufficient to support the UK ABWR 'generic' PCSR. In response to RI-ABWR-0002, Hitachi-GE has improved the event tree analyses compared with Step 3. Event trees are explicitly linked to the level 2 IEAP PSA and consequence analyses for ONR SAPs numerical targets 7 and 8, key assumptions have been captured, and the updated event trees are broadly considered to provide an adequate representation of the progression of the accident sequences.

169. However, this part of the UK ABWR PSA needs additional improvement to support further stages of the NPP development to include omitted dependencies and failure modes, remove undue conservatisms and provide a more detailed description of the accident sequence analysis. This is expected to take place as part of resolution of AF-UKABWR-PSA-001 to address any shortfalls identified in this report, or as part of normal business to update the PSA to reflect site specific aspects or design development.

4.2.7 Level 1 PSA: System Analysis (A1-2.4)

4.2.7.1 ASSESSMENT

170. I conducted a high-level review of the UK ABWR PSA task on 'Systems Analysis' (Fault Trees) against the expectations in the ONR's PSA TAG (Table A1-2.4) during Step 3. This review raised general concerns in the following areas:

- The system descriptions and fault tree models did not provide a characterisation of operation of the structures, systems and components (SSCs) during accident conditions.

- The listing of human failure events (HFEs) appeared to be limited and pre-initiator HFEs were not considered.
- There were missing intersystem and intrasystem CCFs, structural failures, failure modes and dependencies.
- There were systems that should be available for mitigation omitted from the PSA (non-realistic).
- The modelling of the containment isolation failure in the level 2 IEAP PSA did not fully address CCFs and latent failure modes.
- There was a lack of clarity and justification regarding the characterisation of the digital control and instrumentation (C&I) failure modes, CCFs, human interface failures, software failures, reliability data, etc. The impact of dependencies due to the connection of instrumentation to the RPV was also not considered in the PSA.
- The PSA did not include support system initiating event fault trees.
- Some of the 'latent' failure mode probabilities in the PSA were calculated assuming a 24 hours mission time duration instead of considering the standby exposure period (eg test interval).
- The impact of an initiating event on the systems was not explicitly discussed as part of the system analysis.
- Dependency analysis was not captured in a single location which was judged useful to the PSA team, reviewers, and as a communication tool.
- The documentation regarding the use of house events, flags, and mutually exclusive files was not available.

171. To address these concerns I raised RO-ABWR-0053 (Ref. 48) and a number of RQs (Ref. 94).

172. My detailed review of the system analysis element of the UK ABWR PSA in Step 4 has been performed following on from the review I conducted in Step 3 by looking at the application of the methods and techniques applied to several example systems.

173. The systems selected for detailed review were: reactor core isolation cooling system (RCIC), low pressure core flooder system (LPFL), safety relief valves (SRVs) / automatic depressurisation system (ADS), reactor building cooling water system (RCW) / reactor building service water system (RSW), class 1 AC power system and C&I systems. In addition, a high-level review of heating, ventilation and air conditioning system (HVAC), flooding system of specific safety facility (FLSS), flooding system of reactor building (FLSR) and vapour suppression system models were also undertaken.

174. The above selection have a high importance in the PSA and provided a good representation of all the types of systems and components in the UK ABWR and would ensure that the review would cover the various aspects of system performance modelled in the PSA in a comprehensive manner.

175. In response to RO-ABWR-0053, Hitachi-GE submitted an updated IEAP PSA model including revised systems analysis at the end of Step 3. As part of my review in Step 4, I raised several RQs (Ref. 94) and held technical workshops with Hitachi-GE in March and October 2016 (Ref. 79). As a result, Hitachi-GE updated the PSA in January 2016 and June 2016, and the documentation was also updated in March 2017.

176. In addition, Hitachi-GE provided sensitivity analyses to evaluate the impact of some of the remaining shortfalls that were not considered in these updates (Ref. 5).

177. I have considered the adequacy of these sensitivity analyses as part of my review. I have used some of the insights of these analyses, in combination with qualitative arguments and quantitative information from the PSA, to understand the potential risk significance of the findings in this area.

178. My assessment was performed using the latest available models and documentation provided by Hitachi-GE, which are described in Sections 2 and 3 of this report. My assessment was carried out against the expectations in ONR's PSA TAG (Ref. 4).

4.2.7.2 STRENGTHS

179. In response to RO-ABWR-0053, Hitachi-GE developed a system analysis approach that meets many of the expectations of ONR's PSA TAG (Ref. 4).

180. For the cases reviewed I have reached the following general conclusions, with some exceptions noted in the next section:

- The approach used for the definition of system boundaries, inclusion of failure modes and unavailabilities is transparent and adequate.
- The level of detail of the system fault tree models is sufficient to ensure they are realistic and most of the dependencies are captured.
- The models reviewed are correct and result in MCSs for failures of the systems that reflect combinations of failures that can be easily understood.
- The data used is applicable to the boundary selected for each component basic event in the PSA.
- The level of detail of the fault trees is consistent throughout the system analysis.

4.2.7.3 FINDINGS

181. The detailed technical review that supports the findings discussed in the following paragraphs is documented in Ref 90.

182. The review in Step 4 has concluded that the following general concerns are still outstanding and should be addressed following GDA. A summary is presented in this section.

183. Ref. 103 describes the general approach to the treatment of test and maintenance unavailabilities. The assumptions used are generally well recorded, however the review has found some exceptions:

- C&I and HWBS regarding maintenance unavailability;
- RCIC, LPFL and RCW/RSW regarding component availability during the tests.

It is important to note that, due to the early stage of the project, the basis supporting these assumptions, such as detailed technical specifications and a T&M schedule, were not (and were not expected to be) available (see Section 4.2.9).

184. On the basis of the sample considered, my review revealed that most of the system fault trees are sufficiently detailed and include all critical components and critical failure modes. However, my review team found the following failure modes are not considered in the analyses:

- intersystem CCF, with the exception of the ECCS suction strainer plugging (see Section 4.2.9.3.4 for further details);

- failure of the alarms, indications or other hardware failures that could contribute to the human failure events. To justify this exclusion, Hitachi-GE provided a qualitative review of the available instrumentation that shows there are several independent indicators available to the operators (Ref. 104);
- mechanical CCF of MSIVs to close on demand;
- failure of flow diversion valves for the RCIC and LPFL;
- RCIC components may be challenged a number of times during an event response. The RCIC pump modelling includes a restart basic event, but multiple challenges during a fault sequence for other components do not appear to be considered in the PSA model.

In addition, the general approach applied for the inclusion of structural failures into the system models is not clear (see Section 4.2.9 for further details).

185. The failure probabilities associated with the missing failure modes are expected to be low, and therefore have a small contribution to the risk profile. For the case of multiple demands on RCIC components, Hitachi-GE has performed a sensitivity analysis to evaluate the risk impact and demonstrated it to be small. However, all of these failure modes should be explicitly considered in the PSA model as they can be important for future PSA applications.

186. The system models generally do not include supercomponents to subsume more detailed logic. Exceptions to this are the FLSS support systems (such as back-up building emergency equipment cooling water) and FLSR. These systems are expected to be designed in a way such that dependencies are limited. When additional information becomes available, the PSA model for FLSS support systems and FLSR should be developed.

187. Some of the fault trees have asymmetries artificially built in. For example, the same RCW/RSW trains are assumed to be normally operating. This creates an asymmetry in the calculated importance measures. Upon request, Hitachi-GE has provided additional information that has been sufficient to understand the extent of the asymmetries during GDA. However, artificial asymmetries should be removed beyond GDA to ensure the suitability of the PSA to support applications based on evaluation of the results of importance analysis and risk monitoring.

188. Upon request, Hitachi-GE has developed a dependency notebook (Ref. 105) with the aim of capturing dependent interfaces that can substantially increase the risk of the plant. However, my review has identified some missing dependency analyses. A summary of these shortfalls is provided in the following paragraphs and includes issues related with room heat up calculations and adverse environmental conditions.

189. Room heat up calculations have been performed with the GOTHIC computer code and appear to reasonably represent the 'as-designed' plant. However, there is limited discussion regarding how these calculations justify the PSA modelling. In particular, there is no acceptance criteria method for room temperature and no linking of the calculation to equipment qualification limits, including:

- The maximum room temperature limit for the LPFL/RHR pump rooms is not established.
- Cooling requirements for the HWBS are not considered in the PSA. Hitachi-GE has indicated that the reason for this is that no large heat sources have been identified in the rooms where the HWBS equipment is located. However heat up calculations need to be performed to demonstrate this assumption, following completion of the HWBS design.

- The PSA assumes that RCIC operation does not require the HVAC or RCW/RSW operation for 8 hours. However, analysis to demonstrate the system capability without these support systems is not provided. Furthermore, the operability limit for the RCIC room identified in Ref. 104 is lower than the room temperature that Hitachi-GE has shown can be reached within 8 hours without room cooling operating. Hitachi-GE has performed a sensitivity study to show that the impact on the risk profile would be small, with an increase in core damage frequency (CDF) of 5%, if the RCIC was only operable for 4 hours.
- Ref. 104 provides some operational limits for the AC system; primarily temperature and humidity however it is not complete. Furthermore, the operability limit identified by Ref. 104 is lower than the temperature that Hitachi-GE has shown can be reached within 8 hours according to the room heat up analysis.
- The impact of ambient temperature on the heat up calculations and system level success criteria is not clear.

It is not clear that temperature limits have been adequately considered in the PSA, and further substantiation is required beyond GDA to demonstrate that the assumptions made in the PSA reflect the plant.

190. Hitachi-GE considers that adverse environmental conditions are bounded by the consideration of loss of room cooling. However, my review has found the following conditions which have not been analysed in detail and could be more severe than loss of room cooling:

- Environmental conditions after containment failure or high energy line breaks (HELBs) outside containment may compromise equipment availability. Due to lack of detailed design information Hitachi-GE's current evaluation is based upon assumptions (see Section 4.2.5 for further details and estimation of the impact on the risk profile).
- The impact of high temperature of the suppression pool (S/P) on the RCIC pump is assumed to be bounded by the RCIC loss of function due to the pump room temperature but underpinning analyses are not available at this stage due to lack of detailed design information.
- There is no consideration that debris, either internal or external to the system or plant, could block screens or filters (with the exception of suppression pool suction strainers being explicitly modelled).

191. The revised systems analysis report was sufficiently detailed for my review in GDA. In some cases I requested additional information via RQs that was subsequently added to updated revisions of the report. However, due to the way in which the information is structured, the systems analysis report is generally not easy to use. Furthermore, my review has identified specific areas that require enhancement that are summarised below:

- description of system operating and shutdown modes (including abnormal operations);
- detailed information regarding the success criteria of front line and support systems (currently in the accident sequence report (Ref. 49) and dependency matrices topic (Ref. 105) reports);
- all relevant system descriptions should include any secondary functions of the systems and provide a basis for why these functions are or are not included in the PSA. Discussion is included (Ref. 104) for a limited number of systems, but this should be expanded. The PSA model should be updated to reflect secondary functions as appropriate. Functionality during non-power POSs should also be discussed;

- detailed description of the system boundary and interface between systems; only high-level summaries of the interface between systems are currently provided in Ref. 104;
- the PSA documentation did not originally provide a clear discussion regarding circular logic. Hitachi-GE provided additional information as to how circular logic has been addressed in the PSA model in response to an initial query. Ref. 104 was subsequently updated to include these details for the C&I systems, however this was not extended to other support systems such as HVAC, RCIC, class 1 AC system or SRVs, which can also present the same issues.

This has been captured as a minor shortfall.

192. My review of the PSA models for the SRV/ADS system has raised a number of concerns of a different nature. The main limitations identified are discussed below:

- The PSA assumes that opening the SRVs does not require the pneumatic pressure provided by the high pressure nitrogen gas supply system (HPIN). However, this assumption has not been fully justified for all scenarios in the PSA. In particular, the failure to vent the containment would lead to a pressure rise which could result in closure of the SRVs and the pressure provided by the HPIN may be needed to keep the SRVs open (also discussed in Section 4.2.18).
- There is an implicit assumption in the PSA that long term nitrogen makeup to ensure the operability of the SRVs is not required. However, no evidence has been presented to support this claim. Hitachi-GE performed a sensitivity study which showed that the increase in LRF is negligible if long term nitrogen make-up is included in the model. It should be noted that this study relies on the PSA model being updated to reflect the final GDA design of the nitrogen accumulators.
- CCF between the ADS and RDCF SRVs is not considered. Upon request, Hitachi-GE undertook a sensitivity analysis to expand the CCF between ADS/SRVs and diverse ADS. Because of the large numbers of SRVs the impact of this CCF on CDF and LRF was small.

193. In relation to the digital C&I system, my review team identified a specific limitation with regards to the modelling of the bypass of trip and sensor signals, which presumably would be aligned to permit plant testing and/or maintenance. Upon request, Hitachi-GE has presented information to justify some of this treatment (Ref. 106) and this justification has been captured in the PSA documentation (Ref. 104). Consideration of how maintenance will impact the system availability should also be considered and explained.

194. In relation to the class 1 AC power system, my review team raised the following additional specific comments:

- There are missing failure modes in the fault trees. In particular, coordination failure where a fault on a component propagates a fault to higher level equipment is not addressed. This scenario is likely to be bounded by the ‘circuit breaker spuriously opens’ failure mode, however it is not discussed and included as an assumption in the Hitachi-GE documentation. Hitachi-GE indicates that the PSA has considered that this failure mode cannot occur as the properly sized and coordinated electrical protective devices are assumed to function in accordance with their design. This assumption is consistent with the ‘electrical protection and earthing system study report’ which has been reviewed by ONR’s electrical engineering inspector (Ref. 107). Once sufficient information on the design is available this assumption will need to be reviewed and an update to the PSA performed, if required.

- Ref. 104 does not explain the system response to plant trip. In particular, it is not clear whether there is a 'fast transfer' or if the generator transformer responds in some way to allow off-site AC power to maintain supplies to the auxiliary normal transformer. The PSA currently does not credit any of these features.

195. The systems analysis is a large and complex topic area, the number of shortfalls presented in this section appears large, but compared with the size and complexity of the systems analysis they do not represent a significant concern. The majority of the shortfalls identified are related to the need to provide further justification to underpin the assumptions in the PSA once detailed design information becomes available and the need to improve the PSA documentation.

196. My evaluation of the findings in this particular area is based on the PSA results and importance measures and has considered the following sensitivity analyses undertaken by Hitachi-GE:

- The change in risk if, following loss of class 1 AC power, operation of the RCIC is assumed to terminate after 4 hours or 14 hours, as opposed to the base case assumption of 8 hours. The study showed a small increase in risk if the RCIC operation time were reduced and a small decrease in risk if it were increased.
- The change in risk if HPIN is assumed to be necessary for long term SRV operation. The study showed a negligible impact on the risk if the PSA considers the end of GDA nitrogen accumulator design.
- The risk impact due to consideration of multiple demands on valves, relays, etc. This sensitivity study assumes that there would be a small increase in failure probability of each component, proportional to the reliability of an operator action to control RPV water level to reduce the number of component demands. The study showed the risk impact to be negligible.
- The change in risk if a CCF between ADS and RDCF were included in the PSA. The study showed a small increase in core damage frequency (CDF) and a negligible increase in large release frequency (LRF).
- The change in risk if a CCF between the EDGs and BBGs was included in the PSA model. The study showed a small increase in LRF (2.5%).

197. The sensitivity analyses and review of the PSA results and importance measures show that the shortfalls identified are expected to have a small impact on the risk profile of the UK ABWR. However, the PSA is expected to be fully substantiated and representative of the plant.

4.2.7.4 CONCLUSION

198. Based on the outcome of this assessment and the sensitivity analyses performed by Hitachi-GE (which provided clarity on the impact of the key limitations identified), I concluded that the system fault trees developed by Hitachi-GE, in response to RO-ABWR-0053, are sufficient to close the regulatory observation (Ref. 108) and to support the UK ABWR 'generic' PCSR.

199. However, this part of the UK ABWR PSA needs additional enhancement to support further stages of the NPP development. Further substantiation of some of the assumptions related to the UK ABWR design is expected once the information becomes available beyond GDA. This should include further analysis of dependencies and environmental conditions. This is expected to take place as part of resolution of AF-UKABWR-PSA-001 to address any shortfalls identified in this report, or as part of

normal business to update the PSA to reflect site specific aspects or design development. The systems analysis documentation also requires enhancement to aid usability.

200. It should also be noted that there may be limitations in other system fault trees that were not sampled in GDA, therefore the totality of the UK ABWR PSA fault trees should be reviewed and revised as appropriate beyond GDA.

4.2.8 Level 1 PSA: Human Reliability Analysis (A1-2.5)

4.2.8.1 ASSESSMENT

201. ONR's human factors (HF) assessment team conducted an initial review of the UK ABWR PSA task on 'Human Reliability Analysis (HRA)' against the expectations in ONR's HRA (Ref. 39) and PSA (Table A1-2.5 (Ref. 4)) TAGs in Step 3. The more detailed conclusions of the review are documented in Ref. 109. The review identified some concerns regarding the method that Hitachi-GE intended to use for generating human reliability assessments. Overall, the ONR's HF review concluded that Hitachi-GE appeared to have an adequate capability in the area of HRA which was reflected in the HF submission in Step 3.

202. ONR's HF assessment team has conducted detailed reviews, in Step 4, of Hitachi-GE's HRA submission (Ref. 110) against ONR's HRA TAG (Ref. 39) and followed up the concerns raised in Step 3. The scope of ONR's HF assessment has considered the HRA for the PSA for: the reactor, spent fuel pool (SFP), internal fire, internal flooding and fuel route, including:

- the methodologies selected for the HRA, and in particular for the evaluation of human error probabilities (HEPs), including the choice of human reliability data sources;
- the treatment of misdiagnosis and other cognitive failures;
- human reliability quantification method(s) and implementation;
- task analysis supporting the HFEs in the PSA model;
- consideration of facility and HFE specific influences of the factors required by the quantification model (performance shaping factors (PSFs));
- adequacy of the justification of available time for action (time windows), including adequacy for the choice of events that mark the start and end of the time windows, estimation of task action times and time spent on other tasks;
- adequacy in the identification and probabilistic estimation of dependencies.

203. During Step 4, my review team has supported the assessment undertaken by ONR's HF inspector of the UK ABWR PSA HRA as follows:

- reviewing the completeness of the HFEs included in the logic model structure;
- consideration of how alarms and other cues required for human actions are represented in the PSA;
- identification of dependencies between HFEs appearing in the same accident sequence.

Due to the submission of Ref. 110 late in GDA Step 4, the assessment supported by the PSA team was conducted on the previous revision of the submission (Ref. 111). Hitachi-GE has informed ONR that some of findings identified in the following subsections have addressed in Ref. 110.

204. Hitachi-GE's HRA has been developed in parallel to the UK ABWR PSA and revised several times to take into account ONR's HF review comments. The PSA therefore

does not reflect the latest HRA analysis. Instead, Hitachi-GE has provided a sensitivity analysis to consider the impact of the revised HRA on the risk profile. I have considered the adequacy of this analysis as part of my review.

205. The outcomes of my review are reported below.

4.2.8.2 STRENGTHS

206. The methodology selected for the HRA and its implementation meets most of the expectations in ONR's HRA TAG (Ref. 39), with well documented task analysis and structured methods for dependency identification.

207. The PSA includes pre-accident HFEs (eg individual and common cause component misalignments and miscalibration of instrument and protection channels), post-initiating event HFEs (detection, decision errors, omission errors, etc) and some HFEs associated with initiating events.

4.2.8.3 FINDINGS

208. Hitachi-GE has extended the PSA documentation to record the justification for the pre-accident human failure events included and not included in the PSA. This work should be revised once the maintenance and operation procedures are developed. For example, my review has identified an example of misalignment considered in the model which contributes significantly to the failure probability of the RCW/RSW, but could be avoided as it would result in unusual system parameters.

209. In addition, the treatment of CCFs between pre-accident human failure events has not been justified. My review of the SFP PSA results has identified HFEs in the same minimal cutsets (MCSs) with no justification to demonstrate the independence between these events.

210. My review of the level 1 IEAP PSA has also raised some concerns regarding the approach used for the inclusion of post-accident human failure events into the system models and the treatment of dependencies in the accident sequences. These are linked to different technical areas of the PSA (reported in more detail in other sections of this report) and include the following:

- Cases in which operator action for the control of water inventories is needed but is not considered in the related PSA. The same review also raised questions related to the manual initiation of the emergency core cooling system (ECCS) and its dependency treatment (see Sections 4.2.5 and 4.2.6).
- Cases for which modelling of operator actions for manual initiation of a safety system could have been considered, given failure of automatic signals, but were dismissed due to: model simplification, lack of information or feasibility arguments. For example, review of the C&I system level cutsets identified that manual initiation of ECCS is not credited given failure of the automatic signal.
- Hitachi-GE appears to have not adopted the Boiling Water Reactor Owners Group (BWROG) emergency procedure guidelines (EPGs). For example, the BWROG EPGs identify that the CST is the preferred suction source for the high pressure core flooder system (HPCF) and the reactor core isolation cooling system (RCIC) and that consideration should be given to operator actions to defeat high S/P water level suction transfer logic if necessary. Hitachi-GE clarified that this direction is not included in J-ABWR emergency operating procedures (EOPs) (and consequently is not assumed for UK ABWR EOPs at this stage), but agreed that the guidance provided in the BWROG EPGs should be considered in the future.

- Although the UK ABWR specific procedures are not available, the PSA currently assumes that power conversion system (PCS) recovery will be a procedurally directed action. This operator action is claimed in the accident sequences having long term decay heat removal failure. However, there is a lack of clarity regarding: the overall time needed to complete the required tasks, the potential for errors and delays in the use of the PCS under these adverse conditions and equipment needed to reopen the MSIVs (eg MSIV accumulator, HPIN or mechanical vacuum pump). Hitachi-GE has identified that long timescales are available to complete this action. However, further information will be required to underwrite this assumption beyond GDA.
- The treatment of dependencies between the operator action to switch the HPCF suction from the suppression pool to the CST and, to open the vent before HPCF failure on high suppression pool temperature is not fully justified. Hitachi-GE argues that the procedure will direct containment venting between 1 times design pressure (Pd) and 2 Pd regardless of the injection system and water source. Therefore, Hitachi-GE considers that this action is independent from preserving the HPCF. However, no evidence or supporting analysis has been provided at this stage.
- The PSA documentation does not provide sufficient clarity regarding the derivation of timing estimates for operator actions. In particular, specific operator actuations required and latest time for manual actuation which can lead to success are not identified. This information has been provided separately and it has been assessed by ONR's HF inspector (Ref. 112). The outcome of this review has identified that task time estimates are conservative. When sufficient information regarding operating procedures and practical task simulation becomes available beyond GDA, the grace and task timing should be revised to ensure they are 'best estimate'. The PSA should be revised to reflect 'best estimate' and adequately justified operator action time windows. This update should be supported by documentation detailing how operator delay time, response time and manipulation time have been accounted for.
- An assessment of the heavy load drop frequency derivation was undertaken by Hitachi-GE's HF team and reviewed by ONR's HF inspector. However, the frequencies in the fuel route PSA are not aligned with these analyses. The heavy load drop frequencies in the PSA should be revised to reflect the human factors analysis, and take detailed design information and procedures into account when this information becomes available.
- Hitachi-GE did not have sufficient information to undertake a systematic examination of the UK ABWR procedures for changing configurations, equipment testing and maintenance procedures to identify potential human errors during the execution of such normal procedures that are, or may lead, to initiating events. This review should be undertaken and the list of initiating events potentially expanded to include additional HFEs that can lead to initiating events during the site specific stage.

211. As part of normal business beyond GDA, further substantiation of the operator actions will also be required to reflect the detailed design information (eg specific cue or conditions that support operator actions) and procedures when this information becomes available.

212. My review of the interface between the level 1 and level 2 IEAP PSA has identified that not all the dependencies with the HFEs in the level 1 IEAP PSA are treated appropriately. Between level 1 and level 2 IEAP PSA models only the following two dependent actions have been identified:

- RPV depressurisation

- FLSS initiation and RPV depressurisation

Further work has been conducted by Hitachi-GE in Revision F of the human reliability assessment report (Ref. 110), however due to the submission of this revision occurring late in GDA Step 4 it has not been assessed as part of the PSA assessment. Additional analysis detailed in Revision F of the human reliability assessment report (Ref. 110) has not been implemented into the PSA models.

213. The documentation does not provide a sufficiently detailed discussion of other PSA actions such as: RPV injection (ECCS), off-site AC recovery, drywell sprays, FLSR, venting, RHR or competing actions of where to inject water (RPV or containment), or dependencies with the level 1 PSA. A robust discussion of the treatment of operator actions in level 2 PSA along with the dependencies accounted for in the model should be provided.

214. My review of the level 2 PSA has identified that only a limited set of HFEs are included; some examples of missing HFEs are:

- Errors of commission (EOCs).
- The potential for adverse effects of severe accident management actions.
- Drywell venting (filtered or unfiltered). This may lead to a different type of release than containment failure.
- Reactor well flooding above the drywell head to mitigate the potential for elevated temperatures in the drywell head region. The benefit of this action has been considered in the containment performance analysis but is not reflected in the PSA.
- Failure to reclose the containment vent after containment venting, which may lead to the inerting of containment being lost when the reduction in decay heat leads to a reduction in steam generation (which could lead to accumulation of hydrogen). It is acknowledged that MAAP analysis shows that the release is dominated by the fission products released at the initial venting time and that any additional release later as a result of containment failure would be small.
- Coordination of external water injection and containment water level control. For example the PSA does not consider scenarios where the operator fails to terminate FLSS injection when high suppression pool (S/P) water level is reached.
- Drywell spray for radionuclide release mitigation for temperature and temperature control.

215. Additional review comments related to HRA are provided in Sections 4.2.11, 4.2.12 and 4.2.14.

216. As noted previously, the initial HRA values used in the PSA have been revised by Hitachi-GE's HF team using detailed task analysis and human reliability quantification where HRAs were identified to be risk significant. With exception of the internal fire and flood PSA refinement (see Section 4.2.11 and 4.2.12), the PSA has not been updated to include the latest revisions of the HRA. A sensitivity study performed by Hitachi-GE identified that the risk predictions from the PSA (in particular the shutdown and SFP PSA) would increase significantly if the refined HRA were taken into account, however their risk predictions would still remain relatively low and would still contain significant conservatisms, as discussed in Sections 4.2.15 and 4.2.16.

4.2.8.4 CONCLUSION

217. Based on the outcome of this assessment and the human factors assessment, I have concluded that the current HRA supplemented by the internal hazards PSA refinement

and the IEAP sensitivity analysis to reflect the most recent HRA values is sufficient to support the UK ABWR 'generic' PCSR.

218. However, this part of the UK ABWR PSA needs to be updated to reflect the detailed design, operating procedures and severe accident management guidelines (SAMGs), including a potential extension of the PSA to consider potentially omitted operator actions and the treatment of dependencies between the level 1 and the level 2 PSA to better reflect the UK ABWR risk profile. This is expected to take place as part of resolution of AF-UKABWR-PSA-001 to address any shortfalls identified in this report, or as part of normal business to update the PSA to reflect site specific aspects or design development

4.2.9 Level 1 PSA: Data Analysis (A1-2.6)

4.2.9.1 ASSESSMENT

219. I conducted a high-level review of the UK ABWR PSA task on 'Data Analysis' against the expectations in the ONR's PSA TAG (Table A1-2.6 (Ref. 4)) during Step 3. This review raised specific and general concerns (documented in Ref. 77). Some examples are provided below:

- Issues were identified with the initiating events frequencies of LOOP, LOCA, BOC and units of IEs.
- The PSA failure data was based on the 2007 NUREG/CR-6928 (Ref. 99) instead of the more recent update in 2010 (Ref. 113).
- There were limitations noted regarding the C&I data.
- There was a lack of consideration of coincident maintenance unavailability.
- CCF groups and failure modes were not always complete.

220. My review findings were captured in RO-ABWR-0042 and RO-ABWR-0053. In response, Hitachi-GE provided an updated level 1 IEAP PSA in September 2015. My detailed review of the data analysis element of the updated UK ABWR PSA in Step 4 has been performed following on from the review conducted in Step 3 by looking in detail at a large sample of the reliability data used in the PSA.

221. In addition, ONR has commissioned two technical support contracts to review available European and North American operational experience of boiling water reactors (BWRs) (Refs 114 and 115). ONR also raised RO-ABWR-0045 (Ref. 116) to require Hitachi-GE to review the two reports, produce a similar report on relevant Japanese operational experience and demonstrate how the UK ABWR address the issues raised. In Step 4, I have reviewed all the relevant reports to ensure that the PSA fully takes into account the lessons learned from the relevant operational experience.

222. My review of the initiating event frequencies covered the derivation of LOCAs, a wide range of loss of support systems (RSW, RCW, C&I, instrument air system, DC electrical system), ATWS and LOOP.

223. My review scope has also included the random component failure data, maintenance unavailabilities, the methods used to calculate the CCF probabilities as well as the CCF parameters. My review has examined the specific values used in the UK ABWR PSA (with almost full coverage) as well as any relevant assumptions and calculations performed in support of the data analysis.

224. As part of my review in Step 4, I have raised several RQs (Ref. 94) that Hitachi-GE has mostly addressed through a PSA model and documentation update in January 2016, June 2016 or a documentation update in March 2017. The extended PSA information has included improvements in the documentation.

225. For the remaining shortfalls, I have requested Hitachi-GE to perform sensitivity analyses to evaluate the impact on the risk. These sensitivity analyses were provided in Ref. 5. I have considered the adequacy of these analyses as part of my review. I have used some of the insights of these analyses, in combination with qualitative arguments and quantitative information from the PSA, to understand the potential risk significance of the findings in this area.

4.2.9.2 STRENGTHS

226. In response to my Step 3 review, Hitachi-GE updated the criteria for selection / precedence of data sources for the PSA to reflect more recent generic data sources.

227. The basis for the selection of the reliability data values assigned to most basic events and modules in the UK ABWR PSA have been documented.

228. The component populations together with their characteristics are clearly identified. The component boundaries are clearly stated and align with the data sources selected. Assumptions regarding unavailability time are stated and are reasonable.

229. The approach selected for the intrasystem CCFs modelling and analysis is justified, with generic industry data used appropriately.

4.2.9.3 FINDINGS

230. The detailed technical review that supports the findings discussed in the following paragraphs is documented in Ref 90.

4.2.9.3.1 Initiating Event Frequencies (A1-2.6.1)

231. For LOCA events my review of the initiating event frequency derivation identified that different size LOCAs were not explicitly considered on every size of pipe; for example, large LOCAs were considered for large bore pipework, but small and medium LOCAs on the same pipework were not. Upon request, Hitachi-GE demonstrated that their approach produced results that bounded these events, with the overall LOCA frequencies in line with those presented in NUREG/CR-6928 (Ref. 99). In the future the PSA should be revised to adequately reflect all the relevant sizes of LOCAs.

232. The LOOP frequencies currently included in the PSA are based on United States (U.S.) operational experience. I have reviewed these values in coordination with ONR's electrical engineering inspector and considered them, in the absence of site specific data, adequate for GDA. However, these frequencies will have to be updated with site specific data. This determination also extends to the use of conditional LOOP and LOOP recovery (external to site) probabilities currently credited in the PSA.

4.2.9.3.2 Random Component failure Probabilities (A1-2.6.2)

233. The Step 4 UK ABWR PSA uses the 2010 version of NUREG/CR-6928 (Ref. 113) as the main data source which has overall been considered adequate for GDA. However, my review has identified the following limitations associated with the use of these data for the UK ABWR PSA:

- The review identified that the treatment of testable check valves in the PSA may be optimistic. Hitachi-GE has informed the design team to ensure that the detailed design of these valves is optimised. Once the design is known, substantiation of the probability of failure of these valves should be provided and the PSA should be aligned with the most representative data.

- The component failure data does not consider the potential variation in reliability of mechanical components with respect to the water quality of the system (for example, demineralised versus sea water).

234. The data notebook (Ref. 103) provides the failure rate of a number of digital components. However, no specific supporting reference is provided. My review team noted that the failure rates used are higher than those provided by industry sources; including those cited in Ref. 103. Furthermore, the data used for the class 1 C&I platform is not consistent with the data used in the C&I documentation (Ref. 117). Upon request, Hitachi-GE produced a sensitivity study which confirmed that the data used in the PSA was conservative. Future iterations of the PSA model should use justified 'best estimate' data that adequately reflects the UK ABWR design.

235. My review identified an item regarding the ECCS suction strainer reliability. It was initially considered that the reliability figures could be optimistic when compared to the reliability data used in other reactor designs' PSAs, especially for LOCA events. Hitachi-GE indicated that the design of the ECCS strainers in the UK ABWR is an improvement with known issues being designed out. In addition, Hitachi-GE performed some sensitivity analyses to account for the potentially degraded reliability of these strainers during LOCA scenarios. The results of these analyses indicated that the PSA risk predictions only have a small sensitivity to the reliability data of the ECCS strainers. Substantiation of these claims and reliability data used in the PSA should be provided in the site specific stage.

236. Ref. 103 documents how time periods have been defined and applied to the failure data to establish the correct probability of failures on demand. This should be revisited at a later stage of design to ensure the testing regime assumed in the PSA matches with the site specific maintenance plans (e.g. T&M schedule).

237. The review identified that there was not a clear auditable trail for the identification of component failure modes within Ref. 103. This has been captured as a minor shortfall However, the review noted that FMEAs had been produced in support of the IE identification. Additionally my review team noted that all significant failure modes were included with the following exceptions:

- There is no evidence in the documentation that structural failure modes of active components have been considered. It is acknowledged that the impact on the PSA results will be limited as the active failure modes are expected to be the dominant contributors to the component unavailability.
- The review noted that the traceability of the consideration of latent failure modes had improved since Step 3. However, an issue was identified relating to the latent failures that had been screened out on the basis that they would be 'immediately revealed'. For these cases it was unclear whether this would be revealed due to the loss of a function that would manifest itself though several means (but not cause an initiating event), or by a specific detection system. These should be modelled in the PSA when information becomes available.

4.2.9.3.3 Unavailabilities Due to Testing and Maintenance (A1-2.6.3)

238. The testing and maintenance (T&M) unavailabilities are based on assumptions listed in the PSA documentation that should be confirmed or revised when information regarding the inspection and T&M schedules of the UK ABWR is developed beyond GDA.

239. The review has also identified that T&M unavailabilities were not considered for standby components where an unavailability time is not currently defined. Once the technical specifications are available for these systems their maintenance unavailabilities should be incorporated into the PSA.

***Assessment Finding AF-UKABWR-PSA-002**: The licensee shall ensure that the basis for the modelling and assumptions concerning outage, maintenance and test unavailabilities of systems and components (including standby) used in the PSA, is justified and aligned with the technical specifications and maintenance programmes, or alternative values / strategies justified.*

4.2.9.3.4 Common Cause Failures (CCFs) (A1-2.6.4)

240. My review has identified that the approach selected for modelling CCFs does not address intersystem events. Upon request, Hitachi-GE has provided additional analyses with the objective of justifying that intersystem CCFs do not need to be considered in the UK ABWR PSA. These analyses were reviewed with the following issues raised:

- Hitachi-GE has performed a review of the MCS to identify intersystem CCF candidates. However, a re-quantification of the MCS with a sufficiently low cut-off has not been performed. It is considered that several credible CCF combinations may be found in MCS with low failure frequencies below the cut-off used.
- The consideration of diversity between the identified intersystem CCF candidates has not been performed at a sufficient level of detail to provide confidence that the components were sufficiently diverse to exclude an intersystem CCF from being modelled.
- Specific concerns were raised regarding the potential for intersystem CCFs between the EDGs and BBGs. Hitachi-GE provided assurances regarding the planned diversity between these components such that they can preclude the potential for a CCF. A more detailed justification would need to be provided, including consideration of sub-components, at a later stage of the design to ensure that the current diversity claims are upheld.

241. These shortfalls have resulted in the following assessment finding being raised:

***Assessment Finding AF-UKABWR-PSA-003**: The licensee shall use the PSA to identify intersystem common cause failure effects for the UK ABWR following on from the work in GDA. The results shall be used to inform the incorporation of appropriate defences and, where appropriate, intersystem common cause failures should be included explicitly in the model.*

242. My review has also identified the following cases for which the approach selected for intrasystem CCF modelling is not justified:

- The mechanical CCF of the control rod drive (CRD) system is not considered in the PSA. However, the review noted that failure to SCRAM due to other failures relating to the CRDs had been considered. In response to this concern, Hitachi-GE performed a sensitivity study; however I consider the study to be incomplete as only CRD drives were considered as opposed to the complete system.
- There is no explicit consideration of CCFs of the components of the reactor vessel instrument system (RVI) lines. My review of a conservative sensitivity study, produced in response to this concern, revealed that there were some significant risk insights in this area. Hitachi-GE is taking account of the insights identified with respect to future plans for diversity, maintenance and operating procedures. The PSA initiating events and systems analysis should be updated to account for these missing failure modes once sufficient design information is available.

- The probability of CCF of the digital C&I has not been substantiated. This could have an impact on the perceived importance of systems on the HWBS. For example, my review identified that the PSA may underestimate the benefit of the train of HPCF on the HWBS due to the digital C&I CCF data being potentially optimistic. The CCF data used should be substantiated and the model revised as appropriate to adequately represent the potential CCF of the digital C&I.

4.2.9.4 CONCLUSION

243. Whilst a number of shortfalls have been identified, CCF analysis is a complex topic with large uncertainties and is often reliant upon detailed design and operational information that are not expected to be available at GDA. ONR's review in this area has been extensive, with multiple cross-discipline interactions with Hitachi-GE. My evaluation of the findings in this particular area on the basis of Hitachi-GE sensitivity analyses has shown that none of the shortfalls identified would lead to a significant increase in the risk results, as long as the assumptions made in the PSA model are adequately substantiated by the detailed design and operation of the UK ABWR.

244. Based on the outcome of this assessment, I have concluded that overall the current UK ABWR PSA reliability data is sufficiently adequate to support the UK ABWR 'generic' PCSR.

245. However this part of the UK ABWR PSA needs some enhancements to support further stages of the NPP development, including better substantiation of the data used for some of the modelled failure modes, a potential extension of the consideration of CCFs and the substantiation of some of the claims made in the PSA. This is expected to take place as part of resolution of AF-UKABWR-PSA-001 to address any shortfalls identified in this report.

4.2.10 Level 1 PSA: Analysis of Hazards – Prioritisation of Internal Hazards (A1-2.7-1)

4.2.10.1 ASSESSMENT

246. Prioritisation of internal hazards is undertaken to enable ranking all possible applicable internal hazards according to their potential risk significance in order to identify those for which detailed PSA modelling and evaluation is warranted. An initial review of the UK ABWR PSA prioritisation of internal hazards against the expectations in ONR's PSA TAG (Ref. 4) was conducted in Step 3. This review (Ref. 77) identified the following shortfalls:

- The initial list of hazards identified for the UK ABWR PSA was not comprehensive when compared to international literature and did not include combinations of hazards.
- The approach and criteria for prioritisation of hazards presented was not adequate.
- Treatment of beyond design basis hazards was not clear.
- The scope of the hazards analysis did not address plant states other than at power conditions with the containment inerted or other on-site radionuclide sources apart from the reactor.
- There was a lack of clarity regarding references to key hazard information including the characterisation of the hazard and the justification of the criteria used to establish the design attributes.
- The hazard impact analyses to support the screening were not provided.

247. In view of these Step 3 findings, RO-ABWR-0040 was raised. The objective of RO-ABWR-0040 was to state ONR's expectations relating to the identification of

internal hazards for the UK ABWR PSA and request Hitachi-GE to respond to the shortfalls identified by ONR's review.

248. I assessed Hitachi-GE's response to RO-ABWR-0040, and other internal hazard PSA submissions, during Step 4 with the support of ONR internal hazards inspectors. The assessment focused on the following, in line with the expectations set out in RO-ABWR-0040 and the ONR's PSA TAG (Ref. 4):

- completeness of the internal hazards considered for prioritisation;
- the approach and criteria for prioritising the identified hazards;
- consideration of combinations of hazards; and
- adequacy and scope of the hazard impact analysis performed.

249. Hitachi-GE's primary response to RO-ABWR-0040 was the topic report on internal hazards prioritisation (Ref. 20). This topic report was produced by Hitachi-GE over a number of revisions; my review mainly considered revision 2 (Ref. 118). Shortfalls identified by my review team were communicated to Hitachi-GE in a number of RQs (primarily RQ-ABWR-1102 and RQ-ABWR-1243). The final submission for GDA is revision 3 (Ref. 20). I reviewed revision 3 to determine if shortfalls identified in earlier revisions had been addressed.

250. The topic report on internal hazard prioritisation (Ref. 20) identified, following qualitative and quantitative screening, the requirement for detailed PSA studies to be performed within GDA on turbine disintegration and heavy load drop. My assessment included a review of these PSA studies (Refs 18 and 22). The findings from the assessment of turbine disintegration PSA study can be found in Section 4.2.10.3. For the assessment of heavy load drop see Section 4.2.16 on the fuel route PSA.

251. During Step 3, Hitachi-GE also committed to perform detailed internal fire and internal flood PSAs; these were submitted in Step 4 and are considered in Sections 4.2.11 and 4.2.12 of this assessment report. The internal flooding PSA includes consideration of blast hazard and pipe whip.

4.2.10.2 STRENGTHS

252. The initial list of possible internal hazards considered for prioritisation has been improved since Step 3 and is now consistent with international good practice.

253. The approach and criteria for prioritising the analysis of the identified hazards is comprehensive and well documented. This includes consideration of shutdown POSs and the SFP.

254. Combinations of internal hazards are explicitly considered and dispositioned.

4.2.10.3 FINDINGS

255. The detailed technical reviews that support the findings discussed in the following paragraphs are documented in Refs 119 and 120.

256. Following the review of the documentation provided by Hitachi-GE in response to RO-ABWR-0040 it was apparent that significant improvements have been made since Step 3. The review of the Step 4 submissions has concluded that some shortfalls remain outstanding which should be addressed following GDA. These are discussed in the following paragraphs.

257. Hitachi-GE identified in the internal hazards PSA prioritisation topic report (Ref. 20) that further evaluation of hazards within the PCV was required within GDA; however no further evaluation has been submitted to ONR. A deterministic evaluation of hazards in

the PCV has been submitted to ONR internal hazards inspectors (Ref. 121), but the PSA documentation has not been updated to consider the conclusions of this deterministic analysis. To assess the impact on the risk profile for the PSA I performed a review of Ref. 121. My review did not identify any hazards which result in consequential damage within the PCV that are not already considered in the scope of the internal fire, internal flood or dropped loads PSAs. However, following discussions with the internal hazards inspector, I was made aware that a further revision of Ref. 121 is expected late in GDA to address queries from the ONR internal hazards inspector on areas such as consequential pipe whip. This update may result in additional hazards to be considered in the PSA. The PSA internal hazards prioritisation documentation should be updated to explicitly include consideration of hazards within the PCV and identify if any additional hazards are required to be considered in the PSA.

258. The approach and criteria to prioritise the identified hazards is comprehensive and well documented, however a significant number of hazards and locations are screened out on the basis of deterministic information concerning the substantiation of barriers preventing the propagation of the hazard beyond a single division. The substantiation of barriers to prevent hazards in one division affecting neighbouring divisions has been reviewed by the internal hazards inspectors, resulting in RO-ABWR-0082 (Ref. 122) being raised during GDA and a number of assessment findings have been identified in the internal hazards assessment report (Ref. 123).

259. The assumptions and criteria used for the screening of internal hazards for PSA, and the development of any PSAs, should be consistent with the information presented in the internal hazards analyses. This may not be the case for the UK ABWR as the internal hazards analysis has continued to be developed late in GDA. Upon request, Hitachi-GE submitted a comparison of the probabilistic and deterministic internal hazard analyses (Ref. 124). I reviewed this submission and, following discussions with the internal hazards inspector, identified the following shortfalls requiring further consideration:

- A number of class 1 barriers remain unsubstantiated at the time of writing this assessment report, with substantiation planned for submission and ONR review late in GDA.
- Further deterministic internal hazards analysis for dropped loads and substantiation of the relevant floors / operating deck are expected within GDA.
- As mentioned above, further analysis of hazards within the PCV is expected including consideration of consequential pipe break.

The comparison of the probabilistic and deterministic internal hazards analyses should be reviewed following GDA, to take any late developments in the deterministic internal hazards analyses into account along with any relevant PSA or deterministic hazard assessment findings. Where relevant, these shortfalls are also included in the internal hazard assessment report (Ref. 123).

260. The prioritisation of internal hazards identifies toxic or asphyxiant gasses which impact the main control room (MCR) as outside of scope of GDA PSA submissions requiring evaluation during site licencing. I challenged this approach in RQ-ABWR-1102 and RQ-ABWR-1243, as it is known in GDA that large quantities of nitrogen will be stored on-site. In response Hitachi-GE identified that ‘HVAC systems are designed such that hazardous materials located outside of the MCR compartment cannot prevent the delivery of the fundamental safety functions (FSFs) by either SSCs or operators’ (Ref. 125), however no substantiation is provided for this claim. The impact of a nitrogen release on the MCR was also assessed by ONR’s internal hazards inspectors, as discussed in Ref. 123, with an assessment finding being raised in the internal hazards assessment report. As previously discussed, the internal hazards

prioritisation for PSA should not be inconsistent with the information presented in the internal hazards safety case and should take any internal hazards assessment findings into account.

261. The impact of internal hazards on sources of radioactivity other than the reactor and the SFP on-site is partially considered in Ref. 30; however the assessment is at a high-level and needs to be expanded to consider all internal hazards and all sources of radioactivity following GDA, and take account of the response of the SSCs to the specific hazard.

262. The prioritisation of internal hazards for PSA submitted by Hitachi-GE (Ref. 20) relies upon the information available at this stage and arguments made using the GDA PSA model and results, which are expected to undergo further development beyond GDA. Therefore it will be necessary to re-examine the prioritisation following GDA, taking site specific characteristics and detailed design information into account, along with results and insights from the site specific PSA.

> ***Assessment Finding AF-UKABWR-PSA-004****: The licensee shall provide a revised systematic prioritisation of all internal hazards, including combined internal hazards, for all sources of radioactivity on-site that is representative of the site specific design and layout and consistent with the internal hazards deterministic safety case. The prioritisation shall include demonstration that the risk associated with all the screened out internal hazards would be insignificant compared to the ABWR total risk. The licensee shall then provide a revised PSA for internal hazards on the basis of the prioritisation performed.*

263. The internal hazards prioritisation identified the need for a PSA to be developed within GDA on turbine disintegration events. In Step 4 Hitachi-GE submitted an assessment of turbine missiles (Ref. 22), which was assessed by my review team. The initiating event frequencies used in the analysis submitted by Hitachi-GE are consistent with the Hitachi-GE response to RO-ABWR-0079, turbine disintegration safety case, raised by ONR's internal hazards inspector.

264. My review of Ref. 22 raised concerns summarised in RQ-ABWR-1102 and RQ-ABWR-1243 related to the lack of consideration of potential additional damage due to consequential fires and/or flooding resulting from turbine disintegration events. Furthermore, the analysis does not consider that multiple divisions of the heat exchanger building (Hx/B) could be impacted by a turbine missile. I requested Hitachi-GE to provide an estimate for the impact on the risk profile due to this omission. Hitachi-GE provided a response based upon the initiating event frequencies used in the PSA study at the time, which were later updated in the turbine disintegration safety case (Ref. 126). I conclude, based upon Hitachi-GE response to the RQs and additional calculations I performed using the updated initiating event frequencies presented in Refs 22 and 126, that the impact on risk, and therefore the impact on the risk profile, is very low at approximately 1% of the IEAP PSA CDF.

265. My evaluation of the findings in this particular area on the basis of Hitachi-GE's sensitivity analyses has shown that none of the shortfalls identified would lead to a significant increase in the risk results. However, I note that a number of shortfalls are related to the need to provide further justification and analysis to underpin the assumptions in the PSA once detailed design information and further deterministic analysis becomes available. If these assumptions are not supported the PSA would require update and there may be a risk impact. This will be pursued in the site specific stage to ensure that risk is managed ALARP as part of resolution of AF-UKABWR-PSA-001.

4.2.10.4 CONCLUSION

266. The coverage of the internal hazards in the PSA, excluding internal fire and internal flooding, which are discussed in Sections 4.2.11 and 4.2.12, is sufficient, at this stage of the project, to capture the risks due to internal hazards. Sufficient evidence has been presented to support closure of RO-ABWR-0040 (Ref. 127) and to support the UK ABWR ‘generic’ PCSR.

267. The hazards prioritisation performed is comprehensive and well documented, noting a number of shortfalls identified above, however the work has been developed for a generic design and further analysis is required in the site specific stage. The analysis undertaken by Hitachi-GE as part of this work relies upon substantiation of the internal hazards safety case being provided. Therefore, a revised systematic prioritisation and assessment of internal hazards should be performed after GDA, taking site specific characteristics, layout and detailed design into account, along with any developments in the internal hazards safety case. This requirement is captured by assessment finding AF-UKABWR-PSA-004.

4.2.11 Level 1 PSA: Analysis of Hazards – Analysis of Internal Fires (A1-2.7-2)

4.2.11.1 ASSESSMENT

268. My Step 3 assessment focused on the methodology proposed by Hitachi-GE for developing the internal fire PSA which was to be submitted in Step 4. I concluded that the general approach proposed by Hitachi-GE broadly met ONR expectations (Ref. 77). However, at Step 3 there was a lack of information on how key aspects of the PSA for internal fires during operation at power would be developed, such as evaluation of multiple spurious actuations due to hot shorts.

269. I captured all of these requirements in my Step 4 assessment plan (Ref. 2) and Hitachi-GE updated the methodology and PSA programme accordingly (Ref. 33) to deliver the Step 4 internal fire PSA.

270. At GDA Step 4 Hitachi-GE submitted a full set of documents presenting the PSA for internal fires during operation at power. Hitachi-GE also presented a PSA for internal fires during the low power and shutdown plant states. The assessment of the internal fires during low power and shutdown plant states PSA is presented in Section 4.2.15.

271. My assessment developed a high-level understanding of the contribution to overall plant risk from internal fires during operation at power. It also included in-depth sampling of selected elements of the PSA. The in-depth sampling provided the opportunity to assess the details of the risk model to confirm the logical structure, data used, the success criteria applied and the overall numerical evaluation of the risk. The sampling addressed a range of technical areas across the whole internal fire at power PSA to assess the adequacy of the modelling, as follows:

- Confirm whether the plant as a whole has been adequately partitioned into fire areas or compartments.
- Assess whether the scope the equipment included in the internal fire at power PSA is adequate to appropriately characterise the fire induced initiating events and the availability of mitigation equipment.
- Address the identification of cables to the identified equipment and their fire induced cable failure modes, including hot shorts.
- Assess the development of the IEAP PSA model into the fire risk model. Confirm that appropriate changes have been made to address the unique aspects of fire induced failure of the plant equipment.
- Review the frequency of fires occurring across the plant.

- Ensure that any qualitative or quantitative screening of fire scenarios meets the expectations described in ONR's PSA TAG (Ref. 4).
- Assess the detailed fire modelling to ensure it properly reflects the physical development of fires and is consistent with relevant good practice (RGP) as understood by ONR.
- Assess the modelling of the operator actions to ensure they account for the unique challenges associated with fire scenarios.
- Review the accident sequence modelling to confirm that it appropriately models the development of fault sequences between initiating events and consequences, and that the minimal cutsets and overall results of the model are physically and logically reasonable. This included assessing the use of the FRANX computer code.

272. My assessment was performed using the latest available models and documentation provided by Hitachi-GE which are described in Sections 2 and 3 of this report. My assessment was conducted with the assistance of TSCs and is presented in Ref. 128. This work was carried out to the standards required by the ONR's SAPs (Ref. 1) and PSA TAG (Ref. 4). I have also used other standards that represent RGP for nuclear reactor PSA. These include the ASME/ANS PRA Standard (Ref.72) and the EPRI TR-1011989 and NUREG/CR-6850 guidance (Ref. 129). National and international standards and guidance is embodied in ONR's PSA TAG (Ref. 4) and it is this guide that provides the principal means for assessing the PSA in practice.

4.2.11.2 STRENGTHS

273. The internal fire at power PSA presented by Hitachi-GE is based on the guidance in NUREG/CR-6850 (Ref. 129). This is an accepted standard for the production of internal fire PSAs and therefore represents a source of relevant good practice (RGP). My assessment did not identify any significant deviations from the NUREG methodologies. Where necessary, shortfalls have been identified for discussion in this report.

274. The structure of the human factors analysis is based on the NUREG-1921 methodology (Ref. 130). Hitachi-GE has refined this with more specific plant information subject to the level of detail available at GDA. ONR's HF inspector advised that this represented a suitable methodology for the fire PSA.

275. The internal fire at power PSA has been used by Hitachi-GE to risk inform the UK ABWR design. The fire risk optioneering conducted by Hitachi-GE is presented in its topic report on the use of PSA to support ALARP assessment (Ref. 35). Examples include: the identification of risk important cabling and considering the benefits of re-routing these cables, and applying fire rated boundaries in the back-up building.

276. The UK ABWR plant incorporates the following design features:

- Physical separation of equipment using divisional partitions within the plant where, for a given room, only a single train or division of electrical circuits is allowed to be present.
- The more comprehensive use of digital control systems with fibre-optic cables. The fibre-optic cables are not subject to spurious signal generation from fires.

These features have been modelled in the PSA and are shown to have a positive impact on reducing risk.

4.2.11.3 FINDINGS

277. My assessment gave rise to 12 regulatory queries (RQs) (Ref. 94), which raised many individual technical questions. The vast majority of these technical questions were resolved to my satisfaction by written responses and workshop discussions with Hitachi-GE.

278. My findings summarised below are those items I consider need to be reviewed for the post GDA site specific internal fire at power PSA. My findings arise due to:

- Shortfalls in the analysis with respect to the applicable standards, in particular meeting the ONR's PSA TAG (Ref. 4) guidance. These impacted the quality or completeness of the PSA.
- The use of generic assumptions in the GDA analysis that need confirming and incorporating into the detailed design beyond GDA.
- Aspects of the PSA that require additional information on the detailed design for a specific site.

4.2.11.3.1 Shortfalls which Impact the Quality or Completeness of the PSA

279. The following shortfalls need to be included in future updates to the internal events at power PSA. Hitachi-GE has acknowledged these require addressing in its responses to my assessments:

- Multiple spurious operations of the emergency diesel generator (EDG) cooling water valves leading to CCF of EDG cooling.
- Refined modelling for the time the main steam isolation valves (MSIVs) remain open following cable damage due to a fire. The time duration assumed before closure of these MSIVs currently appears to be conservative. A more realistic duration may permit operator initiation of emergency core cooling to be claimed because an adequate suppression pool (S/P) level would still be available.
- The modelling of main control room (MCR) abandonment can be made more realistic by taking greater credit for smoke removal using the HVAC.
- A fire initiated excessive feedwater initiating event can result in water entering the reactor core isolation cooling system (RCIC). This is currently assumed to fail the turbine which results in loss of the water injection function into the reactor pressure vessel (RPV). This assumption is conservative because the RCIC has design features to protect the turbine from becoming water filled. Further consideration of this fault sequence may show that these design features together with operator action to drain the system can ensure it remains operable with adequate time remaining to protect the core.
- The modelling used for self-healing of hot shorts for fire initiated safety relief valve (SRV) opening is conservative because shorter durations than assumed may apply (RQ-ABWR-1139).
- Fire damage to cables can simulate an emergency core cooling demand following the plant trip required after a fire (Ref. 131). A spurious emergency core cooling demand would put loads into the electrical power system that increases the likelihood of a loss of off-site power (LOOP). The internal events PSA considers an increased likelihood of a LOOP for emergency core cooling demands. The internal fire PSA needs to consider whether a higher LOOP probability applies.
- Assessment of the fire initiating event frequencies identified a discrepancy in the data used in the internal fire PSA. Hitachi-GE has adequately explained the discrepancy and states that the correct fire initiating event frequencies for the back-up building and heat exchanger building (Hx/B) are 23% and 25% higher, respectively (Ref. 132). Hitachi-GE has updated the internal fire PSA with the

revised data. Although the frequency of some fires increases, the large release frequency (LRF) reduces because of further refinements which removed selected conservatisms (Ref. 68).

280. Assessment finding AF-UKABWR-PSA-001 requires the future licensee to resolve these shortfalls. Five of the seven shortfalls listed in this paragraph have been raised to highlight conservatism in the PSA. When they are resolved by the future licensee the core damage frequency (CDF) and LRF from the UK ABWR plant will reduce. The other two shortfalls discussed in the first and sixth bullet points are potential optimisms in the PSA. I have reviewed the output from Hitachi-GE's latest PSA refinement study and I am satisfied that the shortfalls do not impact the dominant contributors to the CDF or LRF (Ref. 26). The fault frequencies associated with the shortfalls are therefore very small. This does not adversely impact my understanding of the overall UK ABWR plant risks at GDA.

4.2.11.3.2 The use of Generic Assumptions in the GDA Analysis

281. The multi-compartment fire analysis assumes that propagation of liquid pool fires between adjacent rooms does not occur because doorway curbs are provided. Hitachi-GE states that, although assumed in the PSA, doorway curbs are not currently in the GDA UK ABWR design. My discussions with the ONR internal hazards inspector indicated that Hitachi-GE's latest intent for limiting the extent of liquid pool fires is to provide local bunds at selected locations within the plant (Ref. 133). I consider that review of the internal fire at power PSA is needed to reflect the plant design as it develops.

282. The internal fire PSA includes examples where the protection of cables is assumed by embedding the conduit within concrete. When questioned how such assumptions are tracked and managed Hitachi-GE responded that the PSA assumptions have been endorsed by the electrical engineering team and are appropriately recorded in the cable analysis documentation (Refs 57 and 134).

283. I questioned a specific instance where an embedded cable is assumed protected in the room where it terminates at a valve motor. I questioned how this cable can be protected between its point of emerging from concrete and the valve motor, unless the valve itself is provided with fire protection (Ref. 134). Hitachi-GE stated that the valve did not need fire protection and this was an example where analysis during detailed design is appropriate (Ref. 134). This view is based on an implicit assumption that detailed fire analysis in the future will show that there are no credible fires in this room. Such implicit assumptions need to be identified and controlled. I consider that there may be a significant number of rooms for which this implicit assumption may apply.

284. The MCR fire analysis assumes that it is not possible that spurious pump or valve motor actuations can occur with fire induced bypass of the valve protective torque and limit switches (Ref. 135). I questioned whether the valve limit and torque switches are wired into the control circuit locally to the equipment. The response from Hitachi-GE assumes that the design for the UK ABWR will be the same as the design for the current J-ABWR, and use the local wiring option (Ref. 135). This would exclude the failure modes of concern due to fires in the MCR. I have reviewed the PSA assumptions register and cannot find this recorded (Ref. 32). This issue is therefore a notable example of the issues expressed in AF-UKABWR-PSA-001 (Part 3) as this assumption should be captured and pursued in the UK ABWR design during the site specific phase.

4.2.11.3.3 Site Specific Information Needed to Complete the Fire PSA

285. At GDA Step 4 there is a lack of plant specific operating procedures. My assessment has noted various aspects of the modelling where assumptions regarding the operating procedures have been made. The internal fire at power PSA and the associated

operating procedures will need iterative review and updating as the plant design develops beyond GDA. I also note that full development of the fire PSA requires plant walkdowns to confirm the final design and location of equipment, for example cable trays, conduits and the detection and suppression equipment. This aspect of the fire PSA assessment is deferred by necessity into post GDA detailed design and construction.

286. Multi-compartment fires in the internal fire PSA is currently responsible for more than half of the core damage frequency (CDF) and large release frequency (LRF) predictions. My assessment highlighted the conservative treatment of fire modelling for ‘type 2’ multi-compartment fires. Type 2 fires do not produce a damaging hot gas temperature of greater than 330°C within the compartment of origin, but may expose PSA equipment/cables on the opposite side of a non-rated barrier due to a damaging plume temperature or radiant heat flux.

287. Hitachi-GE is aware of this modelling approximation which is stated to be due to lack of design detail of fire barrier rating and raceway location at the GDA stage. Hitachi-GE has undertaken an internal hazards PSA refinement sensitivity study which indicates a significant reduction in the risk contribution from multi-compartment fires (Ref. 26). Further comment on this item is made in Section 4.2.11.3.5 which discusses the internal fire aspects of the internal hazards PSA refinement.

288. Hitachi-GE has excluded the circulating water building from the global plant analysis boundary (Ref. 136)[1]. I do not consider this to be significant for the purposes of GDA because the frequency for a loss of ultimate heat sink (LUHS) would be bounded by loss of condenser cooling faults. I acknowledge that site specific information is needed on the circulating water system / structures for it to be adequately modelled. However, loss of the circulating water equipment within this building due to fire would require a plant trip during power operations. Therefore, I consider that the circulating water building should be included in the internal fire at power PSA global plant analysis boundary during the site specific analysis beyond GDA.

289. The development of the internal fire at power PSA relies on a large volume of design and spatial information. This is not fully available during GDA because the design is still developing. Hence, Hitachi-GE has used assumptions and conservative modelling where design uncertainty exists. For example, conservative cable mapping information is used in its analysis of fire scenario consequences (RQ-ABWR-1139). The internal fire at power PSA has yet to catch up with all the design changes that have occurred during GDA; this will need to be rectified beyond GDA. In addition, the PSA will need to be developed beyond GDA to reflect the detailed design as it develops. The need to develop the PSA is acknowledged by Hitachi-GE as follows:

- The PSA assumptions list records the assumptions made during GDA which are incorporated into the PSA and have not yet been closed (Ref. 32).
- The model change tracking/risk impact evaluation database contains references to design changes from within GDA up to July 2017 which need incorporating into the PSA (Ref. 137).
- The PSA commitment log records the need to reflect site specific cable routing within the fire PSA (Ref. 82).

The future licensee will need to ensure that it records all these aspects of PSA development identified during GDA, and provides a plan for their resolution in its site specific analysis beyond GDA. The PSA assumptions list (Ref. 32), model change tracking/risk impact evaluation database (Ref. 137) and PSA commitment log (Ref. 82) have been developed for this purpose.

[1] *A further revision of this report was submitted late in GDA Step 4, which was not included in the PSA assessment.*

290. I consider this approach to be reasonable given that Hitachi-GE has produced an internal hazards PSA refinement sensitivity study which supplements the internal fire PSA model results (Ref. 26). This refinement updates a selection of the dominant contributors to risk for internal fires at power by removing various conservatisms and including the latest design information for cable routing. This provides an adequate understanding of the fire risks at the completion of GDA. However, further development and refinement of the internal fire PSA should to be undertaken during the site specific phase.

4.2.11.3.4 Seismic Fire Interactions

291. This task is a qualitative assessment of a seismic event potentially causing the following events:

- seismically induced fires;
- degradation of fire suppression systems;
- spurious actuation of detection and suppression systems; and
- degradation of manual fire-fighting capability.

292. Hitachi-GE has undertaken this qualitative assessment and generated recommendations for enhancing the design and operational procedures where seismically induced fires have the potential to threaten safety measures credited for the mitigation of a seismic event. This includes improving component withstands against seismic failures, and procedural enhancements for improving the response of operators and firefighters to the resulting fires.

293. At GDA much of the detailed design for equipment which may give rise to a seismically induced fire, and for the fire suppression systems is not complete. Therefore the treatment of seismic fire interactions cannot be fully addressed during GDA. Plant walkdowns to identify vulnerabilities and the development of detailed plant procedures is needed. Hence completion of this work must be deferred to the site specific analysis.

4.2.11.3.5 Refinement of the Internal Fire At power PSA

294. My assessment of the UK ABWR PSA, reported above, noted that the large release frequency (LRF) for the plant as a whole was a substantial proportion of the ONR SAP (Ref. 1) BSL for Target 9 (frequency of 100 or more fatalities) (Ref. 11). Approximately one third of this contribution was from the internal fire at power PSA (Ref. 11). This raised regulatory concern regarding the extent of defence-in-depth provided by the design (SAP EKP.3 (Ref. 1)).

295. My assessment also recognised that the overall level of risks was excessively influenced by conservative analysis, and this was reflected in the internal fire at power PSA. The primary reason for this was the lack of detailed design information during the GDA process which required conservative modelling. I considered that this approach was hindering my understanding of risk insights and limited the use of the internal fire at power PSA to inform the design to ensure that risk was being reduced towards ALARP during GDA.

296. To address this I raised a series of RQs for the internal fire at power PSA to: investigate the conservatisms within the risk model, clarify the interface of the PSA with the detailed design, pursue refinement of the internal fire at power risk model and explore the manner in which the PSA was being used to inform the ALARP process (RQ-ABWR-1178, RQ-ABWR-1185, RQ-ABWR-1267, RQ-ABWR-1399 and RQ-ABWR-1471). A number of these RQs were specific to the internal fire at power PSA. However, a number of these RQs were exploring the use of the PSA more broadly to provide risk insights and inform the ALARP process. My investigations of the

latter led to issuing RO-ABWR-0076 (Ref. 138) which addressed my expectations for the use of the UK ABWR PSA as a whole to inform the ALARP process.

297. Hitachi-GE's response to this series of RQs for the internal fire at power PSA was to undertake an internal hazards PSA refinement, which removed some conservatisms from the internal fire at power PSA (Ref. 26). This was done by using newly available design information for the plant and improving the technical approaches used.

298. The output of the internal hazards PSA refinement reduces the CDF and LRF of the fire PSA results by a factor of 3.8 and 6.2, respectively. The CDF from internal fires CDF has been reduced from one third of the total to 12% of the total CDF. The refinement study also shows that the dominance of fire events in the back-up building have been significantly reduced, and that the LRF (used as a surrogate for societal risk) was artificially high with respect to ONR SAP Target 9 (Ref. 1). Additionally PSA insights have been developed to inform the ALARP process. The use of the PSA to inform the ALARP process is discussed further together with the closure of RO-ABWR-0076 in Section 4.2.20 of this assessment report.

299. I have assessed the internal fire aspects of the internal hazards PSA refinement and consider that Hitachi-GE has presented a useful and reasonable analysis (Ref. 139). The internal hazards PSA refinement is part of the overall GDA submission from Hitachi-GE and will be taken forward by a future licensee during the site specific phase.

300. However, my assessment of the internal fire aspects of the internal hazards PSA refinement has also identified new assumptions about the design which are not always clearly identified. I consider it appropriate for Hitachi-GE to review the internal fire at power refinement risk model to ensure that all the important new assumptions are clearly stated in the document and included in the PSA assumptions list (Ref. 32). This is to ensure that they are not overlooked and are taken into account within the detailed design.

301. The most notable new assumption applies to the multi-compartment analysis. It states that all the rooms which contain PSA target equipment that were previously exposed to fires across a barrier are excluded from the risk model. This is based on the assumption that a robust fire barrier is now providing protection (Ref. 26). This significantly reduces the importance of fires in the back-up building by preventing spurious safety relief valve lifts from hot shorts on back-up building cables. Discussions with the ONR internal hazards inspector clarified Hitachi-GE's assumption that these barriers are three hour rated concrete walls. Barriers of this nature should have a substantial withstand to the effects of fire. However, advice from the ONR internal hazards inspector was that justification for these barriers had not been presented at the time of writing this report. Resolution of this item is being addressed within the internal hazards assessment report (Ref. 123).

4.2.11.4 CONCLUSION

302. The PSA for internal fires during operation at power presented by Hitachi-GE is based on accepted standards and guidance for the production of internal fire PSAs. I consider that the methods and guidance used are consistent with international good practice. My assessment did not identify any significant deviations from these standards and guidance. However, my detailed assessment identified a number of shortfalls with respect to the expectations in ONR PSA guidance. These are the use of generic assumptions that need confirming and incorporating into the detailed design beyond GDA, and aspects of the PSA that require additional detailed design information in later stages of the NPP development.

303. Internal fires during operation at power were initially contributing one third of the overall plant LRF. My assessment recognised that this result was excessively influenced by conservative analysis. This hindered my understanding of risk insights and initially limited the use of the internal fire at power PSA to inform the design to ensure that risk was being reduced towards ALARP during GDA.

304. Hitachi-GE addressed this by presenting an internal hazards PSA refinement. The internal fire PSA aspect of this study removed selected conservatisms using newly available design information. The LRF was reduced by a factor of 6.2, insights were developed to risk inform the design and Hitachi-GE was able to demonstrate that the fire risk was being reduced towards ALARP during GDA.

305. I consider that the internal fires at power PSA together with the internal hazards PSA refinement study provides an adequate understanding at GDA of the UK ABWR risk due to internal fires during operation at power. However, the PSA for internal fires during operation at power needs further development during the detailed design phase to address the assessment findings in Annex 5 of this report and to reflect the plant as the design develops.

306. My conclusion on the PSA for internal fires at low power and shutdown plant states is presented in Section 4.2.15 of this report.

4.2.12 Level 1 PSA: Analysis of Hazards – Analysis of Internal Flooding (A1-2.7-3)

4.2.12.1 ASSESSMENT

307. My Step 3 assessment focused on the methodology proposed by Hitachi-GE for developing the internal flooding PSA which was to be submitted in Step 4 (Ref. 77). I concluded that the general approach proposed by Hitachi-GE broadly met ONR expectations. However, additional information was needed on how particular technical tasks were to be approached.

308. I captured all of these requirements in my Step 4 assessment plan (Ref. 2) and Hitachi-GE updated its methodology and PSA programme accordingly (Ref. 33) to deliver the Step 4 internal flooding PSA.

309. At GDA Step 4 Hitachi-GE submitted a full set of documents presenting the PSA for internal flooding during operation at power. The submission from Hitachi-GE is described in Sections 2 and 3 of this report. My assessment of the internal flooding PSA for low power and shutdown plant states is presented in Section 4.2.15 of this report.

310. My assessment addressed the overall methodology used by Hitachi-GE for the PSA as a whole, and also included in-depth sampling of selected elements of the PSA. This in-depth sampling was designed to challenge various detailed aspects of the modelling to confirm the technical basis for the model inputs, the success criteria and the accident sequence modelling. The sampling addressed a range of technical areas across the whole internal flooding at power PSA, as follows:

- the adequacy of the flooding area definitions;
- flooding source identification, flooding inventories, the propagation of floods and the identified targets;
- the basis for qualitative screening of plant areas;
- assessment of whether the flooding scenarios are physically and logically reasonable;

- a check that the flooding initiating event frequencies are auditable and reproducible;
- review of flooding consequences to confirm that the propagation paths, flowrates, and timings are reasonable;
- a check that the human reliability mitigation actions and human error probabilities are reproducible;
- a check that the logical modelling between initiating events and consequences is done properly; and
- the minimal cutsets (MCS) and overall results of the model to confirm that they are physically and logically reasonable.

311. My assessment was performed using the latest available models and documentation provided by Hitachi-GE (Refs 27, 34, 69 and 70). This also included initial assessment of earlier versions of these documents. My assessment gave rise to nine regulatory queries (RQs) and corresponding responses. These are discussed as required to explain my assessment below.

312. My assessment is presented in Ref. 140 and was carried out to the standards required by the ONR SAPs (Ref. 1) and PSA TAG (Ref. 4). I have also used other standards that represent relevant good practice (RGP) for nuclear reactor PSA. These are the ASME/ANS PRA Standard (Ref.72) and the EPRI 1019194 guidance (Ref. 141). National and international standards and guidance is embodied in ONR's PSA TAG (Ref. 4) and it is this guide that provides the principal means for assessing the PSA in practice.

4.2.12.2 STRENGTHS

313. The internal flooding at power PSA presented by Hitachi-GE is based on the EPRI internal flooding guidance (Ref. 141). This is an accepted standard for the production of internal flooding PSAs and therefore represents a source of RGP. My assessment did not identify any significant deviations from the EPRI standard methodologies. The internal flooding at power PSA often uses the latest industry guidance and data.

314. Hitachi-GE makes a series of bounding assumptions within the internal flooding at power PSA. These involve assumptions regarding the plant configuration, the scope of failures assumed, and the feasibility of operator cues and the procedures available. I consider that the majority of these assumptions are reasonable and provide the boundaries necessary to complete the analysis for a plant in GDA for which the design is developing. Hitachi-GE has been thorough in its recording of all the assumptions made in the PSA (Ref. 32). However, I have identified some shortfalls related to these assumptions which are discussed below under 'findings'.

315. The internal flooding at power PSA has been used by Hitachi-GE to risk inform the design. The flooding risk optioneering conducted by Hitachi-GE is presented in Ref. 35. Examples include: alternative flooding pathways, component heights in the nuclear steam supply system instrumentation rack rooms, and the protection of non-return valves in the ECCS from high energy line breaks (HELBs) in the main steam tunnel.

316. The UK ABWR plant incorporates the following design features:

- Dry fire piping headers are used in all buildings except secondary containment to supply hose reels and other fire suppression systems. This reduces the flooding risk when compared with wet fire piping headers.
- No unlimited water sources are routed into the reactor building. The UK ABWR used closed loop systems for component cooling. Feedwater and fire water volumes, while large, are not unlimited.

These features have been modelled in the PSA and are shown to have a positive impact on reducing risk.

4.2.12.3 FINDINGS

4.2.12.3.1 Capping of Drains in the Reactor, Control and Heat Exchanger Buildings

317. The management of water from internal flooding events is performed by designing flood water propagation pathways into the building structures. The structures used to direct flood water includes grated hatches in the floors and ceilings of compartments, doors designed to open or remain closed in the presence of flood water, and stairwells and lift shafts to direct flood water to safe locations. A floor drainage system using pipework to direct flood water is also provided. The internal flooding at power PSA currently assumes that all floor drains in the reactor building, heat exchanger building and control building are ‘capped’. Hence these are not currently modelled as flood water propagation pathways in the PSA.

318. This assumption is made because the floor drains in the current design are not ‘divisionalised’. Non-divisionalised floor drains means that flood water would propagate through uncapped drains across multiple divisions with the potential to fail redundant protection systems. The capped floor drains assumption means that flood water has additional potential to build-up locally instead of being drained away. However, other propagation pathways are still available.

319. I raised RQ-AWBR-1140 (Ref. 94) in order to understand the impact of the assumption that floor drains will be capped. Hitachi-GE provided a sensitivity study in which the floor drains are effective and all flooding due to spray scenarios did not propagate beyond the source room. The core damage frequency (CDF) and large release frequency (LRF) from the internal flooding PSA during operations at power reduced by 19% and 14% respectively (Ref. 27; Section 10.2). I consider that the assumption that all drains are capped is an area of conservatism in the PSA.

320. Hitachi-GE has stated that this item is recognised by its design teams and that as the design matures it is expected that the floor drains will be uncapped. Hitachi-GE has developed a plan for addressing drainage system design (Ref. 142) to ensure that risk is managed ALARP.

321. Hitachi-GE does not intend to improve the modelling within GDA. Therefore this will need to be performed following the development of the drainage design beyond GDA. I consider that the risk will reduce in the future as this design is developed.

4.2.12.3.2 Flooding Scenario Development

322. My assessment highlighted a number of simplifications used by Hitachi-GE for flooding scenarios which I judge lead to conservative estimates of risk. For example, a HELB from the RHR in room 214. The flood water originates from the suppression pool (S/P) and propagates from reactor building level B2F to reactor building level B3F, draining the S/P and is assumed to fail all the susceptible SSCs in the local area and the propagated areas, such as the emergency core cooling (ECCS) pump rooms on the ground floor. This flooding scenario is presented as the third most risk important (Ref. 27: Table 8-4).

323. However, it is unclear, for this scenario, if the S/P would actually be drained. This is because room 214 at elevation B2F is only marginally lower that the normal suppression pool water level (approximately 50cm lower). I raised this with Hitachi-GE in RQ-ABWR-1140 (Ref. 94).

324. The response from Hitachi-GE to this item notes that the pipework location is not currently known and it has been conservatively assumed that the S/P will drain under gravity (Ref. 143: Section 3.7.3). I consider that a modelling improvement is needed to ensure that the flooding consequences represent the actual elevation within the suppression pool for room 214. The presence of flood doors between the flood source and the ECCS pumps should also be considered. This would show that the volume of flood water is a small fraction of the total S/P. The ECCS pumps would therefore not be flooded on the ground floor level. These systems would be used for decay heat removal in response to the event and significantly reduces the risk contribution from this scenario. Hitachi-GE has included the improvement of flooding scenario development into its PSA commitment log (Ref. 82).

4.2.12.3.3 Flooding Source Grouping

325. My assessment of the internal flooding PSA model identified multiple areas in which there appeared to be ‘over-grouping’ of initiating faults. For example:

- Flooding scenario ‘RB-B2F-I-251-FSWP-P-2(5)’. This initiating event groups together pipework breaks from the following flooding sources: P11 (Makeup Water Purified: water volume 2814m^3), P13 (Makeup Water Condensate), P21 (RBCCW: each division water volume is 678m^3), and P25 (HVAC Emergency Cooling: 339m^3). These are widely varying volumes and I consider that the flooding impacts from each separately could vary significantly. The risk model documentation did not provide any justification for the grouping used or how the representative pipework failure is chosen.
- Flooding scenario ‘RB-1F-I-406A-HEWP-P-18’ which is failure of a feedwater line from pipe whip following a steam line break in the main steam tunnel (room RB-406). This scenario assigns all feedwater line ruptures to feedwater system line B (FDW-B). This may be conservative because the rupture of FDW-B compromises the operation of the emergency core cooling systems. I consider that including the likelihood of FDW-B being ruptured, rather than assuming it is always ruptured, would provide a more realistic risk model.

326. I requested that Hitachi-GE explain the reasoning behind the modelling approach in RQ-ABWR-1140 (Ref. 94). Hitachi-GE’s response confirmed that the modelling is based on a bounding approach (Ref. 143). Hitachi-GE stated that initial modelling used very conservative estimates of flood propagation but subsequent modelling was improved. Hitachi-GE has also included less conservative grouping of flooding scenarios in its internal hazards PSA refinement work (Ref. 28). This focused on the most risk significant scenarios for which new less conservative scenarios were developed.

327. I consider that the approach to flooding source grouping used by Hitachi-GE has been improved, but still contributes to a conservative estimate of the internal flooding at power risks. This should be addressed beyond GDA. To ensure this is done assessment finding AF-UKABWR-PSA-001 requires the future licensee to resolve this shortfall.

4.2.12.3.4 Credit for Safety Equipment in Adverse Environmental Conditions

328. My assessment highlighted a flooding scenario in which credit is taken for safety systems operating in adverse environmental conditions but without apparent justification.

329. The example flooding scenario is rupture of a LPFL suction line from the bottom of the S/P. The rupture remains unisolated and drains the S/P. The flood water immediately fails the associated divisional ECCS. However, the ECCS in the other two divisions

would not be directly affected by this flooding event, and therefore available to cool the core.

330. Hitachi-GE explained that the ECCS water would be taken from the condensate storage tank (CST) in the absence of any water in the S/P. However, the volume of water in the CST was insufficient for the 24 hour mission time (Ref. 143). This is addressed by providing automatic additional water supplies from the makeup water purified system (MUWP) to the CST. The operation of this system is assumed by Hitachi-GE to be unaffected by the flooding scenarios. The equipment is located two elevations above the source of the flood with multiple walls between the source and equipment.

331. My concern is that steam will eventually be discharged from the ruptured line into the reactor building. This is because there is no water in the S/P to condense steam from the safety relief valves (SRVs). This could create an adverse operating environment for the safety systems throughout the building that are claimed in the risk model. Hitachi-GE needs to demonstrate that the equipment claimed in the PSA is protected from the adverse steam environment over the mission time. To ensure this is done assessment finding AF-UKABWR-PSA-001 requires the future licensee to resolve this shortfall.

4.2.12.3.5 Human Failure Events

332. My assessment of the internal flooding at power risk model identified only seven internal flooding-related human failure events (HFEs). As the internal flooding contribution to CDF and LRF is currently significant, I investigated this further. This revealed a number of dominant flooding scenarios in which the absence of operator recovery appears conservative. The following flooding scenarios are examples.

333. Flooding scenario ‘RB-B2F-III-214-HEWP-P-8’ is a LPFL suction line failure from the bottom of the S/P which is downstream of the system pump. The S/P is drained by continued operation of the ECCS taking suction from the S/P and discharging the water via the pipe rupture. At present the internal flooding PSA model does not take credit for the operator to trip the pump and close the isolation valve. I consider that an operator intervention should be investigated to determine whether diagnosis and operator action is realistic to trip the pump and close the isolation valve in the time available.

334. There are two flooding scenarios in the top ten MCSs for CDF and LRF which include failure to trip the reactor. One example is flooding scenario ‘RB-4F-III-714A-FSWP-P-11’ (flooding from the reactor building HVAC which contains electrical equipment for the RIPs). This is the most dominant cutset for a large release in the internal flooding PSA (Table 8-4: Ref. 27). The flooding scenario involves water propagation down floor penetrations, stairwells and elevator shafts to reactor building B1F elevation. The modelling used in the PSA conservatively assumes that enough water accumulates at the B1F elevation to fail division A, B and C electrical panels. This fails many systems including the reactor protection system (RPS) and an ATWS is assumed which results in a large release.

335. Based on the information provided to date by Hitachi-GE I consider that the above two flooding scenarios would develop relatively slowly and this would potentially allow adequate time for a manual shutdown of the reactor which could mitigate the large release consequences.

336. In response to my questions in RQ-ABWR-1140 (Ref. 143), Hitachi-GE acknowledged that there is justification for considering additional internal flooding mitigating operator claims. Hitachi-GE has now included credit for manual reactor shutdown in its internal

hazards PSA refinement. This refinement work has been presented by Hitachi-GE (Ref. 28); my assessment of this is discussed below.

4.2.12.3.6 Refinement of the Internal Flooding At Power PSA

337. My assessment of the UK ABWR PSA, reported above, noted that the LRF for the plant as a whole (used as a surrogate for societal risk) was a substantial proportion of the ONR SAP (Ref. 1) Basic Safety Level (BSL) for Target 9 (frequency of 100 or more fatalities at $1x10^{-5}$/year) (Ref. 11). Approximately one third of this contribution was from the internal flooding at power PSA (Ref. 11). This raised a regulatory concern regarding the extent of defence-in-depth provided by the design (SAP EKP.3 (Ref. 1)).

338. My assessment also recognised that the overall level of risks was excessively influenced by conservative analysis, and this was reflected in the internal flooding at power PSA. The primary reason for this was the lack of detailed design information at the GDA stage, which required conservative modelling within the internal flooding at power PSA. I considered that this approach was hindering my understanding of the risk insights and limited the use of the internal flooding at power PSA to inform the design and to ensure that risk was being reduced towards ALARP during GDA.

339. To address this for the internal flooding at power PSA I raised a series of RQs to investigate the conservatisms within the risk model, clarify the interface of the PSA with the detailed design, pursue refinement of the internal flooding at power risk model and explore the manner in which the PSA was being used to inform the ALARP process (RQ-ABWR-1140, RQ-ABWR-1180, RQ-ABWR-1185, RQ-ABWR-1267 and RQ-ABWR-1399). A number of these RQs were specific to the internal flooding at power PSA. However, a number of these RQs were exploring the use of the PSA more broadly to provide risk insights and inform the ALARP process (Ref. 138).

340. Hitachi-GE's responses to this series of RQs for the internal flooding at power PSA was to undertake an internal hazards PSA refinement study which removed selected conservatisms from the internal flooding at power PSA (Ref. 28). This was performed by using newly available design information and removing conservatisms.

341. Hitachi-GE focused on the LRF in its refinement. Therefore the CDF was reduced by a small amount, but the LRF was reduced by a factor of 4.5. The refined analysis shows that HELBs contribute less to risk than previously understood, in particular within the main steam tunnel. Here breaks of the steam or feed pipework have been shown by Hitachi-GE to be less likely than previously understood. Hence the ECCS is more likely to be available in the internal hazards PSA refinement; rather than being assumed failed in the internal flooding at power PSA (Ref. 144: Section 5.3.3).

342. The internal hazards PSA refinement also indicates a reduced risk from fault sequences in which the reactor failed to trip due to flooding damage of the RPS. This is because credit has now been taken for operator action to trip the reactor, when time permits. Refinement of the grouping of flood sources has also been undertaken by Hitachi-GE.

343. I have assessed the internal flooding aspects of the internal hazards PSA refinement and consider that Hitachi-GE has presented a useful and reasonable analysis (Ref. 139). The internal hazards PSA refinement will need to be included within the base case PSA by a future licensee during the site specific phase.

344. The Hitachi-GE internal hazards PSA refinement shows that the previous internal flooding at power PSA presents an artificially high LRF (used as a surrogate measure of societal risk) with respect to ONR SAP Target 9 (Ref. 1). Additional risk insights have been developed to inform the ALARP process. This is discussed further together with the closure of RO-ABWR-0076 in Section 4.2.20 of this report.

345. My assessment of the refined PSA identified two main observations, the impact the assumption of capped drains has on the operator response to flooding events and the documentation provided.

346. Hitachi-GE continues to assume, in its internal hazards PSA refinement, that all the floor drains are capped. A new assumption was included in the internal hazards PSA refinement which states that no credit for isolating flooding sources is taken within two hours of the initiating event. Discussion with Hitachi-GE indicates that the floor drains design is likely to be uncapped which is consistent with operating Japanese BWRs. I acknowledge that additional design detail for the UK ABWR is needed to resolve this item. However, clarity on the status of the floor drains as uncapped would provide greater confidence on the availability of alarms and indications to the operators following a flood. This would have permitted greater credit to be taken for operator isolation of flooding sources, and the assumption of no isolation for two hours could be reviewed and replaced with more realistic analysis. I note that the assumption that all floor drains are capped is under ALARP review by Hitachi-GE (Ref. 145: Table 4.6.5-2).

347. My assessment notes that the description of the flooding scenarios is lacking adequate detail. This continues into the latest documentation of the internal flooding at power PSA, although there is an incremental improvement. My discussions with Hitachi-GE revealed that details of the flooding scenario development are contained within a database that has not been presented for GDA. I consider that a better description of this supporting database is needed and should be included in the report describing the flooding scenarios (Ref. 70). This should also cover the uses and limitations of the database with specific emphasis on how it is used to develop the flooding scenarios from the flooding sources and the location of the safety equipment needed to protect the reactor plant. There are also a number of minor documentation issues that should be corrected (Ref. 139: Section 2.1).

4.2.12.4 CONCLUSION

348. The internal flooding PSA for operation at power presented by Hitachi-GE is based on accepted standards and guidance for the production of internal flooding PSAs. I consider that the methods and guidance used are consistent with international RGP. My assessment did not identify any significant deviations from these standards and guidance. However, detailed assessment identified that Hitachi-GE had applied conservative approaches to simplify the analysis, or because the design detail needed to produce a 'best estimate' risk model was not available during GDA.

349. This initially resulted in the internal flooding at power PSA contributing one third of the overall plant LRF. This hindered my understanding of risk insights and initially limited the use of the internal flooding at power PSA to inform the design to ensure that risk was being reduced towards ALARP during GDA.

350. Hitachi-GE addressed this by presenting an internal hazards PSA refinement study, which removed selected conservatisms using newly available design information and modelled large and small flooding sources separately instead of together. The LRF for the internal flooding at power PSA was reduced by a factor of 4.5. This enabled additional insights to be developed to support the design and demonstrate that flooding risk was being reduced towards ALARP during GDA.

351. I consider that the internal flooding at power PSA together with the internal hazards PSA refinement provide an adequate understanding, at GDA, of the UK ABWR risk associated with internal flooding at power. However, the internal flooding at power PSA needs further development during the site specific phase to address the assessment findings in Annex 5 of this report and to reflect the plant as the design develops.

352. My conclusion on the PSA for internal flooding for the low power and shutdown plant states is presented in Section 4.2.15 of this report.

4.2.13 Level 1 PSA: Analysis of Hazards – Prioritisation of External Hazards (A1-2.7)

4.2.13.1 ASSESSMENT

353. Prioritisation of external hazards is undertaken to enable ranking all possible applicable external hazards according to their potential risk significance in order to identify those for which detailed PSA modelling and evaluation is warranted. An initial review of the UK ABWR PSA prioritisation of external hazards against the expectations in ONR's PSA TAG (Ref. 4) was conducted in Step 3 (Ref. 77). This review identified the following shortfalls:

- The initial list of hazards identified for the UK ABWR PSA was not comprehensive when compared to international literature and did not include combinations of hazards.
- The approach and criteria for prioritisation of hazards presented was inadequate.
- Treatment of beyond design basis hazards was not clear.
- The scope of the hazards analysis did not address plant states other than at power conditions with the containment inerted or other on-site radionuclide sources apart from the reactor.
- There was a lack of clarity regarding references to key hazard information including the site envelope, justification of the criteria used to establish the design attributes and the applicable hazard curves.
- The hazard impact analyses to support the screening were not provided.

354. In view of these Step 3 findings, RO-ABWR-0041 was raised. The objective of RO-ABWR-0041 was to state ONR's expectations related to the identification of external hazards for the UK ABWR PSA and request Hitachi-GE to respond to the shortfalls identified by ONR's review.

355. I assessed Hitachi-GE's response to RO-ABWR-0041 during Step 4 with the support of specialist TSCs for review of the external hazard characterisation and the ONR external hazards inspector. The assessment focused on the following, in line with the expectations set out in RO-ABWR-0041 and ONR's PSA TAG (Ref. 4):

- completeness of the identification of external hazards for PSA;
- the approach and criteria for prioritising the identified hazards;
- consideration of combinations of hazards;
- consideration of cliff edge effects;
- the scope of the hazard analyses;
- characterisation of the hazard frequencies and magnitudes;
- hazard impact analysis; and
- any specific PSA studies performed (not including seismic PSA (SPSA).

356. Hitachi-GE's primary response to RO-ABWR-0041 was the 'Topic Report on External Hazards Prioritisation' (Ref. 21). This topic report was produced by Hitachi-GE over a number of revisions; my review mainly considered revision 2 (Ref. 146). Shortfalls identified by my review team were communicated to Hitachi-GE in a number of RQs (RQ-ABWR-1103, 1270 and 1379). The final submission for GDA was

revision 3 (Ref. 21), which I subsequently reviewed to determine if shortfalls identified in earlier revisions had been addressed.

357. The topic report on external hazard prioritisation identified the requirement for PSA studies to be performed on a number of hazards:

- high wind (wind-blown debris, tornado) – at power, shutdown and SFP;
- seismic activity – at power, shutdown and SFP; and
- aircraft impact – at power and SFP.

358. In addition, upon ONR request (RQ-ABWR-0669) studies were produced by Hitachi-GE concerning biological fouling and external flooding. My review also considered these analyses. In particular, I sampled the following studies for detailed review, on the basis of high risk significance:

- high wind – at power;
- aircraft impact – at power;
- biological fouling – at power; and
- external flooding – at power.

359. Detailed SPSA was performed by Hitachi-GE and is considered in Section 4.2.14 of this assessment report.

360. My review also considered Hitachi-GE's analysis of combinations of external and internal hazards on 'Combination of External Hazard and Internal Hazard in PSA' (Ref. 147).

361. It should be noted that hazards from malicious activities were not included in the scope of this assessment and are considered in the civil engineering assessment (Ref. 148).

4.2.13.2 STRENGTHS

362. The initial list of possible external hazards considered for prioritisation is consistent with international good practice.

363. The approach and criteria for prioritising the analysis of the identified hazards is comprehensive and well documented. This includes consideration of shutdown POSs and the SFP.

364. Combinations of external hazards are explicitly considered and dispositioned.

365. The external hazard PSA studies submitted following the prioritisation allow the risk profile from external hazards to be understood and for identification of further site specific studies and risk informed design activities.

4.2.13.3 FINDINGS

366. The detailed technical reviews that support the findings discussed in the following paragraphs are documented in Refs 119, 149 and 150.

367. Following the review of the documentation provided by Hitachi-GE in response to RO-ABWR-0041 it was apparent that significant improvements have been made since Step 3. The review of the Step 4 submission concluded that the shortfalls identified in the following paragraphs remain outstanding and should be addressed following GDA to ensure that the PSA provides a full and complete representation of the risk profile of the UK ABWR.

368. My review of the prioritisation of hazards identified that substantiation has not been provided that HVAC vents, steel doors and the buildings prevent energetic tornado missiles from entering buildings containing SSCs. Substantiation is expected in the site specific phase.

369. The impact of external hazards on sources of radioactivity other than the reactor and the SFP on-site is partially considered in the 'Topic Report on Assessment of Non Reactor Faults' (Ref. 30), however the assessment is incomplete and needs to be expanded to all external hazards and all sources of reactivity beyond GDA to ensure that the PSA provides a full and complete representation of the risk profile of the UK ABWR.

370. A significant number of external hazards have been excluded from Hitachi-GE analysis due to lack of site specific information to be able to evaluate the impact of the hazard. Examples of hazards that have not been assessed by Hitachi-GE due to reliance on unavailable site specific information are external fire, external explosion and external transport impacts.

> ***Assessment Finding AF-UKABWR-PSA-005****: The licensee shall provide a revised systematic prioritisation of external hazards. The prioritisation shall consider all sources of radioactivity on-site and the specific characteristics of the site. The analysis should address external hazards that could be correlated. The licensee shall provide a demonstration that the risk associated with all the external hazards screened out would be insignificant compared to the total risk. The licensee shall then provide a revised PSA for external hazards on the basis of the prioritisation performed.*

371. Hitachi-GE initially proposed that biological fouling and external flooding hazards were to be considered out of scope for GDA. I challenged this in RQ-ABWR-0669, and in response Hitachi-GE performed a PSA sensitivity study to examine the impact on the risk of a loss of the ultimate heat sink (LUHS) caused by biological fouling and the impact on risk from external flooding (Ref. 23).

372. My review of this PSA sensitivity study (Ref. 23) identified that the CDF calculation for a biological fouling event was calculated based upon conditional core damage probabilities following a manual shutdown rather than following a reactor trip. To explore the sensitivity of this assumption, Hitachi-GE performed an additional sensitivity study to consider a reactor trip in response to a biological fouling event which showed a large increase in CDF over the manual shutdown case. The sensitivity study shows that a biological fouling event could represent a significant proportion of the IEAP CDF, and therefore highlights the importance of considering biological fouling and other loss of heat sink initiating events within the PSA, including substantiation for the accident sequence modelled and any operator actions claimed.

373. LUHS due to external hazard has the potential to be a significant contributor to the UK ABWR overall risk profile and requires further analysis in the site specific phase. It should be noted that the fault schedule (Ref. 95) considers a reserve ultimate heat sink (RUHS) to provide protection against LUHS events. Design of the RUHS is considered by Hitachi-GE out of the scope of GDA and availability of a RUHS is not considered in the PSA sensitivity study. The PSA sensitivity study shows the importance of LUHS events and therefore the design of the RUHS should be risk informed using PSA insights during the site specific phase. External flooding events should also be explicitly considered in the PSA, taking site specific information into account.

> ***Assessment Finding AF-UKABWR-PSA-006****: The licensee shall consider loss of ultimate heat sink initiating events (including biological fouling) and external flooding initiating events within the site specific PSA, or adequately*

justify their exclusion. The analysis shall take site specific heat sink design and expected operator actions into account. The licensee shall use the analysis to identify any relevant PSA insights to aid improvement of the design or operation of the UK ABWR.

374. There are residual concerns over the tornado frequencies applied in the PSA study, as consideration of historical occurrence of large tornados in the UK could result in a higher initiating event frequency for high intensity tornados than is used by Hitachi-GE, due to the emphasis on recent, rather than historical, data. In addition, the frequencies used by Hitachi-GE are based upon the annual probability of occurrence of tornado in the British Isles, rather than the higher England specific data. However, these are both judged to be a small gap as the PSA study presents very low conditional core damage and large release frequencies for tornado events. Therefore, I conclude that the potential optimisms in frequency present a small impact on the UK ABWR risk profile. A site specific hazard characterisation will be required to be performed by any future license and is expected to address these shortfalls.

375. A significant claim in the PSA study on tornado missiles is operator commissioning of FLSR within 8 hours. FLSR is a mobile system, with the storage location considered out of the scope of GDA by Hitachi-GE. Substantiation of the availability of FLSR following external hazard events should be performed in the site specific phase.

376. My evaluation of the findings in this particular area of the PSA has shown that the most significant numerical gaps could be associated with the omission of LUHS and external flooding events; these shortfalls are captured in assessment finding AF-UKABWR-PSA-006.

4.2.13.4 CONCLUSION

377. From the submissions and information provided by Hitachi-GE in response to RO-ABWR-0041, a number of relevant RQs responses, and the outcome of my assessment, I have reached the conclusion that sufficient evidence, proportionate to this stage of the project, has been presented for me to adequately understand the risk profile of the UK ABWR due to external hazards and support closure of RO-ABWR-0041 (Ref. 151).

378. I conclude that the identification, prioritisation, scope and analysis of external hazards for UK ABWR PSA meets many of the regulatory expectations in ONR's PSA TAG (Ref. 4), with specific shortfalls being identified above.

379. I judge, based upon the information evaluated, that external hazards (excluding seismic; see Section 4.2.14) considered within GDA are not dominant contributors to the UK ABWR overall risk profile. However, LUHS due to external hazard has the potential to be a significant contributor to the UK ABWR overall risk profile and requires further analysis in the site specific phase.

380. The analysis performed for GDA is comprehensive and well documented, however it is generic and defers consideration of a number hazards to the site specific phase. Therefore, prioritisation and assessment of external hazards should be revised after GDA, taking site specific characteristics into account.

4.2.14 Level 1 PSA: Analysis of Hazards – Seismic Analysis (A1-2.7-4)

4.2.14.1 ASSESSMENT

381. My Step 3 review of the methodology in this area (Ref. 77) established that Hitachi-GE planned to submit a seismic margins analysis (SMA) to assess seismic risk. Although

the approach presented met some of ONR's expectations, the information provided was too general. In particular, the following issues were raised:

- There was a lack of clarity regarding how the level 2 PSA aspects were going to be treated and how the results will be compared against ONR SAPs probabilistic numerical targets.
- There was a lack of clarity on the approach to consider other operational modes, hazards (eg seismic internal flood interactions, etc) and radioactivity sources outside the reactor at power.

382. In response to my review comments, Hitachi-GE revised the seismic analysis methodology (Ref. 33) and produced a level 1 and level 2 seismic PSA (SPSA) (Ref. 152) covering the reactor at power, consideration of shutdown states and SFP. The hazard curve included in the PSA is based on preliminary information for the Wylfa site. The review of the adequacy of the derivation of the hazard curve has not been undertaken in GDA. However, I have confirmed with ONR's external hazards team that it can be considered to represent a typical curve for a UK site, although it may be somewhat conservative at the $1x10^{-4}$ per annum return period frequency.

383. The seismic fragility methodology (Ref. 153) was submitted to ONR at the end of Step 3.

384. In Step 4, I conducted a detailed review of some aspects of the UK ABWR SPSA submitted for GDA with the support of ONR's external hazards team. This review focused on sampling of specific fragility derivations. This sampling was based on the following:

- PSA risk importance measures;
- additional fragilities selected to cover each SSC type;
- dominant failure modes relating to steel members, foundation bolts, shear walls, ceramic insulators and damper functions were included in the selected fragilities and this covers the majority of failure mode types;
- floor levels (ensuring that the sampled covered the range from basement to high building elevations);
- high confidence low probability of failure (HCLPF) values (ensuring the two lowest HCLPF values per SSC type were reviewed); and
- a wide range of fragility derivation methodologies, including the probabilistic fragility method, similarity scaling, and use of generic databases.

385. My review of seismic fragility also included the evaluation of major assumptions listed in Hitachi-GE's documentation. Most of the major assumptions were investigated via the review of the sampled fragilities, and the remaining were briefly reviewed to ensure completeness of the overall assessment.

386. In response to queries and comments from my review, Hitachi-GE submitted several revisions of the seismic fragility derivation and methodology with a final version produced taking account of all comments (Ref. 154).

387. The detailed review of the SPSA was conducted by ONR's PSA team in Step 4. The following main technical aspects of the UK ABWR SPSA have been reviewed primarily on a sampling basis:

- whether the seismic equipment list covers level 1 PSA, level 2 PSA, shutdown and spent fuel pool SSCs;
- whether the accident sequence modelling is reflective of plant design and systems;

- whether SSC fragilities are appropriately included in modelling;
- whether SSC fragility correlation modelling is adequate;
- seismic-induced relay chatter assessment, and other secondary seismic effects;
- level 2 SPSA, including level 1 to level 2 interface;
- the quantification process; and
- the interpretation and review of results.

388. In response to queries and comments from this review, Hitachi-GE submitted several updates of the SPSA (Refs 155, 156, 157 and 158). In addition, a final revision of the SPSA was provided in June 2017 to reflect the updated fragility values to address my review comments (Ref. 152). The changes compared to previous revisions were small and therefore I did not judge it necessary to review this last update.

389. For the remaining shortfalls related to the SPSA, I requested Hitachi-GE to perform sensitivity analyses to evaluate the risk impact of my review findings. These sensitivity analyses were provided in Ref. 8. I considered the adequacy of these analyses as part of my review. I used some of the insights of these analyses, in combination with qualitative arguments and quantitative information from the PSA, to understand the potential risk significance of the findings in this area.

390. My assessment was performed using the latest available models and documentation provided by Hitachi-GE, which are described in Sections 2 and 3 of this report. My assessment was carried out against expectations in ONR's PSA TAG (Ref. 4).

4.2.14.2 STRENGTHS

391. The SPSA quantitatively models both level 1 and level 2 aspects related to reactor and SFP radionuclide sources for both at power and shutdown conditions. The SPSA is broadly consistent with modern standards and ONR expectations in ONR's PSA TAG (Ref. 4), although, as identified below, it is expected that further development will be needed during the site specific stage. The level 2 PSA provides input into the level 3 PSA, the review of which is documented in a separate section of this report.

392. Hitachi-GE has generally applied relevant good practice (RGP) in the derivation of the seismic fragilities for use at GDA, with some limitations identified by ONR's review and summarised below. Hitachi-GE has specifically identified and documented seismic fragility GDA major assumptions, including a brief discussion on their validity and possible approaches for dealing with them at a later stage. All of the major assumptions were found by ONR's review team to be acceptable for GDA.

4.2.14.3 FINDINGS

4.2.14.3.1 Seismic Fragility Analyses

393. The detailed technical reviews that support the findings discussed in the following paragraphs are documented in Refs 159 and 160.

394. Due to the lack of detailed design information and site specific information at GDA, it has not been possible for Hitachi-GE to always adopt an approach with the same depth and level of detail expected during the site specific phase. Specific methodological concerns identified by my review of the sampled seismic fragilities have been summarised below.

395. A number of fragility calculations have been derived using a conservative approach, usually due to simplistic assumptions being made. The approaches used to conservatively derive these fragilities include the following:

- Using success data from testing undertaken in Japan as failure capacities. In essence, a stepped profile fragility has been adopted (ie zero variability) based on the lowest value of success data from this testing without knowledge of the threshold for the loss of functionality.
- Using zero variability for certain parameters but adopting corresponding conservative scale factors.
- Under-predicting of the reactor building (R/B) spectral shape factor by adopting fixed base natural frequency.

396. Some fragility calculations were noted to not be in accordance with RGP and guidance documents identified by Hitachi-GE and my review team (Refs 161, 162, 163 and 164). Hence, these are potentially non-conservative. The approaches used to derive these values include the following:

- adopting 0.8 for vertical-to-horizontal ratio for the derivation of GDA seismic design spectra, with no consideration of the variability on this ratio when deriving fragilities;
- using variability in frequency directly rather than propagating this through to obtain the actual variability for structural response;
- not following the guidance in Ref. 161 for fragility computations based on dynamic testing; and
- basing the fragilities on an European Utility Requirements (EUR) spectrum which is the maximum of two horizontal directions whereas the hazard curve is expressed as geometric mean of two directions.

397. A simplified approach has been adopted for a number of cases due to the lack of detailed design and site specific information, including the following practices:

- averaging across the seismic frequency range to calculate parameters that should be frequency-specific;
- not considering separate wall piers for the R/B capacity calculation;
- the omission of the evaluation of buckling due to lack of information regarding configurations;
- adopting Japanese variabilities in similarity scaling rather than UK specific variabilities;
- assuming utilisations for anchorage will be limited to 50% for some items at the design stage; and
- assuming all SSCs are located in the R/B, as the R/B is the only analysis model currently available (although the correct floor levels were selected).

398. The numerical values used to generate the seismic fragilities were found by ONR's review team to be conservative when the conservative approach was claimed to be used. For fragility calculations that were not in complete accordance with RGP and guidance documents identified in Hitachi-GE's methodology, there are potentially optimisms in the numerical values that are used in the calculations. For fragility calculations that followed simplified approaches due to lack of detailed design information and site specific information, it is unknown whether the numerical values used result in a conservative or optimistic fragilities compared to a more thorough evaluation during the site specific phase.

399. At this stage, it is not possible to assess the influence of the above limitations on future fragilities based on more accurate design and site specific information. Overall, the approach adopted is considered by my review team (with expert knowledge on seismic

fragility analysis) to be reasonable for GDA, with the following key limitations that will require consideration beyond GDA:

- The treatment of major openings in the R/B should be explicitly documented.
- The modelling parameter variability should be considered for individual buildings and if not propagated through; its direct use should be demonstrated to be conservative.
- The treatment of capacities should be revised to follow RGP.
- Buckling failure modes for tanks, piping and R/B crane should be evaluated.
- The assumption regarding the strength factor for anchor bolt design should be revisited when design information becomes available, in order to adequately represent the seismic fragility of the equipment.
- Flexural failure should be considered in the R/B shear wall piers.
- Adjustment to the hazard curve should be considered because the hazard curve is expressed as a geometric mean of two directions, but the fragilities have been derived based on a EUR spectrum which is the maximum of two horizontal directions.
- The GDA approach to ducts and dampers is simplistic and it should be revised in line with RGP.
- The crane wire rope design accelerations should be analysed in line with relevant good practice.
- Site specific fragilities should be evaluated for batteries and other electrical items.

400. In addition to the above, Hitachi-GE's work in GDA has required certain assumptions to be made due to limited information being available; these assumptions are in general specifically identified. The supplier and design of many SSCs is yet to be chosen, and at the time of the origin of the fragility work, only limited building analysis work had been undertaken, covering only the R/B. The assumptions identified by Hitachi-GE should be used to inform the detailed design and will need to be reviewed once the detailed design information becomes available.

4.2.14.3.2 Seismic PSA

401. The main concerns identified during the review of the SPSA model and documentation are summarised in this section.

402. The scope of the SPSA does not fully consider shutdown states and all the relevant SFP POSs. This is mainly due to a lack of information related to the detailed design and operating procedures. The information provided by Hitachi-GE was sufficient to demonstrate that the contribution from these states is small (and the overall risk will remain well within the SAP BSLs (Ref. 1)). Specific limitations in the scope of the SPSA are presented below:

- Hitachi-GE describes the shutdown seismic analysis for GDA as a semi-quantitative analysis that addresses only level 1 PSA. The purpose of the analysis was to provide confidence in GDA that that the CDF is much lower than at power. Further development will be needed beyond GDA.
- The modelling of the crane collapse in the shutdown SPSA (reactor and SFP) is limited to POS B1 and POS B2 when the reactor well is flooded up to the level of the SFP. These two POSs represent 50% of the modelled shutdown schedule. POSs A, C and D (which represent the other 50% of the shutdown schedule) are not modelled with the potential for seismic-induced crane collapse onto the SFP. In view of the lower risk resulting from this analysis and

the overall results of the seismic PSA, I have confidence that the risk associated with seismic events on the SFP is small. However, further development is needed beyond GDA.

403. A small number of modelling simplifications have been made in the SPSA. Hitachi-GE has demonstrated, through sensitivity analyses, that the impact of these simplifications on the PSA results is small or that they result in a conservative risk estimate. However, these approaches are not consistent with RGP and should be revised in the future:

- The SPSA does not include quantification of cutsets for the final seismic hazard interval. Hitachi-GE has argued that this interval is bounded by the final quantified interval modelled as an exceedance interval. It is acknowledged that the difference in the PSA results is expected to be non-significant. However, this approach is not consistent with RGP. This applies to each of the SPSA models (ie, reactor at power and shutdown SPSA, SFP SPSA).
- Additional clarification and justification of why the impact of seismic-induced failure of the reactor building does not lead directly to core damage need to be provided or the PSA model needs to be extended to adequately consider these failure modes. It is acknowledged that given the seismic capacities of these structures, a small change in risk results is expected, which has been confirmed by a sensitivity study undertaken by Hitachi-GE.
- It has been assumed that class 3 systems are not suitably robust to withstand seismic events and are therefore not claimed in the SPSA, although they are considered with respect to inducing initiating events. This modelling assumption is considered to be conservative. A sensitivity study was performed by Hitachi-GE to investigate the change in risk prediction from claiming the class 3 MUWC system for the SFP SPSA; this study revealed that a reduction in fuel damage frequency (FDF) and LRF of approximately 10% could be experienced if MUWC was claimed with a reasonable seismic fragility. Further development of the PSA will be needed in the future to avoid undue conservatism and masking of risk insights.

404. Some of the identified simplifications are specifically due to lack of detailed design information or site specific information in GDA. Once this information becomes available the PSA should be updated. For example:

- The fragility for the reactor vessel instrument system (RVI) small LOCA is assumed to be uncorrelated (seismic-induced failure occurs only in one division of the RVI) in the UK ABWR SPSA. This approach is based on Ref. 165 which states that the RVI will be designed to preclude dependent seismic failures of the RVI. On the basis of this information provided by Hitachi-GE, my review team considers that the SPSA and design approach pursued by Hitachi-GE is reasonable and appropriate, but implementation of the design intent cannot be confirmed until plant construction.
- The SPSA report (Ref. 152) discusses the topic of seismic-induced relay chatter and lists relays that are included in the generic design for the systems modelled in the PSA. My review team considers that the information provided is reasonable at this stage. However, explicit assessment of relay chatter and incorporation of relay chatter fragilities in the SPSA model is needed beyond GDA.
- Earthen buried dams and other seismically susceptible structures are identified for inclusion in the seismic equipment list (SEL) and the seismic analysis. Hitachi-GE revised the methodology report (Ref. 33) to specifically state earthen and buried items are site specific and the analysis of these items and inclusion in the PSA is deferred to the site specific phase. My review team agrees that the seismic-induced failure of earthen or buried items and the

potential impact on a site needs to take into account site specific information. This topic should be addressed in a site specific manner (which will require undertaking walkdowns).

These shortfalls should be addressed, or a plan provided for how they will be addressed as the detailed design is developed, as part of the resolution of AF-UKABWR-PSA-001.

405. Hitachi-GE has provided a qualitative assessment of key seismic secondary hazards which provides me with confidence that their contribution to the overall risk is expected to be small (and the overall risk will remain well within the SAP BSLs (Ref. 1)). This evaluation identifies design considerations that should be used to inform the detailed design phase. Quantitative analysis will also be needed at a site specific stage. My review has identified the following specific concerns that will need consideration beyond GDA:

- Ref.152 discusses the topic of seismic-induced internal flooding and seismic-induced internal fires and identifies design considerations to reduce the seismic-induced flood and fire risk (see Sections 4.2.11 and 4.2.12). These insights should be used to inform the detailed design phase. The seismic-induced internal flooding and seismic-induced internal fires should be addressed in the PSA in a site specific manner (including walkdowns).
- Ref. 152 discusses the topic of seismic-induced failure of masonry block walls and their potential impact on safety functions. The SPSA report includes a qualitative FMEA type table of block wall failure impacts and identifies block walls in the generic design and distinguishes those with the potentially highest risk impact from seismic-induced failure. In addition, the qualitative analysis identifies design considerations (eg, steel beams) to reduce the risk impacts of postulated seismic-induced block wall failure. These insights should be used to inform the detailed design phase. This topic should be addressed in the PSA in a site specific manner (including walkdowns).
- Ref. 152 discusses the topic of seismic-induced external dam failure leading to external flooding of the site or loss of the intake level. The SPSA report defers the assessment of this topic to the site specific stage. My review team agrees that seismic-induced failure of upstream or downstream dams, or similar structures, and the potential impact on a site is a site specific issue that need to be addressed in the PSA in a site specific manner (including walkdowns, as appropriate).

406. Hitachi-GE's HF team has not developed analyses to support the SPSA HEPs. The current approach is based on adjustments to the IEAP PSA HEPs to take into account the impact of a seismic event. The adjustments follow a methodology described in Ref. 166, which is judged reasonable for the GDA stage. Hitachi-GE has also performed sensitivity analyses conservatively assuming higher stress factors; which resulted in a negligible increase in the large release frequency (LRF) associated with the reactor but a more significant increase of 25% for the LRF associated with the SFP. This is likely due to the higher reliance on operator actions in response to faults affecting the SFP. However, it should be noted that longer timescales available to the operator were not credited to reduce the operator stress. Site specific human factors analysis supporting the SPSA HEPs and the treatment of dependencies should be developed to reflect the site specific characteristics and procedures. My review has also identified that the current documentation does not explain the treatment of pre-initiator HEPs for the SPSA and should be extended. Hitachi-GE have informed ONR that this shortfall has been considered in an addendum to revision F of the HRA report (Ref. 110), however due to the late submission within GDA of this report the PSA assessment was conducted on revision E (Ref. 111).

407. My review has identified that the SPSA documentation needs to be extended to provide additional information and justification related to some aspects of the PSA, including the following:

- Consistent with the SPSA methodology (Ref. 33), the seismic hazard curve is extrapolated beyond approximately 1.2g peak ground acceleration (PGA). Due to the lack of site specific information, the SPSA documentation provides limited explanation and justification for the extrapolation scheme used and its potential impact on the results. This applies to each of the SPSA models (ie, reactor SPSA, SFP SPSA and shutdown SPSA).
- The SEL (Ref.167)[2] does not include columns for the SSC status (ie normal, post-demand; loss of pneumatic supplies; loss of electric supplies, etc). This information is typical of SELs as it is required to confirm whether active failure mode fragility applies to a given SSC. It should also be explained how this information has been taken account in the SPSA.
- There is limited discussion of the shutdown seismic event tree accident sequence modelling in the documentation.

These are captured as a minor shortfall.

408. As noted above, Hitachi-GE has provided a number of sensitivity analyses to evaluate the impact of my review findings on the risk (Ref. 8). None of the identified shortfalls resulted in a significant increase in the risk results, but the sensitivity analyses highlighted that the results are sensitive to the SFP HEPs.

409. Overall, the shortfalls identified by my review are primarily related to the lack of detailed design, operating procedures and site specific information (which results in PSA relying on a number of assumptions). Once the information becomes available during the site specific stage, the risk profile may change. However, the information provided in GDA was sufficient to give me confidence that analysis is likely to be conservative at the GDA stage, but further work will confirm this beyond GDA.

4.2.14.4 CONCLUSION

410. Based on the outcome of this assessment, I have concluded that the SPSA developed by Hitachi-GE is sufficient to support the UK ABWR ‘generic’ PCSR.

411. It is important to note that the SPSA has identified that the risk associated to seismic events for the UK ABWR can be significant (in comparison with the risk from internal events), but this is dependent on specific characteristics of each site. To support future stages of development of the NPP, the SPSA and seismic fragility analysis needs to be revised to take into consideration, in as a realistic manner as possible, site specific characteristics and plant specific design. The SPSA should also be extended to address the issues that could not be considered during GDA due to the need of site specific and detailed design information.

412. The licensee is expected to develop a site specific SPSA as part of normal business during the detailed design phase beyond GDA. However, I note that the seismic PSA, at its current stage of development, suggests that seismic events are likely to be the dominant risk from the plant. For this reason, during the site specific stage, it is essential that the future licensee addresses the shortfalls and SPSA developments identified in this report at an early stage. I will maintain regulatory oversight in this area through resolution of assessment finding AF-UKABWR-PSA-001.

4.2.15 Level 1 PSA: Low Power and Shutdown Modes (A1-2.8)

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[2] *A further revision of this report was submitted late in GDA Step 4, which was not included in the PSA assessment.*

4.2.15.1 ASSESSMENT

413. My Step 3 review in this area (Ref. 77) covered the methodology, identification and grouping of initiating events and source term analyses approach. Shortfalls related to the following areas of the PSA were identified:

- Plant Operational States (POSs) characterisation definition and analyses did not include all the critical aspects of the POSs characterisation.
- The approach and criteria adopted to screen out initiating events did not meet regulatory expectations and the list of initiating events was not complete.
- Accident sequences analyses and success criteria methodology did not meet regulatory expectations.
- The methodology provided did not adequately address the level 2 PSA.

414. At the end of Step 3, I raised RQ-ABWR-0610 to seek confirmation from Hitachi-GE that these shortfalls were going to be addressed as part of the resolution of RI-ABWR-0002.

415. In response to my review, Hitachi-GE provided a shutdown PSA at the beginning of Step 4 and updated it in July 2016 in part to take into account my Step 4 review comments. The scope of the shutdown PSA includes the reactor core for those periods when the reactor well gate is closed. When the reactor well gates are open, the scope of the shutdown PSA also includes the risk of damage to fuel stored in the spent fuel pool (SFP). A separate SFP PSA has been produced to analyse the period when the reactor well gate is closed. The review of the fuel route and SFP PSA is documented in Section 4.2.16 (including heavy load drops frequency derivation).

416. The scope of my review in Step 4, has covered all the aspects of the shutdown PSA including:

- identification and grouping of IEs, derivation of the initiating event frequency (IEF) and their applicability to each POS (Refs 168 and 169);
- success criteria supporting analysis (Ref. 170);
- event tree analysis (Ref. 170);
- system and dependency analysis (Ref. 171);
- level 2 PSA for shutdown conditions (Ref. 17);
- quantification analysis (Refs 16 and 17);
- results of the analysis when compared with a typical BWR shutdown risk profile;
- internal fire PSA for the shutdown states (Ref. 25); and
- internal flood PSA for the shutdown states (Ref. 27).

417. The review focused on SSCs that are typically important during shutdown or are dominant contributors to the risk. For example, a detailed review of the treatment of support system IEs was undertaken; RSW/RCW and AC safety related divisions were sampled. The systems analysis review focused on shutdown cooling (SDC), fuel pool cooling and clean-up system (FPC) and AC power.

418. In addition, Hitachi-GE provided sensitivity analyses to evaluate the impact of some of the shortfalls identified by my review (Ref. 6). I have considered the adequacy of these analyses. I have also used some of the insights of these analyses in combination with qualitative arguments and quantitative information from the PSA to understand the potential risk significance of the findings in this area.

419. My assessment was performed using the latest available models and documentation provided by Hitachi-GE, which are described in Sections 2 and 3 of this report. My assessment was carried out against expectations in ONR's PSA TAG (Ref. 4).

4.2.15.2 STRENGTHS

420. The Plant Operational States (POSs), as defined in Annex 8, during non-full power modes have been clearly defined, characterised and consistently used through the revised shutdown PSA. This includes the consideration of unique UK ABWR features.

421. The analysis of IEs has considered plant failures and operator interactions based on industry operating experience.

422. The shutdown PSA includes system models developed specifically for the shutdown PSA (eg, SDC, FPC), as well as those incorporated from the IEAP level 1 PSA, which are integrated consistently with the characteristics of each POS.

423. An extensive list of accident sequences is provided with links to initiating event logic, functional fault trees, and level 1 end states, although a number of shortfalls related to simplifications and potential conservatisms of the analysis have been identified below.

424. The shutdown PSA has been extended to cover level 2 and level 3 PSA. Detailed deterministic analyses were performed for the severe accident phenomena that apply during shutdown modes (eg, molten core concrete interaction (MCCI), hydrogen burning and explosions). The deterministic analyses supported the evaluation of the magnitudes and timings for the shutdown level 2 PSA release categories.

4.2.15.3 FINDINGS

425. The detailed technical reviews that support the findings discussed in the following paragraphs are documented in Ref. 172.

426. As noted previously the POSs considered in the shutdown PSA were clearly defined. However, it should be noted that the POSs identified for the PSA do not align with the operating states identified in the technical specifications (Ref. 173)[3].

427. The T&M schedule for each POS is identified in the PSA documentation; however it has not been updated to reflect the final GDA position, such as the number of divisions available in POS C. Upon request, Hitachi-GE provided a qualitative discussion of the impact of some of the changes (Ref. 35), which concludes that the updated outage schedule would result in a benefit being seen in the PSA results. The PSA is expected to be used to risk inform the outage and T&M schedules for the UK ABWR in the site specific phase.

428. Furthermore, the review noted that the assumed configuration of the plant during some of the POSs is conservative. Specifically it was considered that some of the assumptions regarding the water level and status of the RPV head in POS A and POS C were unduly conservative, with the worst cases being assumed for the duration of these states. Whereas, in reality, these configurations are only applicable for a fraction of the total POS duration. Hitachi-GE undertook a sensitivity study which revealed that the refinement of the plant configuration during these states would yield a significant risk reduction. Once site specific procedures are developed, the POSs should be aligned with technical specifications and refined to ensure no undue conservatisms are introduced into the shutdown PSA.

[3] *A further revision of this report was submitted late in GDA Step 4, which was not included in the PSA assessment.*

429. As noted previously (see Section 4.2.8), during GDA there was not sufficient information for Hitachi-GE to undertake a systematic examination of procedures for changing configurations and equipment T&M procedures to identify potential human errors during the execution of such normal procedures that may lead to initiating events. This review should be undertaken and the list of initiating events potentially expanded during the site specific stage.

430. Furthermore, although the most significant low power initiating events are covered, a review of the site specific procedures and T&M schedule is required to confirm that the shutdown PSA adequately considers the unavailability of systems such as containment isolation and electrical systems. In particular, justification should be provided that a rod withdrawal error at low power is adequately addressed by the PSA. The discussion should include whether the event is physically possible and, if so, what the likelihood would be.

431. My review also identified that the LOOP frequencies used in the shutdown PSA were identical to the IEAP PSA. This approach may be optimistic, as the risk of a LOOP can be higher during shutdown, due to increased levels of maintenance which can lead to an increased frequency of plant-centred and/or switchyard based LOOP faults. When the LOOP frequencies are reassessed for the site specific phase of the project the increased contribution to LOOP from plant-centred and/or switchyard based faults should be considered for the shutdown POS (see Section 4.2.9).

432. Most success criteria analyses supporting the shutdown PSA have been developed using realistic bases. However my review has identified the following exceptions that could result in distortions of the risk profile:

- The PSA conservatively considers fuel damage when the RPV water level drops below the top of active fuel (TAF). In POS S this condition was assumed much earlier than potential fuel damage would occur. Upon request, Hitachi-GE clarified that this conservatism only applied for fault sequences where high pressure faults led to fuel rods being uncovered. A sensitivity study performed by Hitachi-GE suggested that this conservatism resulted in a minor overestimation in the risk prediction.
- Some operator recovery actions are not considered even when it is apparent that long timescales are available. Hitachi-GE performed a sensitivity study to assess the impact of claiming certain operator recovery actions, which indicated that the inclusion of these claims in the PSA would yield a significant risk reduction.
- Fire protection makeup to the SFP for POS B (modelled as part of the shutdown PSA scope) is not claimed. Hitachi-GE has undertaken a sensitivity study which identified that if the fire protection system was modelled in the PSA it could result in a significant risk reduction.

433. The review has identified that a number of event trees will need to be modified to adequately represent the progression of the accident sequences or for which additional substantiation is required:

- The PSA assumes that the primary circuit SRV sheets are the weakest point of the primary circuit pressure boundary in case of RPV over-pressurisation. This assumption impacts the fault sequence as Hitachi-GE's analysis considers that the primary circuit cannot be sufficiently depressurised via the failed SRV sheets to enable high pressure makeup. This assumption has not been substantiated and could be overly conservative. This assumption is inconsistent with the also unsubstantiated assumption in the at power level 1 PSA model (see Section 4.2.6).

- The PSA assumes that breaks outside containment (BOCs) only damage equipment in the division where the BOC occurred. However, no substantiation has been provided and this assumption could be optimistic. As reported below, a sensitivity analysis performed by Hitachi-GE assumed that an operator action to isolate the BOC could prevent the failure of all divisions. A similar issue has been raised for the level 1 IEAP PSA (see Sections 4.2.5 and 4.2.6).
- Modelling of BOCs implicitly assumes that the operators would manually control the RPV water level during these scenarios to conserve feed water stocks; it should be noted that an identical issue was identified for the level 1 IEAP PSA. The PSA should be extended to explicitly include operator actions needed to establish and control the feed from the relevant water sources.
- FLSR preparation is assumed to occur in parallel following the onset of water level decrease in POS D. However, this assumption could be optimistic if this approach was found to be overly burdensome for the operators, especially when other systems are available.
- Modelling of LOCAs above normal water level (NWL) did not account for their environmental impacts on the reactor building (and associated SSCs) representing a potential optimism.
- No credit was taken for the operators successfully injecting water into the reactor once the fuel had been uncovered. Hitachi-GE clarified that this omission was not necessarily conservative, with late water injection having the potential to increase the severity of the consequences. Further analysis of the impact of water injection after the fuel assemblies are uncovered should be performed with a view to providing clear guidance to the operators on the best course of mitigation action.
- My review of the PSA results, as reported in Ref. 35, identified many of the most significant components as those in the combined FLSS and FLSR injection route to the RPV. In Ref. 35, Hitachi-GE identifies that alternative injection routes are available; these alternative routes should be modelled explicitly in the PSA.

434. Similarly, as for the at power PSA, the review raised queries regarding the potential for intersystem CCFs including the potential for a CCF between the EDGs, BBGs and the FLSR mobile pumps (assumed to be diesel–driven pumps).

435. My review also identified that the SRV failure data used in the PSA encompasses active and passive failures of the SRVs. However, there are cases in which this is applied in sequences when only the passive failure mode is relevant. Hitachi-GE performed a sensitivity study to understand the impact on risk of this conservatism. Due to the risk importance of the SRVs, this data assignment has a significant impact on the shutdown PSA results and may distort the risk profile. More appropriate ‘best estimate’ failure data for the SRV failure modes should be integrated into the PSA and properly justified.

436. Simplified internal fire and flood analyses have been undertaken for shutdown states. These analyses indicate that the risk arising from these hazards during the shutdown states is lower than the at power states, however the at power analysis indicates that internal fire and flood are some of the dominant contributors to risk for the UK ABWR. In addition, the at power analysis has been refined during GDA, to address conservatisms and design development;, the shutdown analysis has not been refined in a similar way. The shutdown states analysis should be extended as required to be consistent with the at power internal fire and flood PSAs and reflect the site specific design, operation and maintenance of the UK ABWR. My assessment of the internal fire and internal flooding PSAs can be found in Sections 4.2.11 and 4.2.12, respectively.

> ***Assessment Finding AF-UKABWR-PSA-007****: The licensee shall provide revised internal fire and internal flood PSAs for shutdown and spent fuel pool operations which are consistent in detail and scope to the at power analysis. The revised PSAs shall reflect the site specific design, operation and maintenance of the UK ABWR and take any relevant shortfalls identified by the GDA review into account.*

437. Evaluation of point in time risks has not been provided by Hitachi-GE. However, I undertook a calculation for each shutdown POS based on the PSA results for CDF and POS durations. My calculation confirmed that the point-in-time risk remains below the BSLs and meets the regulatory expectations outlined in SAP NT 2 (Ref. 1). I did not have sufficient information to confirm this for a full scope shutdown PSA as hazards are currently not fully considered in this PSA model. The analysis should be provided when sufficient information becomes available beyond GDA.

438. I have conservatively evaluated the impact on the risk of the shortfalls identified in my review. I have identified that the following shortfalls could have the highest impact on the risk profile of the shutdown PSA:

- The potential underestimation of LOOP frequency during shutdown as shown by Hitachi-GE's sensitivity analysis using U.S. operational experience (Ref. 174).
- Hitachi-GE developed a sensitivity study assuming a BOC results in damage on all divisions and found that this could lead to a significant increase in risk. However, the sensitivity study also noted that this damage spread would be caused by flooding, which would usually be isolated by the operators before the flooding reached a level where it could spread to another division.
- Hitachi-GE performed a sensitivity study to investigate the importance of potential environmental impacts on the reactor building due to LOCAs above NWL. The study assumed that only FLSS and FLSR were available due to adverse environmental conditions in the reactor building. The study revealed that if these conditions were included within the model, then a significant increase in risk would occur (a seven fold increase in the CDF for the shutdown PSA). It is expected that substantiation of equipment survival will be provided in the site specific phase. However, it is also recognised that the success criteria for the shutdown PSA in response to this fault are conservative. For example, there is significant time available to provide makeup to the SFP to respond to the fault (in excess of 29 hours) and that not all available mitigation systems are claimed, such as the firefighting system. It is also recognised that the key contributor to LOCAs above NWL are heavy load drops, which were considered to be conservative in terms of frequency (see Section 4.2.16) it is also noted that Hitachi-GE claim that the consequences of dropped loads modelled in the PSA are also conservative (Ref. 6). Taking these factors into account it is considered that the position is acceptable for GDA, however substantiation of equipment survival or further demonstration that the risks are reduced ALARP will be required in the site specific stage.

4.2.15.4 CONCLUSION

439. Based on the outcome of this assessment, I have concluded that sufficient analysis has been performed to have a good understanding of the level of risk and the risk profile from the UK ABWR during shutdown and support the UK ABWR 'generic' PCSR.

440. A number of shortfalls have been identified, however the risk associated with the shutdown states is low and the current estimation is likely to be conservative and should be refined in the site specific stage. Many of the shortfalls are expected to be

resolved following site specific operational, design and procurement information becoming available.

4.2.16 Level 1 PSA: Spent Fuel Pool and Fuel Route PSA

4.2.16.1 ASSESSMENT

441. I conducted a review of the SFP and fuel route PSA methodology and initiating events identification in Step 3 (Ref. 77). My review identified shortfalls related to the following areas of the PSA:

- Plant operational states (POSs) characterisation definition and analyses did not include all the critical aspects of the POSs characterisation.
- The approach and criteria adopted to screen out initiating events did not meet regulatory expectations and the list of initiating events is not complete.
- Accident sequences analyses and success criteria methodology did not meet regulatory expectations.
- The methodology provided did not adequately address the level 2 PSA.

442. RI-ABWR-0002 identified the regulatory expectation for a full scope PSA. In addition I captured Step 3 specific shortfalls following the review of the SFP PSA in RQ-ABWR-0609.

443. In response to my review comments, Hitachi-GE revised the PSA for Step 4 GDA and produced a SFP PSA covering all POSs, with detailed documentation of initiating event identification and accident sequence analysis, and extended it to level 2 PSA. The level 1 SFP PSA covers all POSs when the reactor well gates are closed. POSs where the reactor well gates are open are considered in the level 1 shutdown PSA, discussed in Section 4.2.15. The level 2 SFP PSA considers all POSs.

444. In addition, RO-ABWR-0041, raised at the end of Step 3, requested Hitachi-GE to provide an internal hazard prioritisation. Following the internal hazard prioritisation, Hitachi-GE identified the need for a fuel route PSA, including dropped loads initiating events, which was submitted to ONR in Step 4. Assessment of the internal hazard prioritisation is presented in Section 4.2.10.

445. Similarly as for the shutdown PSA, in Step 4 I carried out a detailed review of the SFP PSA, CAFTA fault tree and event tree model (Refs 175 and 176) and supporting documentation, covering the main technical aspects of the UK ABWR SFP PSA. The systems selected for detailed assessment, due to their risk significance and the high-level of claims made upon them in the SFP PSA, were:

- FPC
- AC power supplies
- RSW/RCW

446. The fuel route PSA is contained in a separate fault tree and event tree model to the SFP PSA and is reported in the fuel route (including dropped loads) PSA topic report (Ref. 177). The fuel route PSA documentation identifies all dropped loads initiating events to be considered in the PSA. However, the fuel route PSA only considers mechanical failure of fuel, with any dropped load initiating events which lead to thermal failure of fuel dispositioned to the SFP PSA or the shutdown PSA. The fuel route PSA also considered LOOP events during fuel cask movement.

447. My review team reviewed the initial Step 4 submission of the fuel route PSA, fault tree and event tree model (Ref. 177) and supporting structural analysis (Ref. 178). The review was based on a sampling approach which focused of the aspects of the fuel

route PSA model, documentation and supporting analysis which were expected to be critical to an accurate assessment of the risk profile.

448. The following categories of initiating events and accident sequence analysis, were reviewed:

- heavy load drop impacting irradiated fuel in the reactor or SFP;
- cask drop onto spent fuel storage rack in the SFP;
- cask drop onto cask pit or preparation pit;
- cask drop into the truck bay; and
- LOOP during cask handling.

449. The following systems were reviewed:

- reactor area (R/A) HVAC;
- standby gas treatment system (SGTS);
- impact limiter; and
- canister cooling system (CCS).

450. My Step 4 review was performed on the initial Step 4 submissions of the SFP PSA (Refs 175 and 176) and fuel route PSA (Ref. 177), with the shortfalls identified in RQ-ABWR-1055 and RQ-ABWR-1090, respectively.

451. For the SFP PSA, in response to the shortfalls I identified, Hitachi-GE submitted updated revisions of the SFP PSA and associated documentation (Refs 14 and 15). I reviewed these submissions to determine if they adequately addressed the shortfalls. For the fuel route PSA, I raised RQ-ABWR-1257 to follow-up and request further information on some of the shortfalls identified in RQ-ABWR-1090. Hitachi-GE submitted a revised fuel route PSA to address the shortfalls I had identified (Ref. 179). A further revision of the fuel route PSA (Ref. 18) was submitted late in Step 4 to update the seismic fragility data used; this submission was not assessed.

452. Hitachi-GE also provided sensitivity analyses to evaluate the impact of some of the shortfalls identified by my review (Ref. 7). I have considered the adequacy of these analyses and used some of the insights, in combination with qualitative arguments and quantitative information from the PSA, to understand the potential risk significance of the findings in this area.

453. It should be noted that ONR's PSA TAG (Ref. 4) does not include specific expectations for SFP or fuel route PSA. However, my review team found that the general requirements for level 1 and level 2 PSA and for low power and shutdown modes set out in ONR's PSA TAG (Ref. 4) were generally applicable and were considered as part of the review. My review team has a significant amount of expertise and experience in BWR SFP and fuel route PSA, which was used to identify relevant good practice (RGP) in this area of the UK ABWR PSA.

4.2.16.2 STRENGTHS

454. The SFP and fuel route PSAs quantitatively model internal events related to spent fuel pool and fuel route radionuclide sources for all POSs in an integrated PSA model to both level 1 and level 2.

455. The SFP and fuel route PSAs are broadly consistent with modern standards and ONR's expectations in ONR's PSA TAG (Ref. 4) (although it, as identified below, is expected that further development will be needed during the site specific stage). The PSAs also provide inputs to the level 3 PSA.

456. The identification of the initiating events for the SFP PSA is performed using a systematic process that makes use of:

- a master logic diagram to group similar challenges;
- a review of previous BWR SFP PSAs;
- a comprehensive review of systems and their potential to cause SFP loss of cooling or loss of inventory, including review of FMEAs for design basis analysis (DBA) systems; and
- previous applicable BWR operating experience.

457. System models have been developed for the SFP PSA taking into consideration the specific characteristics of each POS. The SFP PSA identifies all relevant POSs, and is aligned with the shutdown PSA.

458. Sensitivity and uncertainty analysis has been performed and documented. Sensitivity analyses have been carried out to evaluate the risk significance of key assumptions.

4.2.16.3 FINDINGS

459. The detailed technical reviews that support the findings discussed in the following paragraphs are documented in Refs 180, 181, 182, 183, 184 and 185.

460. As noted above, the identification of the initiating events is performed using a systematic process; however a number of shortfalls have been identified related to the identification and characterisation of the initiating events considered in the SFP and fuel route PSAs which are detailed in the following paragraphs.

461. The dominant initiating event in the SFP PSA is loss of AC power in POS A due to type-B human error, contributing almost 50% to SFP LRF; I queried this in RQ-ABWR-1090. Discussions with Hitachi-GE revealed that the initiating event had two potential causes: local operator error at the AC switchgear and operator error in the MCR. As part of the RO-ABWR-0076 (Ref. 138) resolution a modification to prevent operator access to the switchgear, when energised, was identified along with potential improvements to the detailed MCR design to reduce the frequency of this initiating event. I expect that when these modifications are taken into account the SFP PSA fuel damage frequency (FDF) will be significantly reduced.

462. Loss of inventory in POS E is not included in the SFP PSA. The SFP PSA does not contain a dedicated event tree for loss of inventory in POS E and the SFP PSA documentation states that loss of inventory in POS E is considered within the analysis for loss of inventory for POS C. Review of the IEFs for POS C did not confirm this. Based upon review of the SFP PSA, loss of inventory during POS E is not expected to be risk significant; however the PSA should be complete and consider the loss of inventory in POS E explicitly.

463. The review of the SFP PSA during Step 3 identified a shortfall concerning the lack of supporting analysis to demonstrate that a dropped load event of a spent fuel cask into the SFP would not result in a large SFP leak (defined as greater than the ability of UK ABWR systems to make up the leak). At my request, Hitachi-GE submitted a report on SFP structural analysis (Ref. 178). This analysis was assessed by my review team, which included a structural integrity specialist. RQ-ABWR-1240 was raised to request Hitachi-GE to provide evidence that a dropped cask would not cause a SFP leak beyond the ability of the plant to make up. However, Hitachi-GE's response did not provide the information requested. In particular the review identified lack of clarity regarding the following:

- The relationship between stress / strain on the liner caused by dropped load and cracks leading to leakage (including through wall cracks).
- Vulnerability of the liner plate to rupture at joints where much higher stress / strain concentration is expected and could cause the joint to rip.
- The potential for a different orientation of the cask upon impact, such as impact by edge or corner of the cask.
- The presence of the water generating a compressive stress wave.
- The potential for reorientation of the cask by the tube bundles and other items in the SFP.

464. I judge that the impact on the risk profile is not significant based upon cask drop being a low frequency event and therefore assignment of a proportion of the cask drop events to a large leak would have a small impact on the UK ABWR risk profile. It should be noted that the cask drop frequency is not aligned with the human factors analysis (see Section 4.2.8.3), and ONR's human factors inspector has raised concerns about the substantiation of the data used as a basis for the human factors analysis. ONR's human factors inspector considers that the initiating event frequency used in the PSA is likely to have been derived on a conservative basis. Further substantiation, based upon detailed design and site specific operating procedures, is expected in the site specific phase.

465. In addition, the fuel route PSA model considers a conservative number of cask lifts. Hitachi-GE provided a sensitivity study showing that the impact of reducing the number of casks lifts to a 'best estimate' value was significant to the release frequency for cask drop faults outside the SFP. The contribution of cask drops to the total risk profile is small with the main impacts being cask drop resulting in a SFP leak (discussed above) and cask drops away from the SFP, which are limited to lower dose band releases.

466. The success criteria and accident sequence analysis is well described for the SFP PSA and fuel route PSA, allowing the PSA models to be understood and interrogated. Following detailed review of the PSA models and documentation a number of shortfalls were identified and are detailed in the following paragraphs.

467. A number of conservatisms were identified in the SFP PSA success criteria and accident sequence analysis:

- Many of the sequences in the SFP PSA have significant time available before fuel damage in which recovery actions could be considered. Without consideration of recovery actions the risk profile of the SFP PSA is not reflective of the plant and the importance of loss of cooling events is likely to be significantly inflated. A sensitivity study performed by Hitachi-GE shows that there is a significant reduction in fuel damage frequency (FDF) and large release frequency (LRF) when recovery actions are considered.
- The fuel damage criterion of 'fuel uncovery' is conservative. A sensitivity study performed by Hitachi-GE showed that if FLSS spray was claimed in the level 1 PSA to prevent fuel damage, there would be a significant reduction in FDF. It is acknowledged that the sensitivity study shows no impact on LRF, as FLSS spray is already considered in the SFP level 2 PSA to prevent a large release.

468. A number of optimisms or omissions were identified in the SFP PSA success criteria and accident sequence analysis:

- During POS E the inventory in the SFP is higher than in other POSs. The level 2 PSA for the SFP does not take this increased inventory into account when determining the source term and radiological consequences from a severe accident in the SFP during POS E. This gap is considered small as the radiological consequences for a severe accident in the SFP in any POS greatly exceed the SAPs Target 9 criterion (Ref. 1).

- The SFP PSA assumes that there will be no adverse effect on reactor building equipment following a severe accident in the SFP. This assumption is based upon there being a large ventilation route which will be opened to prevent hydrogen accumulation in severe accident sequences (Ref. 186). In the SFP PSA this ventilation route (via a blowout panel on the refuelling floor and a door on floor 1F) is assumed to be available, and no associated operator action or support systems are considered. Assessment of the UK ABWR hydrogen management strategy is discussed in the severe accident assessment report (Ref. 42). The SFP PSA should be aligned with this strategy, including probabilistically considering any claims being made on operators or support systems to prevent hydrogen accumulation in severe accident sequences.

469. Sufficient design information is not available for a number of the systems included in the scope of my assessment:

- The canister cooling system (CCS) and back-up CCS are defined at a high-level and represented by a single supercomponent in the PSA. The design of the CCS and back-up CCS is not complete and further design development is outside of the scope of GDA. These systems are expected to be simple, with clear and precise operating instructions and sufficient time available to respond to any relevant faults. The PSA should be updated when appropriate following design development of these systems beyond GDA.
- The design information on reactor building crane power supplies is not available within GDA. The PSA includes an assumption on the power supplies to the reactor building crane. This assumption should be reviewed when detailed design information is available beyond GDA.

470. My review identified that the fuel route PSA assumes a low reliability for the SGTS and R/A HVAC. The low reliability is caused by an assumption concerning the test interval for the digital C&I. Hitachi-GE performed a sensitivity study which showed a significant reduction in the release frequencies if the test interval was reduced from once per cycle to once per month. The schedule for maintenance and testing will need to be developed by the future licensee. The PSA should be used to risk inform the schedule for maintenance and testing and should be updated once the schedule is defined.

471. A sensitivity study performed by Hitachi-GE also shows that the large release frequency (LRF) from the SFP PSA has high sensitivity to the dependency modelled between operator actions in the level 1 SFP PSA and the level 2 SFP PSA. This highlights the importance of appropriately modelling the operator response to faults in the PSA. The assessment of the HRA in the PSA models, including dependency between level 1 and level 2 PSA operator actions, is discussed further in Section 4.2.8.

472. In addition to the shortfalls identified above, improvements could be made to the documentation to aid readability and usability, including:

- The link between the initiating events (IEs) defined in the initiating event analysis report and the IEs included in the event trees developed in the event sequence report is not clear.
- The system analysis for SFP PSA (Ref. 171) identifies the assumption that FPC requires support from a HVAC system, however this assumption is not recorded in the PSA assumptions list (Ref. 32). Substantiation of this assumption is also needed.
- The derivation of the IEF for heavy load drop has undergone extensive revision during GDA, resulting in the documentation being unclear.

- System and initiating event importance measures are not reported for the SFP level 2 PSA.
- The potential flooding consequences following SFP leak, SFP overfill or actuation of FLSS sprays is not well described.
- There is a potential uncharacterised conservatism in the fuel route PSA model concerning the application of an unmitigated source term (ie failure of R/A HVAC isolation) to sequences with successful R/A HVAC isolation, but subsequent failure of SGTS to operate.

This has been captured as a minor shortfall.

473. Simplified internal fire and flood analyses have been undertaken for the SFP. These analyses indicate that the risk arising from the impact of these hazards on the SFP is lower than the impact on the reactor at power; however the reactor at power analysis indicates that internal fire and flood are some of the dominant contributors to risk for the UK ABWR. In addition, the reactor at power analysis has been refined during GDA, to address conservatisms and design development; the SFP analysis has not been refined in a similar way. The SFP analysis should be extended as required to be consistent with the reactor at power PSA analyses and reflect the site specific design, operation and maintenance of the UK ABWR. My assessment of the internal fire and internal flooding PSAs can be found in Sections 4.2.11 and 4.2.12, respectively.

> ***Assessment Finding AF-UKABWR-PSA-007****: The licensee shall provide revised internal fire and internal flood PSAs for shutdown and spent fuel pool operations which are consistent in detail and scope to the at power analysis. The revised PSAs shall reflect the site specific design, operation and maintenance of the UK ABWR and take any relevant shortfalls identified by the GDA review into account.*

474. My evaluation of the findings in this particular area of the PSA has shown that the shortfalls identified in this area would not result in a significant impact on the UK ABWR risk profile.

4.2.16.4 CONCLUSION

475. Based on the outcome of this assessment, I have concluded that sufficient analysis has been performed to have a good understanding of the level of risk and the risk profile from the UK ABWR SFP and fuel route PSAs as part of the UK ABWR 'generic' PCSR.

476. Although a number of shortfalls have been identified, the risks associated with the spent fuel pool and fuel route are low, and the current estimation is likely to be conservative and should be further refined in the site specific stage. Therefore the shortfalls identified will have a limited impact on the overall risk profile of the UKABWR.

4.2.17 Level 1 PSA: Uncertainty Analyses, Quantification and Interpretation of the Level 1 PSA Results (A1-2.9)

4.2.17.1 ASSESSMENT

477. My Step 3 review (Ref. 77) identified that the sensitivity analyses performed by Hitachi-GE were insufficient to demonstrate that the modelling assumptions and uncertainties had minimal impact on the PSA conclusions. In addition, parametric uncertainty propagation analyses for the UK ABWR level 1 and level 2 PSA had not been undertaken. The PSA database identified some assumptions that had a significant impact on the UK ABWR PSA results. However, it was not clear how Hitachi-GE proposed to reduce these uncertainties.

478. The UK ABWR PSA documentation provided limited information regarding the UK ABWR PSA quantification and results interpretation. The following was not explicit in the documentation:

- There was a lack of clarity regarding the justification of the UK ABWR PSA truncation value.
- A detailed examination of individual cutsets including an assessment of a sample of individual cutset and their validity to adequately represent a path to core damage and to release.
- A summary of all individual accident sequence frequencies contribution to plant damage states (PDS), total CDF and to total LRF.
- The importance measures for basic events and systems relative to CDF and LRF.
- Initiator importance in terms of CDF and LRF.
- A description of the UK ABWR risk profile and insights based on the PSA results including identification of potential plant modifications, operator procedure modification, training requirements and other potential improvements that would help to reduce the UK ABWR risk ALARP.

479. The issues identified by my review were captured in RQ-ABWR-0560 which is a supporting reference to RI-ABWR-0002 (Ref. 3).

480. During Step 4 my team evaluated in detail Hitachi-GE's response to RQ-ABWR-0560, and an updated level 1 IEAP PSA was submitted to ONR at the end of Step 3 (and updated several times in Step 4). PSA sensitivity analyses were also submitted in Step 4 and updated in response to my review comments.

481. My assessment was performed using the latest available models and documentation provided by Hitachi-GE, which are described in Sections 2 and 3 of this report. My assessment was carried out against expectations in ONR's PSA TAG (Ref. 4).

4.2.17.2 STRENGTHS

482. Hitachi-GE has developed a single assumptions list (Ref. 32) to include all PSA assumptions.

483. The PSA documentation provides a discussion of some potentially important modelling uncertainties.

484. Sensitivity analyses have been developed to understand the significance of some of the potentially important uncertainties and assumptions.

485. In response to RO-ABWR-0076, Hitachi-GE has enhanced its approach to the use of the UK ABWR PSA to support the design development and the demonstration that the risk is ALARP. This has included a significant amount of work to review the UK ABWR PSA to identify insights to inform potential: design development and modifications, operator procedure modification, and other improvements that would help to reduce the UK ABWR risk ALARP. My review of this work is reported in Section 4.2.20.

4.2.17.3 FINDINGS

486. The updated PSA model (Ref. 12) has been quantified to convergence at low truncation values. Hitachi-GE has identified that the use of flag files complicates the system modelling and the existence of sequence markers generates non-minimal cutsets, which make the results slightly conservative. Additional model development is needed to ensure the model is easier and faster to quantify. The final results should be reviewed to identify any additional problems such as the impact of duplicate cutsets or non-minimal cutsets. This has been captured as a minor shortfall.

487. Ref. 5 provides uncertainty analysis for CDF and LRF. However, the monte-carlo generated mean has not been demonstrated to be consistent with point estimate results. Further investigation regarding the differences between the mean and the point estimate is needed, and the PSA model and documentation to be updated, as appropriate, to allow for the uncertainty analysis to be taken into account in any decisions made on the basis of PSA results; and provide confidence that the overall conclusions obtained from the PSA are valid.

> ***Assessment Finding AF-UKABWR-PSA-008**: The licensee shall review the uncertainty analysis for core damage frequency and large release frequency, to identify the cause for the significant difference in the monte carlo generated mean and the point estimate results and, if appropriate, the licensee shall put in place measures to resolve the cause of the significant difference.*

488. Ref. 5 provides a good starting point for a systematic approach to identifying modelling assumptions and uncertainties. This approach involves the review of the PSA assumptions list, Hitachi-GE peer review outstanding comments and regulatory review comments. The process should also include the review of international guidance such as Ref. 187 to extend the list of potential uncertainties that need consideration, such as ECCS strainer reliability data and severe accident phenomenology.

489. In addition, my review identified a number of specific shortfalls related to Hitachi-GE's sensitivity analysis that have been reported in previous sections. Additional comments are summarised in Ref. 90. These concerns point to a completeness issue in the identification of modelling assumptions and uncertainties. These concerns could result in different parametric and modelling uncertainties contributing to the overall uncertainty of the results of the level 1 PSA and the insights that can be obtained from the results.

490. Results are presented in the PSA summary report (Ref. 11). However, vulnerabilities associated with design or operation of the UK ABWR are not specifically identified within the level 1 PSA documentation. However, this is partly done in the work produced as part of response to RO-ABWR-0076 (see Section 4.2.20). This is captured as a minor shortfall.

491. My review identified that the definition applied to categorise a release as a 'large release' is not aligned with the potential of a release to lead to 100 fatalities used to provide an assessment of societal risk for SAP NT.1 Target 9 (Ref. 1). The large release definition should be reconsidered to determine if an enhanced definition could provide better alignment to the SAP NT.1 Target 9 criterion (Ref. 1), and therefore allow for a smoother interface with the level 3 PSA and for risk insights for the large release frequency (LRF) to be representative of risk insights for societal risk. This has been captured as a minor shortfall.

4.2.17.4 CONCLUSION

492. Based on the outcome of this assessment, I have concluded that the current model quantification, together with the presentation of results in Refs 11 and 35, are sufficient for the 'generic' PCSR.

493. However, these aspects of the PSA need improvements for further stages of the NPP development. The sensitivity analyses should be extended to consider a comprehensive list of uncertainties and address my review comments. Additional model development is needed to ensure the model is easier and faster to quantify. A proper presentation and discussion of the PSA results should be presented as part of the next update of the PSA.

4.2.18 Level 2 PSA (A1-3)

4.2.18.1 ASSESSMENT

494. A high-level review of the UK ABWR level 2 PSA against the expectations in ONR's PSA TAG (Ref. 4) (Table A1-3.1, 3.2, 3.3, 3.4, 3.5 and 3.6) was conducted during Step 3. This review raised concerns in the following areas (Ref. 77):

- The level 2 PSA documentation did not adequately explain the interface between level 1 and level 2 PSA models and how it would adequately ensure that dependencies are addressed.
- The PSA did not represent all accidents that end in core damage by a specific PDS.
- The proposed quantification process appeared to neglect radionuclide release characterisation dependencies from the level 1 PSA model, such as timing effects or certain functional dependencies.
- The proposed accident progression event trees were simplistic, did not address all aspects of the severe accident progression and dismissed, without robust justification, certain severe accident phenomena.
- The level 2 PSA for containment failure sequences and bypass sequences did not realistically evaluate the plant capability and resilience for severe accident sequences.
- The scope of the deterministic accident progression analyses that supported the level 2 PSA was limited to two postulated accident types with no clear justification.
- There was a lack of containment performance analyses for the UK ABWR. The containment failure envelope presented was not based on UK ABWR analyses and was based on relatively low pressure and temperature limits compared with other BWR containment failure curves. In addition, no criteria for failure were provided and no size or location of possible failures was discussed.
- The release categories developed for the level 2 PSA did not appear to be comprehensive or represent the spectrum of releases expected, such as from leakage through containment or containment bypass. The limited number of source term calculations and the lack of characterisation of certain accident progression sequences did not meet the regulatory expectations in ONR's PSA TAG (Ref. 4).
- The sensitivity analyses performed by Hitachi-GE were not sufficient to demonstrate that the modelling assumptions and uncertainties had minimal impact on the PSA conclusions. In addition, parametric uncertainty propagation analyses for the UK ABWR level 1 and level 2 PSA had not been undertaken. Some assumptions that had a significant impact on the UK ABWR PSA results were identified in the documentation. However, it was not clear how uncertainties could be reduced.
- The UK ABWR PSA documentation provided limited information regarding the UK ABWR PSA quantification and results interpretation.

495. From this review, two regulatory observations were raised. RO-ABWR-0046 (Ref. 46) covers the issues related to the containment performance analysis and RO-ABWR-0048 covers the issues related to the level 2 PSA (Ref. 47) identified by my review.

496. In response to these ROs, Hitachi-GE provided a revised level 2 PSA in January 2016 (Ref. 188) that was subsequently updated in June 2016 (Ref. 189) to address most of my Step 4 review comments, with a further documentation update performed in

April 2017 (Ref. 13). Hitachi-GE also submitted a containment performance analysis in July 2016 (Ref. 190), which was updated in January 2017 (Ref. 191) and June 2017 (Ref. 29).

497. The objective of my Step 4 assessment was to review, at a detailed level, the interface between the level 1 and the level 2 PSA (including the definition of PDSs and allocation of level 1 PSA sequences to PDSs), the level 2 PSA containment event trees (CETs), the source term and release category (RC) grouping structure, supporting MAAP analysis, containment structural analysis, detailed branch point quantification, model quantification, and the scope of the level 2 PSA. My review considered whether the expectations identified in RO-ABWR-0046 (Ref. 46) and RO-ABWR-0048 (Ref. 47) were addressed.

498. Key topics covered by my review were the following:

- Detailed review of the PDSs and the assignment of level 1 PSA sequences to PDSs. This included examination of the PSA models to review the treatment of dependencies across the level 1 and level 2 PSA interface, and review of the level 1 PSA end states including their treatment in the level 2 model.
- Assessment of the CET structure, including identification of function events and treatment of dependencies within the CET. A representative sample of the CETs was reviewed in detail.
- Review of the CET supporting analysis, including review of analysis results, the MAAP (Version 4) input deck parameter file used for the level 2 PSA supporting analysis (and level 1 PSA), and other results. In addition to the general review, my review team sampled some specific cases. My review team also checked whether the outcomes were reasonable compared with analyses performed for similar plants, including RPV breach time, containment pressurisation, drywell temperature and radionuclide release. This review also included the information provided by Hitachi-GE to investigate the potential impact of the most recent version of MAAP (Version 5) on the outcomes of the analysis (Ref. 87).
- The review of the completeness and adequacy of deterministic accident progression analysis. In response to my review, Hitachi-GE extended the deterministic analysis (Ref. 192, 193[4], 194) to consider additional accident mitigation and accident sequences and to provide a better understanding of the uncertainties of the severe accident phenomena. This included consideration of core melt progression and success of the RHR without venting, debris quenching effect on the containment response, success states for COPS with RHR and spray success, extended time of core melt progression with continuous water addition to containment from external water sources, molten core concrete interaction (MCCI) coolability, etc.
- Assessment of the adequacy of the source term and RC grouping structure. The specific attributes needed for a level 3 PSA were also assessed and their applicability to the UK ABWR was evaluated.
- Review of CET / PDS quantification and the impact of dependency modelling covering areas such as human dependencies and sequence timing dependencies.
- Examination of the phenomenological analyses, such as those used to assess the impacts of direct containment heating, steam explosions, and MCCIs. The uncertainty characterisation of severe accident phenomena within the level 2 PSA was also reviewed.

[4] *A further revision of this report was submitted late in GDA Step 4, which was not included in the PSA assessment.*

499. ONR's severe accidents team commissioned independent analyses using the MELCOR code for selected severe accident scenarios. The outcomes of these analyses identified a number of uncertainties as reported in Ref. 42. The scope of my review included the areas of uncertainty highlighted by these confirmatory analyses.

500. In parallel to the above activities, the containment structural analysis for the level 2 PSA was also reviewed in detail. This review has been carried out in two stages.

501. Initially, the containment performance analysis topic report (Ref. 195) was submitted in December 2015 in response to RO-ABWR-0046, and later updated in July 2016 (Ref. 190) and January 2017 (Ref. 191) in response to the findings of my review. The first stage of the review covered the following aspects:

- identification of failure modes included in the overpressure analysis and characterisation of failure modes in terms of the size and location of the failure;
- 'best estimate' failure pressure presented for each failure mode, including any degree of optimism or conservatism in the supporting analysis;
- modelling of penetrations and temperature effects, especially around seals and penetrations, and the analysis and justifications presented;
- adequacy of the structural model, criteria and the analysis for evaluating the liner integrity and structural capability of components when subjected to negative containment pressures;
- adequacy of the structural model, criteria and analysis for evaluating the maximum hydrodynamic loads that the containment can withstand ie containment walls, access tunnels, penetrations under various containment flood conditions; and
- identification of other conditions imposed on the containment that could lead to containment failure and assessment of the conditions identified.

502. In order to address my review comments, Hitachi-GE developed reinforced concrete containment vessel (RCCV) models specific to the UK ABWR to evaluate the containment performance (concrete structure and the liner). The outcome of this work is summarised in a revised containment performance analysis report submitted to ONR in June 2017 (Ref. 29).

503. The second stage of the review was led by the ONR civil engineering inspector. This considered whether these RCCV models represent the UK ABWR GDA design and the adequacy of the civil engineering parameters. ONR's structural integrity inspector reviewed the methods used in the analysis of the RCCV liner. The details of this review are reported in Ref. 89. ONR's civil engineering inspector, in conjunction with ONR's severe accident inspector and my review team, considered whether the outcomes of the RCCV models were adequately reflected in the June 2017 containment performance analysis report (Ref. 29).

504. My assessment was performed using the latest available models and documentation provided by Hitachi-GE, which are described in Sections 2 and 3 of this report. My assessment was carried out against expectations in ONR's PSA TAG (Ref. 4). The findings of the Step 4 review of the level 2 PSA are presented in Section 4.2.18.3.

4.2.18.2 STRENGTHS

505. The updated level 2 PSA methodology is generally appropriate and its implementation reflects the approaches outlined in the PSA documentation. A number of changes from previous submissions have been adopted to reduce the conservatisms, eliminate shortfalls, or highlight the key assumptions. These changes enable the level 2 PSA to generally meet the expectations in ONR's PSA TAG (Ref. 4) and to provide a

framework that has the capability to reflect a realistic risk profile for the UK ABWR, subject to resolution of a number of identified shortfalls. Specifically the following strengths are highlighted (noting the shortfalls summarised in the sections below):

- The entirety of the level 1 PSA is taken forward to the level 2 analysis, including internal initiating events at power and shutdown for the reactor and SFP, internal and external hazards, and the fuel route.
- Overall the analysis of the interface between level 1 and level 2 PSA has systematically addressed the attributes of the level 1 core damage sequences that can affect the accident progression. Dependencies between level 1 core damage sequences and the level 2 PSA model are adequately represented.
- The PDSs are generally a good representation of the possible states of the plant following core damage and the CETs are a good representation of the possible accident progression.
- The source term and RC grouping was also considered adequate.

506. Hitachi-GE has also developed a number of sensitivity analyses (Ref. 5). These, in combination with Ref. 192, explore the uncertainty associated with severe accident phenomena and PSA assumptions. These analyses also considered some of the MAAP assumptions and optimisms.

507. In the review of the UK ABWR containment structural analysis for the level 2 PSA the following strengths are highlighted:

- The loads and combinations of loads studied are clear, including temperature effects.
- In response to my review comments, Hitachi-GE developed RCCV models specific to the UK ABWR using more realistic material properties and performed a wide range of sensitivity studies (Ref. 29). Although the analysis was completed too late to be reflected in the UK ABWR PSA, they demonstrate the assumptions in the PSA are conservative.

4.2.18.3 FINDINGS

4.2.18.3.1 Level 2 PSA: Interface Between Level 1 and Level 2 PSA (A1-3.1)

508. The detailed technical review that supports the findings discussed in the following paragraphs is documented in Ref. 91.

509. The review identified a limitation in the justification of the event trees success sequences end point (24 hours). Hitachi-GE's justification relies on Ref. 105, which presents a summary of the plant capacity in terms of water, DC power and pneumatic supply. This reference indicates that long timescales are available to allow for long term measures to be put in place, if needed, to ensure that the reactor reaches a safe, stable shutdown state. As mentioned previously, there is a lack of clarity regarding whether scenarios involving containment failure which occur after 24 hours have been adequately considered in the PSA. However, my review has not identified any specific sequences that could lead to containment failure, which are not included in the PSA. In addition, Hitachi-GE identified, in response to RQ-ABWR-1286, that analysis of the mitigated severe accident sequences was performed for 72 hours after the initiating event (Ref. 193) and that it was confirmed for all cases that containment pressure and temperature do not increase near the end point. This provides me with confidence that any relevant sequences are already captured or that the contribution of any missing sequences to the large release frequency (LRF) is small. However, a systematic demonstration has not been provided.

510. The outcomes of the review of the human failure event (HFE) dependencies between the level 1 and level 2 PSA is reported in Section 4.2.8.

4.2.18.3.2 Level 2 PSA: Deterministic Accident Progression Analysis (A1-3.2)

511. The detailed technical review that supports the findings discussed in the following paragraphs is documented in Ref. 91.

512. My review has identified some severe accident phenomena for which there is a lack of clarity and justification regarding their consideration or omission in the PSA. A summary of key shortfalls is provided below (paragraphs 513 to 520).

513. There is limited discussion of reactivity control in the level 2 PSA documentation. Hitachi-GE has clarified that information is available in Ref. 193. This information has been reviewed by ONR's severe accidents inspector and the review outcomes are reported in Ref. 42. The PSA model and documentation should be updated to reflect these scenarios that have been analysed deterministically and reflect the resolution of any shortfalls identified in Ref. 42. This has been captured as a minor shortfall.

514. Hydrogen combustion in the reactor building (see Sections 4.2.16 and 4.2.15) is not explicitly modelled in the PSA. Hitachi-GE has defined a hydrogen management strategy that relies on the operation of: passive autocatalytic recombiners, SGTS, and the reactor building ventilation system (RBVS) for severe accidents on a reactor at power. For severe accidents on a shutdown reactor or the SFP the hydrogen management strategy relies on operation of reactor building blowout panels. These measures are not explicitly included in the PSA. It should also be noted that the use of the containment vent could result in hydrogen entering the vent pathways which could lead to additional severe accident sequences following its closure. The UK ABWR also has specific design features to address this (eg alternative nitrogen injection). However, the reliability of these systems is not quantitatively addressed in the level 2 PSA as a safe and stable state is considered to be reached without the need to close the containment vent. ONR's severe accidents inspector has reviewed the adequacy of the deterministic safety case for the management of hydrogen (Ref. 42) and considered that the measures in place to deal with the generation of hydrogen are adequate. Taking into account that the frequency of core damage of the UK ABWR is low and the presence of dedicated systems to deal with the generation of hydrogen, which are expected to be reliable and are considered adequate from a deterministic point of view, I expect that these scenarios will not be significant contributors to the large release frequency (LRF) when modelled in the PSA. Nevertheless, to ensure the PSA is complete and adequate for future applications, the scenarios identified above and the associated safety systems should be included in the PSA model.

515. The PSA model includes a single basic event which appears to represent all combustion-related failure modes in the containment during de-inerted conditions. Currently, the PSA assumes a de-inerted condition once per cycle. Hitachi-GE undertook a sensitivity analysis considering the de-inerted period was three times longer; this analysis showed a small increase in LRF. In the future, the basis and derivation of this basic event and the associated probability should be documented. It is also considered that the PSA should be updated to account for multiple forced or planned outages during a fuel cycle that involve power operation (coast down, start-up) with a de-inerted containment.

516. The analysis performed by Hitachi-GE involved an assessment of the heat load on the lower drywell (LDW) assuming that core material is discharged from the RPV and some fraction is retained on the structures beneath the vessel. This analysis was dependent on the assumption that the core discharge from the RPV occurs in the most central region of the lower head. However, in reality, all of the CRD penetrations are exposed to approximately the same thermal challenge and any or all could fail. Recent

observations from Fukushima Dai-ichi Unit 2 reveal that the release of material was not from the centre. Upon request, this assumption was re-evaluated and additional arguments provided by Hitachi-GE in response to RQ-ABWR-1364. The analysis and arguments provided were assessed by my review team and were considered to be adequate to justify the approach adopted in the PSA. However, the PSA documentation should be updated to reflect the response to RQ-ABWR-1364. This has been captured as a minor shortfall.

517. Bypass of the suppression pool (S/P) due to vacuum breakers (V/Bs) failed open or other structural failures of the wetwell to drywell interface have not been considered in the PSA. For example, severe accidents that result in drywell spray actuation will result in differential pressure cycles that in turn cause wetwell to drywell V/Bs to cycle. This cycling may lead to stuck open V/Bs. Upon request, Hitachi-GE provided a sensitivity study (Ref. 5) that shows a small risk impact. However, to ensure that future revisions of the PSA are complete and adequate for future applications, these scenarios should be explicitly modelled.

518. The PSA assumes that PDS AC (LOCA sequences with failure of reactivity control) results in containment failure due to overpressure. However, this analysis does not take into account the potential containment conditions specific to PDS AC, such as low containment temperature, high containment water level, high containment pressure, and ‘chugging’ discharge from the RPV that could result in more severe challenges. For example, Hitachi-GE identified that a primary system pipework break due to high RPV pressure may result in RCCV boundary failure due to hydrodynamic loads, which then may result in a more severe consequence than overpressure. As identified previously, the modelling of the ATWS scenarios (and therefore PDS AC) is potentially conservative; there are also conservatisms in the containment performance analyses (see Section 4.2.18.3.3). Taking this into account, it is my opinion that the consideration of other containment conditions that could lead to more severe scenarios than containment overpressure failure will not result in an overall increase of LRF. However, further analyses to investigate these conditions beyond GDA are needed to ensure the completeness of the PSA.

519. Creep rupture of the main steam line (MSL) was identified as a result of analysis carried out with MELCOR during the state of the art reactor consequence analysis (SOARCA) project (Ref. 196). MAAP does not predict high enough temperatures in the upper head of the RPV to result in MSL creep failure. As part of the consideration of uncertainties related to the severe accidents, additional sensitivity analyses should be performed as part of the level 2 PSA to investigate cases where MSL failure occurs at some point following the onset of core damage, but before vessel breach. The containment pressure and temperature response to MSL creep rupture should also be considered.

520. The outcome of the MELCOR independent confirmatory analysis commissioned by ONR’s severe accidents inspector has identified a number of areas of uncertainty related to the modelling of key severe accident phenomena in both MELCOR and MAAP (such as, zirconium-steam oxidation, candling and blockage models during core degradation, RPV failure mode and MCCI). These are reported in Ref. 42 and are in line with the outcomes of my review. ONR’s severe accidents inspector has judged that the conclusions of the analyses performed by Hitachi-GE remains valid considering the degree of uncertainty and lack of knowledge associated with these phenomena (Ref. 42). Follow-up is needed beyond GDA to capture new insights and learning from the Fukushima Dai-ichi accident. These should be taken into account in future updates of the PSA as part of normal business.

521. In general, the accident progression analyses have been performed on a ‘best estimate’ basis and are specific to the UK ABWR. However, my review has identified

some areas where there may be excess conservatism or optimism in the accident representations. In particular:

- The PSA assumes that FLSS injection at any point prior to core plate failure is sufficient for achieving in-vessel melt coolability and will not result in RPV vessel breach, and that injection following core support failure will not achieve in-vessel melt coolability and will result in RPV failure. There is a lack of justification provided for this assumption, specifically for injection just prior to core plate failure. Hitachi-GE has provided some MAAP analysis (Ref. 193) to show that in-vessel melt coolability would be successful at low RPV pressure following core support failure; however it should be noted that there are large uncertainties in the analysis which have not been examined by ONR. Given the uncertainties associated with recovery of a damaged core, the PSA should be updated and include a success probability for in-vessel melt coolability, supported by further analysis as necessary. Hitachi-GE has performed a sensitivity analysis to evaluate the impact on the large release frequency if timely in-vessel core cooling recovery is not credited which shows a moderate increase in the LRF of 10%. The sensitivity study identified that this increase would be reduced if alternative means of water injection such as control rod drives (CRD) were credited. Further analyses are needed beyond GDA to better inform a potential strategy for in-vessel melt coolability and ensure the PSA adequately reflects the associated uncertainty.
- High pressure injection is not claimed in the level 2 PSA to mitigate the accident in-vessel or ex-vessel. A claim on high pressure injection could be possible following level 1 PSA sequences that were not able to claim HPCF (eg SBO events where AC power is restored in level 2 PSA, RPV rupture where no HPCF credit is taken, etc). Conversely, the adverse impact of the additional hydrodynamic load on containment from an initial HPCF injection has not been analysed. HPCF failure after an initial injection of a large amount of water to the containment may result in an additional hydrodynamic load not considered in the containment analyses (see Section 4.2.18.3.3). HPCF should be explicitly included in the IEAP level 2 PSA to assess both the benefits and possible adverse impacts.
- The level 2 PSA does not include consideration of the positive or negative effects of venting from the drywell. The drywell vent could be claimed following wetwell vent failure to prevent containment failure, and conversely operator error which results in alignment of the drywell vent instead of the wetwell vent could result in an increased source term. In addition, the risk significance of the filter could be underestimated. Hitachi-GE characterised the impact on the risk profile in response to RQ-ABWR-1316, which showed a small impact. In addition, claims on PCV venting in high drywell temperature conditions could result in a small decrease in LRF. Overall, I judge that the PSA presents a conservative result and an extension of the analysis should be undertaken to provide a ‘best estimate’ characterisation of the UK ABWR risk profile. This may have an impact on the importance and modelling of molten core concrete interaction (MCCI) and operator action to control PCV water level. Results of the analysis should be considered during the development of the UK ABWR accident management strategy.
- My review team questioned whether the modelling of suppression pool (S/P) bypass in the PSA was unduly conservative as analyses were based on the simultaneous failure of all 16 SRV tailpipes in the wetwell airspace and with no mitigation measures that are effective in preventing containment failure. Hitachi-GE investigated whether RPV emergency depressurisation given one SRV tailpipe break in the wetwell air space would cause containment overpressure. The results of this study (considered adequate by my review team) show that the conservatism in the current PSA is small. The future

development of the site specific PSA should consider whether the PSA model needs refinement to adequately represent these scenarios.

4.2.18.3.3 Level 2 PSA: Containment Performance Analysis (A1-3.3)

522. The detailed technical review that supports the findings discussed in the following paragraphs is documented in Ref. 197.

523. My review of the July 2016 (Ref. 190) and January 2017 (Ref. 191) containment performance analysis report found that the analysis presented was not fully representative of the UK ABWR. In particular, the failure criteria adopted to construct the containment limiting pressure-temperature curves presented for the UK reinforced concrete containment vessel's metallic components made use of data from Mark-II BWR. Comparison between the two containment designs highlighted large differences and therefore I requested Hitachi-GE to provide further justification in RQ-ABWR-1235.

524. In addition, the analysis presented to evaluate the containment's performance was not 'best estimate' and included different degrees of conservatism normally included for substantiating the design integrity.

525. In response to RQ-ABWR-1235, Hitachi-GE provided a review of the containment capacity at the majority of the failure locations using UK ABWR design information and more realistic information. Subsequently, Hitachi-GE developed RCCV models (concrete structure and liner) to evaluate the containment performance for the UK ABWR using more realistic material properties and provided an updated containment performance analysis report (Ref. 29).

526. Hitachi-GE also provided an evaluation of the impact of potential differences in failure location and size (eg failure of the drywell head flange at a higher pressure which could result in a larger leak size) on the source term. The conclusion of these studies show that the current LRF is bounding of the impact of a delayed containment breach combined with a larger failure area. These analyses were assessed by my review team and considered an adequate treatment for sensitivity to containment failure location and size.

527. My review of the above documents has identified that the following shortfalls are still outstanding:

- The reinforced concrete containment vessel (RCCV) models are based on more realistic assumptions regarding material properties for the UK ABWR RCCV. These assumptions are judged reasonable by my review team, but further work will be required to confirm the values for concrete and steel reinforcement are 'best estimate' when RCCV detailed design information is available beyond GDA.
- The RCCV models will also need to be reviewed once the RCCV detailed design information is available to confirm they reflect the UK ABWR detailed design.
- The review noted that the RCCV models did not address thermal deformations for regions of the RCCV where the steel liner interfaces with the concrete containment, which could be prone to liner tearing during cooling following an accident. Hitachi-GE's current analysis for liner tearing is described in the response to RQ-ABWR-1235. Once information on the detailed design of the liner and operating procedures is available further analysis should be provided.
- The RCCV models did not include the drywell head. However, the results from the analysis undertaken have provided me with confidence that the initial failure mode of the containment will occur at the drywell head (and therefore the current PSA model is conservative).

- Leakage at the drywell head flange is assumed to be the limiting failure mode and therefore provides the basis for the pressure-temperature envelope for the UK ABWR containment. The PSA is currently based upon pressure-temperature limits for the drywell head that are not a 'best estimate' evaluation and the flange opening criteria used is not in alignment with the results of testing data presented by Hitachi-GE. A 'best estimate' pressure-temperature envelope to understand the ultimate performance of the drywell head should be developed and reflected in the PSA.

528. The Hitachi-GE analyses also considered hydrodynamic loading, assuming a maximum water level of 17.15 m. The 17.15 m height is based on the elevation of the wetwell vent. However, it is not clear whether there are any possible scenarios in which the water level could exceed that level, either due to emergency procedure requirements or due to operator error. Hitachi-GE argued that the water level will always be maintained below the wetwell vent line; and for cases where FLSS or FLSR are used for long term injection from an outside source of water containment would be depressurised and even with a higher water level, hydrodynamic loading would not be severe. Hitachi-GE also stated that, in the unlikely event that the water level exceeds the wetwell vent, drywell venting is possible. My review team agrees that cases with the vessel at high pressure and water levels above the assumed maximum value of 17.15 m would be of low likelihood (and therefore these scenarios will not result in a noticeable increase of the large release frequency). However, it is still not clear why these scenarios are not explicitly considered in the PSA, including credit for drywell venting. Containment flooding, albeit unlikely, is a strategy developed within BWR-specific severe accident guidelines. Also, RPV flooding is an EOP strategy which, by its nature, would involve containment water level above 17.15 m. UK ABWR SAMGs will be developed beyond GDA; when further information becomes available and these scenarios should be considered in the PSA.

529. Hitachi-GE provided a demonstration that negative containment pressure would not represent a challenge for the UK ABWR containment. However, the list of scenarios considered did not include the failure to open of the vacuum breakers in case of drywell spray. A sensitivity analysis was provided by Hitachi-GE to evaluate the risk impact of failure to open of the vacuum breakers. Hitachi-GE identified that in the case of a LOCA, if the vacuum breakers do not open after the initial blowdown, a large pressure difference could be created between the upper drywell and wetwell. Hitachi-GE conservatively assigned these failure cases to large release. Due to the low likelihood of a vacuum breaker failing to open, the overall contribution to the large release frequency is small. In the future development of the PSA, this failure mode should be considered in the model and supported by the required analysis.

530. The review also identified that there was a lack of substantiation regarding the following level 2 PSA assumptions (additional details are summarised in Refs 42 and 89):

- The integrity of the V/Bs during low pressure severe accident sequences due to potential high temperatures prior to or during RPV failure (Ref. 42).
- The load-bearing capacity of the pedestal wall when impacted by corium. It is also considered that Hitachi-GE do not adequately consider the vent pipes within the pedestal wall.

I consider that the assumptions adopted in the analyses are reasonable, but justification is needed once detailed design information becomes available beyond GDA

531. The outcomes of my review of the containment performance analyses indicate that the current PSA is conservative regarding the consideration of the failure of the containment. Further work is needed beyond GDA to ensure the analyses are 'best estimate'. My review has also identified a limited number of shortfalls which could have

an impact on the risk profile. On the basis of the sensitivity analyses provided by Hitachi-GE and my evaluation of the PSA results (as explained in previous paragraphs), I do not consider the shortfalls will have a significant impact on the large release frequency.

4.2.18.3.4 Level 2 PSA: Probabilistic Modelling Framework – Accident Progression Event Trees (A1-3.4)

532. The detailed technical review that supports the findings discussed in the following paragraphs is documented in Ref 91.

533. The review identified that some SSCs and failure modes are omitted from the PSA without justification, which could affect the accident progression and limit potential future uses of the PSA including the risk insights. The key shortfalls identified by my review are listed below:

- Containment integrity has been identified in operating experience data to be challenged by latent failures that remain unrevealed without adequate testing. However, these are not considered in the PSA without adequate justification provided.
- In general, with a few specific exceptions, operator cues have not been explicitly modelled in the PSA. Hitachi-GE's response to RQ-ABWR-1263 provides information regarding the operator cues needed to support operator actions claimed in the PSA, including identification of related instrumentation and C&I platform. The information provided shows that where operator cues have not been included in the PSA there is more than one cue available to the operator. This gives me confidence that when the cues are included in the PSA, the impact on the risk profile will be small. To ensure that all relevant dependencies are captured, operator cues should be explicitly modelled in the PSA following GDA.
- The PSA does not explicitly address the loss of level indication. Hitachi-GE has indicated that, given loss of RPV level instrumentation, operators will align RPV flooding. However, for scenarios following core damage, there is a lack of clarity of specific actions as to how the operators will align RPV flooding and if there are any key dependencies e.g. power supplies.
- Furthermore, there is a lack of justification that injection at flow rates equivalent to compensate for decay heat are sufficient when exothermic reactions due to steam-zirconium reactions and significant stored heat within the debris may be present in the reactor. In addition, as identified in Section 4.2.18.3.3, it is not clear whether there are limitations on containment water level conditions (that could challenge the containment integrity) that would force termination of external water injection and how this is incorporated into the PSA.
- The suppression pool pH control system is not explicitly modelled in the PSA. Failure of the suppression pool pH control may have an impact on the source term calculated for various release categories. Success criteria for this system should be developed, included in the PSA in a probabilistic manner and the release categories updated as necessary to reflect potential failure of the pH control system. Hitachi-GE confirmed (Ref. 137) that consideration of pH control would have an impact on the PSA risk profile. Following my review of the information provided in conjunction with ONR's level 3 PSA inspector, it is not clear if the analysis presented in Ref. 137 correctly characterises the risk impact. However, any safety system is expected to be reliable and any sequences following failure of the system are expected to be low frequency, therefore the impact on the risk profile is expected to be small.

- Hitachi-GE has confirmed that there is a range of elevated containment pressures that can lead to SRV re-closure unless containment venting is initiated or HPIN is used. For lower containment pressure scenarios, the ADS and RDCF accumulators are sufficiently sized to ensure the SRVs remain open (also discussed in Section 4.2.7). Currently these scenarios and support systems are not modelled in the PSA and should be considered in the future. Hitachi-GE performed a sensitivity study (Ref. 5) which showed that the large release frequency (LRF) increase is negligible if long term nitrogen makeup is included in the model. It should be noted that this study relies upon the PSA model being updated to reflect the final GDA design of the nitrogen accumulators (Ref. 198).

534. Additional comments regarding potential missing human failure events (HFEs) in the PSA are provided in Section 4.2.8.

535. It is important to note that, during GDA, the UK ABWR's severe accident management guidelines (SAMGs) were not available. The SAMGs should feed through into a future update of the PSA. It is a regulatory expectation that the link between the CETs and the SAMGs is transparent and explicitly documented. Further comments are provided in Section 4.2.18.3.6.

536. My review has identified, similar to the shortfalls raised on the level 1 PSA system analysis, limitations in how the level 2 PSA considered system operation under degraded conditions with respect to:

- adverse environment;
- system limitations, interlocks or trips;
- operator manipulation success when high radiation may be present (in containment or on-site); and
- actions that would be taken given degraded plant indication systems, such as unavailable or unreliable containment pressure, temperature, or level indications.

In particular, environmental conditions related to core damage progression, containment leakage or containment failure are not always included in the PSA. Furthermore, although the PSA documentation has been improved in response to my review queries and provides further information on assumptions related to the systems survivability and effectiveness during severe accident progression, these assumptions will need to be demonstrated beyond GDA. This should be performed when information regarding the detailed design will be available. Some examples of environmental conditions which are not considered are identified in Section 4.2.18.3.3.

537. In addition, ONR's severe accidents inspector identified that there is a lack of substantiation regarding the assumption that the SRVs will remain open during the core degradation phase, given the high heat loads expected (this was also identified by Hitachi-GE in response to RQ-ABWR-1299), and that the SRV tailpipes remain intact. These shortfalls have been captured in ONR's severe accidents assessment report (Ref. 42); any changes in response to this shortfall should be reflected in the PSA.

4.2.18.3.5 Level 2 PSA: Source Term Analysis (A1-3.5)

538. The detailed technical review that supports the findings discussed in the following paragraphs is documented in Ref 91.

539. My review considered the containment performance analysis relative to its impact on the level 2 PSA and identified the following limitations:

- Hitachi-GE has undertaken a new containment performance analysis representative of the UK ABWR late in GDA. The level 2 PSA will need to be updated to reflect the outcomes of this analysis. The current PSA is conservative in the modelling of containment failure and does not reflect all the potential releases pathways and sizes. A probabilistic treatment of containment failure is likely to reduce overall conservatism in the PSA.
- The PSA was updated during GDA to consider direct debris interaction (DDI) following RPV high pressure failure. The update had an impact on the containment failure frequency but not on LRF, due to the source term not meeting the LRF criterion of a caesium iodide (CsI) release fraction greater than 10%. Additional information was requested to justify that these sequences do not constitute 'large' magnitude releases. In response to RQ-ABWR-1362, Hitachi-GE provided additional details on the source term. A sensitivity analysis was performed in which the hatch seal failure area was increased by a factor of ten and the results show that the CsI release fractions increased but remained below the LRF threshold. In addition, plots of the CsI mass distribution prior to PCV failure were provided along with additional justification for the failure area. The information supplied should be reflected in the PSA documentation. This has been captured as a minor shortfall.

540. The PSA does not consider retention of radioactive releases in the reactor building (R/B) because the blowout panel is assumed to open following containment failure. Further discussion of potential improvements in this area is captured in the severe accident assessment report (Ref. 42). The PSA documentation should provide sufficient discussion of potential decontamination factors and how they could impact the release category representative sequences and source terms. Any insights regarding plant or procedural improvements that could be made to improve plant capability for fission product retention should also be identified. Given that the building is expected to be a single volume with release through the blowout panel, the impact on removing fission products is expected to be small. This has been captured as a minor shortfall.

541. ONR's severe accidents inspector has identified potential optimisms in the calculation of the source term for non-LOCA accident sequences with containment failure prior to RPV failure. Further information is provided in Ref. 42; any changes in response to this shortfall should be reflected in the PSA.

4.2.18.3.6 Level 2 PSA: Presentation and Interpretation of the Level 2 PSA Results (A1-3.6.)

542. The detailed technical review that support the findings discussed in the following paragraphs is documented in Ref 91.

543. It should be noted that some of the methods and approaches adopted in the UK ABWR level 2 PSA may lead to a conservatively biased evaluation of the risk profile for internal events at power. During the site specific development of the PSA, the conservatisms should be reduced to make the assessment both more realistic and more useful to decision-makers. Furthermore, it is important to ensure that the PSA is 'best estimate' to support the evaluation of external event and hazard risk profiles.

544. The level 2 PSA documentation does not identify potential design vulnerabilities. ONR's expectation is that qualitative and quantitative insights, based on the PSA, are used in the development of the design and the demonstration that the risks are ALARP. It is acknowledged that this work has been done to some extent as part of Ref. 35 (see Section 4.2.20). In addition, Hitachi-GE has not provided a summary of individual sequence contribution to plant damage states (PDSs), release categories (RCs) and total LRF. This has been captured as a minor shortfall.

545. During the site specific stage the PSA is expected to be used to inform the development of procedures and SAMGs. My review has identified a number of shortfalls in this area that should be considered further beyond GDA:

- While it was shown with MAAP analysis, considered by Hitachi-GE to be bounding, that containment failure can be avoided by venting at twice the design pressure (2Pd), given the uncertainties in severe accident phenomena and in the containment performance analysis, the point at which containment is vented should continue to be investigated.
- As discussed in Section 4.2.18.3.2, consideration of PCV venting at high containment temperature may result in risk reduction.
- Insights from the DDI evaluation performed as part of the containment performance analysis should be used to understand the prioritisation of operator actions and aid in consideration of whether injecting water after vessel breach directly into the RPV should be a preferred to other injection routes. This could be done to provide cooling to core debris that may be frozen on the structural steel beneath the RPV lower head.
- As identified in Sections 4.2.5 and 4.2.6, a number of accident sequences with failure of the automatic ATWS logic are assumed to result in catastrophic over pressure failure of the containment. Further consideration for the potential for credible operator actions in ATWS sequences should be investigated.

546. I reviewed the release categories (RCs) which contribute to SAP NT.1 Target 9 (Ref. 1) and identified that a number of RCs contribute to SAP Target 9 (Ref. 1) which are not categorised as a 'large' release. This results in a large difference between the LRF and the total frequency being compared to SAP Target 9 (Ref. 1), as can be seen in Table 2. This difference could result in incomplete or incorrect PSA insights being identified on the basis of the LRF results. On this basis, the criteria for a large release should be reviewed to consider if it is appropriate. This is captured as a minor shortfall.

4.2.18.4 CONCLUSION

547. My evaluation of the shortfalls in this particular area on the basis of Hitachi-GE sensitivity analysis has shown that none of the shortfalls identified would lead to a significant increase in the risk results. This judgement is made on the basis that the core damage frequency (CDF) is low. Many shortfalls have been identified, however the level 2 analysis is large and ONR review has been extensive. The number of shortfalls identified does not necessarily compromise the integrity of the analysis or results. Sensitivity analyses have been performed on many of the shortfalls, which showed that the shortfalls analysed generally have a small impact on the risk profile.

548. However, I note that a number of shortfalls are related to the need to provide further justification and analysis to underpin the assumptions in the PSA (eg survivability and operation of systems and modelling of operator actions) once detailed design information and SAMGs becomes available. If these assumptions are not supported, the PSA will require update and there may be a more significant risk impact.

549. On the basis of the assessment of the level 2 PSA described above I concluded that Hitachi-GE's level 2 PSA is sufficient to close RO-ABWR-0048 (Ref. 199) and for the 'generic' PCSR. However, improvements to support further stages of the NPP development are required to extend the consideration of severe accident phenomena, reduce uncertainty and conservatisms, reflect the UK ABWR detailed design (including demonstration of the current PSA assumptions) and SAMGs when available, and to reflect the results of the containment performance analysis.

550. Hitachi-GE's updated containment structural analysis for the level 2 PSA (Ref. 29) is generally reasonable and is sufficient to close RO-ABWR-0046 (Ref. 200), but will

require revision to reflect the UK ABWR detailed design and to confirm or revise assumptions regarding 'best estimate' material properties for the UK ABWR RCCV. In particular, a 'best estimate' pressure-temperature envelope for drywell head should be developed and reflected in the PSA. This information is important for the characterisation of the containment success criteria and fission product release that are key elements to ensure the accurate representation of the UK ABWR risk profile in the PSA.

4.2.19 Level 3 PSA (A1-4)

4.2.19.1 ASSESSMENT

551. In Step 3, my assessment (Ref. 77) considered the methodology and capability of Hitachi-GE to carry out level 3 PSA. Hitachi-GE also provided non-UK ABWR input data to proof test the level 3 PSA methodology. My review raised four RQs for resolution in Step 4 (RQ-ABWR-0547, RQ-ABWR-0548, RQ-ABWR-0550 and RQ-ABWR-0549), which included the following issues:

- Records for verification and validation (V&V) of the computer codes used to develop the consequence analysis for the level 3 PSA and ONR SAPs numerical targets 7 and 8 needed to be provided (PUMA used for Target 8 off-site dose calculations).
- I identified the need for a benchmark of the level 3 PSA analysis against Fukushima off-site measurements.
- It was not clear whether some of the assumptions made in the consequence analysis were consistent with other non-PSA radiological analysis.
- Further clarification regarding the use of countermeasures in the analysis was needed.
- Sequence end states not resulting in core damage were not clearly identified and defined in the PSA documentation, including consideration of the overall contribution to the PSA results and how they compare against ONR SAPs numerical targets such as Target 8.

552. My Step 4 review considered the level 3 PSA documentation submitted by Hitachi-GE (Refs 30, 31, 201, 202, 203 and 204) during Step 4. This review also considered the shortfalls identified in Step 3 as follows:

- As the stand-alone PUMA code, used for Target 8 off-site dose calculations, is solely owned by Hitachi-GE's TSC, I requested a demonstration of the V&V of this code (RQ-ABWR-0548 and RQ-ABWR-0709). This was fulfilled by an inspection at Hitachi-GE's TSC's offices (Ref. 79) where the internal software quality plan and the V&V work performed, after modifications to include various changes including ingestion dose, were presented.
- To gain further confidence in the results of PUMA, I performed confirmatory benchmark calculations of level 3 PSA deterministic results (Target 8) and non-reactor fault dose calculations (Ref. 205).
- I requested Hitachi-GE to review the Fukushima accident off-site consequences data to benchmark the codes PUMA and PC COSYMA which were used in the UK ABWR level 3 PSA (RQ-ABWR-0547 and RQ-ABWR-0640) and performed an inspection of this review at Hitachi-GE's TSC offices (Ref. 79).
- I requested clarity on counter measure assumptions used by Hitachi-GE in the level 3 PSA (RQ-ABWR-0549) and compared them with ONR's guidance (Ref. 40).

- I raised a number of queries (RQ-ABWR-0550) regarding consistency across the radiological analyses. To assess this, I reviewed non-PSA radiological consequence Hitachi-GE submissions (Refs 84, 206[5], 207[5] and 208).

553. In addition to the above I undertook a detailed review of level 3 PSA. My assessment plan (Ref. 209) included a review of internal events at power, with and without core damage, shutdown and spent fuel pool faults, hazards and non-reactor faults. My review did not require the commissioning of an external independent calculation to confirm the adequacy of the level 3 PSA as:

- I consider that the Hitachi-GE level 3 PSA team are highly experienced in UK off-site consequence assessment. This has been demonstrated in my interactions with them (Ref 79).
- Hitachi-GE's calculations for Targets 7 and 9 were performed using the computer code PC COSYMA. PC COSYMA is the principal modern level 3 PSA code in use in the UK.
- Target 8 calculations were performed using the stand-alone deterministic code PUMA. My assessment of this code and calculations in general are discussed above.
- For probabilistic assessment against Target 9 faults with core damage; where leakage is the release path, the consequences of the release is far below levels that would trigger consequences of the magnitude considered in Target 9; even with pessimistic assumptions. For containment failure or bypass faults, where core damage has occurred, the release consequences would be far above Target 9 criteria. The remaining faults (filtered venting via the wetwell after core damage and containment failure with perforated fuel) are less clear cut. My assessment of these faults is presented in Ref. 205.
- For probabilistic assessment against ONR SAP Target 7 (Ref. 1), the calculations performed for Target 8 (as discussed above) are sufficient to provide confidence in the results, with a generalised assumption of plume direction frequencies; further details of this assessment are provided in Ref. 205.

554. My Step 4 review also considered Refs 31, 203, 210 and 211[6] which were provided by Hitachi-GE to investigate the sensitivity of the consequence analyses to level 3 PSA inputs.

555. I also reviewed the adequacy of the definition of the source terms used in the level 3 PSA, including quantities of radionuclides, frequencies, timings and their consistency (including consistency with the outcomes of the level 2 PSA). My review of the methods used to model dispersion in the environment and calculate doses is presented in Ref. 205.

556. Hitachi-GE also submitted the 'Topic Report on Assessment of Non Reactor Faults and Reactor Lower Dose Sequences against Target 7 and Target 8' (Ref. 30). This topic report takes input from the IEAP, SFP and shutdown PSA event sequence analyses in addition to the fuel route PSA. Non-reactor faults are also considered, along with qualitative consideration of the contribution from hazards. My assessment of this submission focused upon:

- completeness of inputs;
- identification and application of bounding sequences; and

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[5] A further revision of this report was submitted late in GDA Step 4, which was not included in the PSA assessment.
[6] A further revisions of this report were submitted late in GDA Step 4, which was not included in the PSA assessment.

- calculation of off-site consequences.

557. To achieve this I performed a high-level review of the relevant documentation and sampled specific areas chosen for their potential importance to the analysis or because they are specific to the UK ABWR, including:

- leaks from the turbine system, including confirmation that it is bounded by main steam line (MSL) break;
- sequences where containment venting is claimed; and
- spent fuel cask drop.

558. I did not assess the qualitative consideration of the contribution from hazards, as the hazards PSAs require further development in the site specific phase (as discussed in Sections 4.2.10, 4.2.11, 4.2.12, 4.2.13, and 4.2.14) and are likely conservative at this time. However inclusion of hazards within the analysis submitted for GDA is considered a strength.

4.2.19.2 STRENGTHS

559. Hitachi-GE has developed a level 3 PSA that includes internal events at power, shutdown faults, spent fuel pool faults, success sequences, non-reactor faults and internal/external hazards. The approach used, including the use of countermeasures is consistent with most of the expectations in the PSA (Ref. 4) and radiological fault analysis (Ref. 40) TAGs.

560. My review confirmed that the source terms considered in the level 3 PSA were consistent with the release categories from the level 2 PSA. I also consider them to be clearly defined; including quantities of radionuclides, frequencies and timings (Refs 30 and 204).

561. In addition, Hitachi-GE also performed sensitivity analyses (Ref. 203) that, except for the shortfalls identified below, provide adequate justification for the parameters used in the level 3 PSA.

562. From the inspections I undertook in Step 4 (Ref 79), I am confident that a number of experienced individuals had been involved in the verification and validation (V&V) of the PUMA code. Furthermore, the results of the calculations discussed above confirm the results obtained by this code. During these inspections, I was also presented with information which demonstrated sufficient consideration of the post-Fukushima off-site measurements had been given to benchmark the codes being used (Ref. 79).

563. My confirmatory analysis on non-reactor fault dose calculations led to the identification of errors in the ground shine dose assessment, which I raised via an RQ (RQ-ABWR-1032). These were subsequently corrected and I found no further issues with the dose calculations for the PSA.

564. My review of the consequence analysis for ONR SAP Target 9 (Ref. 1) faults (in particular where filtered venting via the wetwell after core damage and containment failure with perforated fuel) found the results to be acceptable.

565. Hitachi-GE's assessment of lower dose band sequences against ONR SAPs numerical targets 7 and 8 (Ref. 1) considered contributions from a wide range of inputs, including direct inputs from the IEAP, SFP, shutdown and fuel route PSAs. A high-level analysis of the contribution of non-reactor faults and a qualitative consideration of the contribution from hazards was also included.

4.2.19.3 FINDINGS

4.2.19.3.1 General

566. The GDA level 3 PSA has been made as site specific as possible for the expected first site for a UK ABWR (Wylfa). For example, the population and local meteorological data are centred on Wylfa. However, certain parameters, such as stack height, will require update during the site specific stage, and it should be noted that other sites would require the GDA level 3 PSA to be further updated.

> ***Assessment Finding AF-UKABWR-PSA-009****: The licensee shall provide a revised level 3 PSA model and documentation, as part of the development of the site specific PSA, which takes into consideration the following:*
>
> - *Justification for the decontamination factors applied to the barriers to fission product release. This shall including those for the standby gas treatment system.*
> - *Updating the population data to reflect the most recent census, when reasonably practical to do so. This is needed to provide a more realistic assessment of dose uptake.*
> - *Consideration and justification for the expected increase in notional fatalities projected to the end of station life. The use of the most recent census data will assist this.*
> - *Model multiple release phases to more realistically model spent fuel pool fault sequences, or use and justify an alternate method for comparison against SAPs Target 7.*
> - *Revise the method for comparison to SAPs Target 9 to release frequency multiplied by conditional probability of exceeding 100 fatalities.*

4.2.19.3.2 Sensitivity Analysis

567. Hitachi-GE provided sensitivity analyses (Ref. 203) to justify the parameters used in the level 3 PSA. As noted previously, I consider that these analyses are overall sufficient for their purpose, with the exception of the shortfalls presented below.

568. Iodine speciation in faults is an important aspect of the source term for UK ABWR faults. This is because the behaviour of iodine in the suppression pool (S/P), containment atmosphere and filters depends on its chemical form. Therefore a relatively small fraction of iodine of a particular chemical form can, due to low decontamination factors (DFs) in these components, result in a large fraction of off-site dose. Sensitivity analyses considering different iodine speciation were performed and reported in Ref. 204. These analyses illustrate the relative insensitivity to organic iodine's DF in the filtered containment venting system (FCVS) in assessment against Target 9.

569. I consider that the elemental iodine deposition velocity used in Ref. 204 is optimistic because an urban lawn (as opposed to meadow grass) deposition velocity is used. This would underestimate ingestion dose (Ref. 205). I reviewed the sensitivity analyses provided in response to RQ-ABWR-0874 and RQ-ABWR-1012 (Ref. 210) and considered they adequately demonstrate the results are acceptable. As a result Ref. 31 does now use a reasonable deposition velocity.

570. I consider that the SGTS decontamination factor (DF) applied for elemental and organic iodine claimed in Ref. 204 is optimistic. This is because it does not take into account that the ageing of a filter prior to re-test will result in some reduction of performance from previous tests (RQ-ABWR-1013). I consider that Hitachi-GE's response to RQ-ABWR-1013 does not provide a complete substantiation of this DF. This is based upon Ref. 211, which is Hitachi-GE's topic report for chemical effects

during faults, where information about the DF after 100 days of operation is presented. Typically, the interval between filter tests is two years or more for UK NPPs. Furthermore, the generally accepted DF during tests exceeds the value used by Hitachi-GE, which does not account for any degradation in performance. However, I consider that this issue can be addressed beyond GDA as the level 3 PSA results are insensitive to this DF (Ref. 211).

4.2.19.3.3 Non Reactor Faults and Low Consequences Faults

571. The detailed technical reviews that support the findings discussed in the following paragraphs are documented in Ref. 212.

572. There are a number of shortfalls concerning the identification and characterisation of non-reactor faults:

- The link between the non-reactor faults considered in the PSA and the faults identified in the design basis analysis was not clear. Clarification was sought and provided in RQ-ABWR-1259, but the information has not been included in the PSA documentation.
- A number of faults identified by FMEA in the DBA topic report (Ref. 206) are screened out due to them being bounded by other initiating events (IEs). This approach is not directly applicable to the PSA, which should include the group of IEs represented by the relevant bounding accident sequences or individual faults. It should be confirmed that the groups of IEs in the PSA are adequate and the frequency of the individual faults is added to the IE group frequency as appropriate.
- Currently all faults on the turbine system have been bounded by the consequences for the main steam line (MSL) break fault. This fault has been used in the topic report on DBA to bound (screen out) a number of break sizes and locations, taking available protection into account. This approach is not adequate for the PSA. I consider that a range of sizes of breaks on the turbine system, in a range of locations, should be considered within the PSA. The PSA should also probabilistically consider the protection available for each fault.

573. Due to the low consequences expected for non-reactor faults, and the current margin to the basic safety objectives (BSOs) for Target 7 and Target 8 for non-reactor faults and lower dose band faults, the impact on the risk of the above shortfalls would be small. However, they should be addressed by a future licensee, to allow a comprehensive risk profile to be developed.

574. In addition, for faults involving the reactor, groups of sequences are represented by a single bounding sequence to calculate the radiological consequences. The approach applied should be justified and recorded. I challenged the lack of information in RQ-ABWR-1104, and in response information on a number of example sequence groups was provided, however this information was not added to the PSA documentation and not expanded to all sequence groups.

575. In general, the PSA documentation in this area needs to be updated to include the additional information provided in RQ responses, including RQ-ABWR-1104. This is captured as a minor shortfall.

4.2.19.3.4 Presentation and Interpretation of the Level 3 PSA Results

576. Hitachi-GE presented level 3 PSA results which incorporated internal events at power, shutdown faults, spent fuel pool faults, success sequences, non-reactor faults, internal hazards and external hazards. The results were compared to ONR SAPs numerical targets 7, 8 and 9. I found no major issues with the presentation and interpretation of the level 3 PSA results.

577. Hitachi-GE's interpretation of the Target 9 criteria is not aligned with ONR's expectation that all the sequences that could result in more than 100 fatalities should be considered to determine the Target 9 contribution (ie conditional probability multiplied by their fault frequency). This was raised in RQ-ABWR-1238. In response to this RQ Hitachi-GE was able to demonstrate that resolving this issue had a negligible impact on the overall result. The level 3 PSA final report (Ref. 31) included a footnote to clarify the discrepancies in the interpretation of Target 9 when compared with ONR's expectation to help resolve this issue.

578. The Hitachi-GE ALARP case did not take proportionate consideration of faults for which the consequences greatly exceed Target 9 in terms of the number of notional fatalities. This was raised in RQ-ABWR-1304 and is discussed further in Section 4.2.20.

579. The UK ABWR level 3 PSA results exceed the BSO for Target 9 and, to a lesser extent, Target 8 for doses greater than one Sievert. However, Hitachi-GE's initial ALARP case did not meet ONR's expectations. This was raised in RO-ABWR-0076 and is discussed further in Section 4.2.20.

4.2.19.4 CONCLUSION

580. On the basis of the assessment of the level 3 PSA and Target 7 and 8, described above, I conclude that Hitachi-GE's submission in respect of level 3 PSA provides sufficient evidence that the UK ABWR is capable of being constructed and operated in compliance with UK requirements and hence is adequate for GDA.

4.2.20 Use of PSA in ALARP Demonstration

4.2.20.1 ASSESSMENT

581. My review during Step 3 identified that Hitachi-GE's approach to use of the PSA (for example, use of the PSA to support the development of the design) did not meet expectations in ONR's PSA TAG (Ref. 4) Tables A1-2.9.3, A1-3.6, A1-6.2 and SAPs (Ref. 1) (ie FA.14). Furthermore, there was insufficient information and clarity regarding consideration of potential design enhancements to ensure the UK ABWR risk is ALARP.

582. Shortfalls identified by my review were summarised in RQ-ABWR-0560, which I identified as a reference of RI-ABWR-0002.

583. In September 2016, Hitachi-GE produced a topic report on use of PSA in ALARP assessment (Ref. 213). The aim of this report was to provide evidence that the PSA was used to support the demonstration that the UK ABWR risk is ALARP and to identify any areas where further risk reduction should be considered and investigated further either during GDA or during the site specific stage which will follow the completion of GDA.

584. The outcome of my review of this submission and the PSA results presented (which were above the BSO for ONR SAPs Target 9 (Ref. 1)) led me to raise RO-ABWR-0076, which identified ONR's expectations related to the use of PSA to support the demonstration that the risks for the UK ABWR are ALARP. In response, Hitachi-GE submitted further revisions of the 'Topic Report on Use of PSA in ALARP Assessment' (Refs 214 & 35).

585. One of the key expectations of RO-ABWR-0076 was for Hitachi-GE to develop and implement a process to review the PSA results and consider whether it would be reasonably practicable to implement further safety enhancements. In response to RO-ABWR-0076, Hitachi-GE has developed a process to review the PSA results and

to identify PSA insights to be considered by the engineering departments to develop options to reduce the risks ALARP. The Use of PSA in ALARP Topic Report was updated to revision 1 in response to RO-ABWR-0076 (Ref. 214), and contains details of the process developed.

586. My review of this process and its implementation considered the following:

- Hitachi-GE's approach to review the PSA results to identify vulnerabilities and areas where there are potential ALARP improvements which could be made. To support this, I compared the insights identified by Hitachi-GE against the outcome of my independent review of the PSA results.
- The completeness of the insights identified by Hitachi-GE's review of the PSA. This included how they were communicated within Hitachi-GE to the engineering teams to develop potential options for design or operational enhancement. The insights and recommendations of the engineering teams were recorded in what Hitachi-GE refers as 'Use of PSA Sheets'.
- The adequacy of the justification for: dismissal of PSA insights and design options or the justification for deferral of the assessment to beyond GDA.
- Through my interactions with other ONR specialist inspectors, I also considered how the PSA had been used to provide input to the ALARP assessment in other topic areas.

587. I raised RQ-ABWR-1399 to record the findings of my review and to request additional information from Hitachi-GE.

588. To follow-up the findings of my review, I conducted an inspection at Hitachi-GE offices (Ref. 215) to assess how the PSA team and engineering teams work together and how the PSA insights were used to inform the development of options to reduce the risks. Where development or implementation of options was identified as an activity for beyond GDA I reviewed a sample of the 'Use of PSA Sheets' to confirm the rational for the deferral was appropriate. Following my inspection, Hitachi-GE updated the Use of PSA in ALARP Topic Report to revision 2 (Ref. 35) to address feedback provided during the inspection.

589. During Step 4 GDA, Hitachi-GE refined the internal fire and internal flooding PSAs from those submitted to ONR earlier in Step 4 GDA. This internal hazards PSA refinement greatly reduced the contribution of internal fire and flood events to the risk profile of the UK ABWR, and included crediting in the PSA some of the ALARP options identified by the Hitachi-GE review of the PSA. My assessment of the internal fire and internal flooding PSAs, including the refinement, is covered in Sections 4.2.11 and 4.2.12 of this assessment report.

590. As mentioned previously, the results of the PSA for the UK ABWR indicate that the numerical comparison against ONR SAPs Target 9 is above the BSO (Ref. 1) and, to a lesser extent, the BSO for Target 8 for doses greater than one Sievert. Therefore my review of Hitachi-GE's ALARP submission in response to RO-ABWR-0076 has focused on core damage scenarios that could lead to large releases, on the basis that these scenarios contribute to SAP Target 9 (Ref. 1).

591. It is important to note that the demonstration that the UK ABWR risk is ALARP has been considered by ONR's GDA process across the totality of the safety case. My review has been focused on the PSA aspects. I have undertaken this review against ONR's SAPs (Ref. 1), in particular, FA.10 and FA14 and ONR's PSA TAG (Ref. 4).

4.2.20.2 STRENGTHS

592. The full scope UK ABWR PSA results have been systematically reviewed by Hitachi-GE to identify vulnerabilities and insights to allow potential ALARP options to be developed.

593. Hitachi-GE has performed PSA sensitivity studies to support risk informed development of options and to support ALARP studies in other technical areas.

594. I sampled the ‘Use of PSA Sheets’ during the PSA inspection and found that the options identified were feasible and addressed the PSA insights identified. During the PSA inspection undertaken as part of my review, it was clear that the Hitachi-GE PSA team and the engineering departments work closely together, with the engineers aware of the importance of the PSA and the insights provided by the PSA.

595. Options for potential ALARP improvements for further investigation following GDA have been clearly identified and have been recorded in the Hitachi-GE database for managing and transferring commitments to future licensees.

4.2.20.3 FINDINGS

596. The results of the PSA inspection conducted in the Hitachi-GE offices in May 2017 are recorded in Ref. 215, which forms the basis for the findings described below. The inspection followed on from my review of the Hitachi-GE submissions identified above.

597. The process followed by Hitachi-GE applies a cut-off to the frequency of the sequences considered for the identification of the PSA insights. My expectation is that there is no cut-off for the level of risk which could be reduced by implementing ALARP improvements. However, my review has identified that the use of this cut-off has been limited. For example, an integrated review of the PSA reported by Hitachi-GE in Ref. 35 identified any insights that were common in multiple PSAs and goes beyond this cut-off. During the PSA inspection, I reviewed a sample of the detailed records for Hitachi-GE’s review of individual PSA results. I compared these insights with my independent review of the PSA results. My review did not identify any shortfall related to completeness of the PSA insights for the records I sampled.

598. My review identified concerns regarding the lack of specific consideration of fault sequences with low frequencies, but very high consequences in Hitachi-GE’s approach. I communicated ONR’s expectation that any approach developed should be able to measure a reduction on consequences in addition to frequencies in RQ-ABWR-1306. Hitachi-GE did not explicitly address this expectation as part of the PSA submission in GDA. Following parallel interactions with ONR’s severe accident inspector, Hitachi-GE submitted a demonstration of practical elimination of early or large releases (Ref. 216). ONR’s severe accident inspector considered Hitachi-GE’s claim that such sequences have been practically eliminated (Ref. 42) with no specific concerns raised from a deterministic point of view. However, it was identified that there was no specific consideration in Ref. 216 of fault sequences with low frequencies, but very high consequences, and internal hazard initiating events have limited consideration. Further work is needed beyond GDA to consider these aspects, including using the PSA to ensure that the risk from sequences with higher consequences is ALARP when taking the disproportionate nature of the consequences into account. An assessment finding in the severe accident assessment report (Ref. 42) has been raised to capture the need for this further work.

599. From review of Ref. 213 it initially appeared that a number of potential options to reduce risks were identified by the engineering departments, but implementation of the options was not being pursued within GDA. For example, prior to the refinement of the internal fire PSA, spurious signals from the back up building were the dominant risk contributors for the UK ABWR, however development of options to reduce the risks were deferred to beyond GDA. I challenged this in RQ-ABWR-1399 and followed up on

it during the PSA inspection in May 2017. This dominant fault sequence has now been addressed by Hitachi-GE within GDA by identifying three hour fire barriers as a design option to restrict the spread of fire to the cables that give rise to these spurious signals, and implementing this in the PSA (see Section 4.2.11).

600. Hitachi-GE's 'Use of PSA Sheets' were not initially submitted to ONR and were only available to inspect. The review of this information during the PSA inspection (Ref. 215) provided evidence that potential options had been identified beyond those described in Ref. 213 and significant progress had been made to implement them within the GDA design. Following ONR's request, Ref. 35 includes additional information on the options identified by the engineering departments recorded in the 'Use of PSA Sheets' and any progress in implementation in the GDA design, including references to detailed design documents. However, this addition is only a summary and it does not include the totality of the work done during GDA. These records should be reviewed and added to the relevant PSA documentation in the future as appropriate. This detailed documentation should then be transferred to any future licensee. Hitachi-GE has included this in the list of GDA requirements to be transferred to a future licensee (Ref. 217).

601. Over the course of Step 4, the total large release frequency (LRF) for the UK ABWR was reduced by approximately a factor of four, significantly increasing the margin to the BSL of ONR SAPs Target 9 (Ref. 1). The main cause of this reduction was the refinement of the internal fire and internal flooding PSAs, which removed selected conservatisms and took ALARP options identified as part of the review of PSA insights into account. The refinement relies on a number of assumptions about the design, which require substantiation following GDA. Discussion of my assessment of the internal fire and internal flooding PSAs is in Sections 4.2.11 and 4.2.12.

602. As mentioned previously, as part of my assessment, I have undertaken an independent review of some of the PSA results and considered the ALARP position (from a PSA point of view) for the dominant contributors to risk. This has informed the sample of my review. When appropriate, I have also used the PSA to provide risk insights to other technical areas where design alternatives have been part of other inspectors' considerations. I considered the following areas for their contribution to demonstrating that risks are ALARP:

- CCF of the reactor SRVs to open. The SRVs are required to open to allow RPV injection (passively for high pressure feed and actively for low pressure feed) in all sequences where the RPV is isolated. Whilst all 16 SRVs are identical, there is diversity in the actuation systems to protect against CCFs. This is considered in Refs 35 and 218, which have been assessed by ONR's fault studies and mechanical engineering inspectors, with input from PSA. Hitachi-GE has identified that the data used for SRV reliability is conservative, as it does not distinguish between active and passive failures. ONR's fault studies assessment report (Ref. 41) identified the need for future consideration of measures to reduce the CCF probability ALARP for the SRVs.
- Reliability and availability of FLSS. The most important failure modes for FLSS seen in the PSA results are failure of the operator action to initiate FLSS and unavailability of FLSS due to maintenance. The importance of reliability of the operator to initiate FLSS is identified in Ref. 35, in addition the T&M schedule is expected to be risk informed by the PSA.
- Reliability and availability of FLSR. FLSR is currently modelled as a supercomponent in the PSA. Further PSA modelling based upon increased design detail, operating instructions and substantiation is expected in the site specific phase. The importance of FLSR is identified in Ref. 35.

- Injection lines common to both FLSS and FLSR. FLSS and FLSR have a common injection route into the RPV. Blockage of the injection route leads to failure of both FLSS and FLSR. The shared injection route has a low significance in the PSA results, but a high importance when reliability of the components is discounted. This is due to the expected high reliability of the components. Ref. 35 identifies this vulnerability, and identified that 'alternate pathways are available to prevent fuel damage', which are not considered in the PSA. The PSA should reflect the pathways available, including any operator actions, and demonstrate that this is the ALARP solution.
- Loss of class 1 AC bus initiating event. This initiating event contributes to approximately 50% of the fuel damage frequency (FDF) for the SFP and approximately 35% of FDF for internal events during shutdown. This insight is identified in Ref. 35, which identifies design improvements made within GDA, but not reflected in the PSA, and further options for design improvements to be considered beyond GDA. The work beyond GDA needs to ensure that these design improvements are included in the PSA so that their risk benefits can be considered within ALARP decisions beyond GDA.
- The PSA shows that seismic events are the dominant contributor to risk for the UK ABWR, however this risk is dependent on specific characteristics of each site. The seismic PSA (SPSA) needs to be revised to take into consideration, in as a realistic manner as possible, site specific characteristics and plant specific design. The insights identified by Hitachi-GE both during GDA, as identified in Ref. 35, and following development of the site specific PSA should inform the development of the detailed UK ABWR design.
- Manual SCRAM and loss of condenser initiating event. Many of the most risk significant minimal cutsets (MCS) for the IEAP PSA contain the initiating event TM (loss of condenser) and failure to shut down the reactor, leading to plant damage state (PDS) AC. The importance of initiating event TM is identified in Ref. 35, which identifies that further work is required in the site specific phase to identify if reasonably practicable measures could be taken to reduce the initiating event frequency (IEF). A sensitivity study (Ref. 5) performed by Hitachi-GE shows a significant risk reduction if manual SCRAM is able to be claimed in the event of CCF of the RPS. The potential importance of a manual SCRAM claim is also identified in the internal flooding PSA, where ATWS sequences contribute significantly to the internal flooding LRF. This should be investigated further in the site specific phase. The PSA has been effective at highlighting the importance of manual reactor shutdown across a diverse group of plant faults.
- Back-up building barriers. During Step 4 the design of the back-up building has been developed to include rated fire barriers. This was required as the fire PSA identified the spread of fire across unqualified barriers was damaging cabling and giving rise to spurious SRV actuations. The CDF and LRF for the internal fire PSA was reduced significantly in the refined internal fire PSA; see Section 4.2.11. A significant proportion of this risk reduction is due to the additional claims being made on barriers in the back-up building. However, the back-up building fire barriers have not been substantiated within GDA, and therefore an assessment finding has been raised in the internal hazards assessment report (Ref. 123) identifying the need for substantiation. Further options to reduce the internal fire risk from the back-up building are identified in Ref. 35 for consideration in the detailed design beyond GDA.
- Reactor building drains. Reactor building floor drains are assumed capped in the internal flooding PSA. This assumption is not realistic and the drains are unlikely to be capped. Refinement of the PSA to consider uncapped reactor building drains could lead to a reduction in the risk from flooding, as additional time may be available to claim operator action to mitigate the impact of the

flood. Additional prompts to the operator may also be available from alarms and indications in the drainage system. This insight is identified in Ref. 35 and is discussed further in Section 4.2.12 of this assessment report.

- Recovery of fuel pool cooling and clean-up system (FPC) or shutdown cooling (SDC). Many loss of FPC and loss of SDC scenarios have long time periods available before boiling or fuel uncovery. Recovery actions could potentially be claimed for these sequences, and the PSA used to risk inform the measures available to operators. Hitachi-GE performed sensitivity studies which confirm the potential significant risk benefit if recovery actions are able to be claimed. This insight is identified in Ref. 35 and is discussed further in Sections 4.2.15 and 4.2.16 of this assessment report.
- ISLOCA on the RHR. ISLOCA on the RHR caused by valve failures leading to low pressure pipework being exposed to RCS pressure is the most risk significant sequence in the IEAP level 2 PSA. The option to increase the thickness of the RHR piping to be able to contain RCS pressure is identified in Ref. 35 for consideration beyond GDA.
- Reserve ultimate heat sink (RUHS). A sensitivity study submitted by Hitachi-GE identifies that a loss of ultimate heat sink (LUHS) event could be a significant contributor to risk for the site specific PSA (Ref. 23). Loss of the reactor building cooling water system (RCW) / reactor building service water system (RSW) is also a significant contributor to risk for the internal events PSA. A RUHS is included in the deterministic fault schedule, but not the PSA. The PSA should include consideration of RUHS, and the PSA should be used to risk inform the development of the RUHS design and operation. The potential importance of the RUHS is identified in Ref. 35 and inclusion of LUHS in the PSA is identified in assessment finding AF-UKABWR-PSA-006.

603. On the basis of my review, I consider that Hitachi-GE has identified the most significant PSA insights that need to be considered as part of the ALARP demonstration. However, the work submitted by Hitachi-GE does not fully demonstrate that the risks for the UK ABWR are ALARP from a PSA point of view. Further work is needed beyond GDA to address the following shortfalls identified by my review:

- My assessment has identified a number of shortfalls in all the technical areas of the PSA, as reported in the above sections of this assessment report. These will have an impact on the UK ABWR risk profile and risks insights. Once they are addressed beyond GDA, the new PSA results should be reviewed to determine if any additional PSA insights can be identified.
- As part of normal business, site specific aspects shall be included in the PSA, and a review of the PSA results performed to determine if any additional PSA insights can be identified.
- Identified ALARP options incorporated into the PSA models (eg back-up building fire barriers) should be fully substantiated including any assumptions being explicitly identified in the PSA documentation.
- PSA insights are expected to be identified and prioritised based upon risk significance rather than a cut-off criterion being applied. Adequate consideration of higher consequence large releases is also needed.
- A number of options have been identified by Hitachi-GE for implementation or consideration beyond GDA. These options are expected to be considered and implemented if ALARP. I also expect that the PSA should be updated to reflect their impact in the UK ABWR risk profile.

604. In view of the significance of the expectation to demonstrate that risks are reduced ALARP, and taking into account the total risk for the UK ABWR predicted by the PSA,

it is important that the items identified above are scoped, prioritised and addressed, as appropriate, prior to pouring of nuclear island concrete.

> ***Assessment Finding AF-UKABWR-PSA-010****: The licensee shall develop and implement processes and procedures to ensure that PSA insights are systematically identified, prioritised and considered as part of design development. This shall take into account the shortfalls identified by the GDA review in Section 4.2.20 of this assessment report. These describe risk reduction options identified, but intended for implementation beyond GDA, and shortfalls that when resolved may alter the identification and sentencing of ALARP options. The process shall ensure that:*
>
> - *The ALARP options identified in GDA submissions for implementation or consideration beyond GDA have been adequately considered and sentenced by the licensee. This shall be done at the appropriate time to ensure the PSA insights from these options are available to risk inform the appropriate aspects of the detailed design.*
>
> - *The PSA is sufficiently technically developed to support this process, with any relevant shortfalls and insights identified by ONR during GDA being considered and implemented, as appropriate. These shortfalls are identified in Section 4 of this assessment report.*

605. In addition, it is expected that the future licensee further develop the ALARP demonstration throughout the detailed design process using the PSA and develop and implement procedures to ensure that the PSA is considered as part of any design modification or development during the site specific phase.

4.2.20.4 CONCLUSION

606. It is ONR’s policy that new reactors meet the BSLs and strive to meet the BSOs. Comparison of the results of the UK ABWR PSA against SAP Target 9 (Ref. 1) shows that the estimated risk, considering Hitachi-GE internal hazards PSA refinement work, is well below (approximately an order of magnitude) the BSL. However, the risk remains above the BSO for SAPs Target 9 (Ref. 1), and to a lesser extent the Target 8 BSO for doses greater than one Sievert. Therefore, increased regulatory attention was given to the demonstration by Hitachi-GE that the large release frequency (LRF) was reduced ALARP. The PSA was used to inform this process and to identify ALARP options across the plant. The risk is presented as being below the SAPs Target 7 BSO and for Target 8 for doses less than one Sievert.

607. Hitachi-GE has developed and implemented a systematic process for identifying PSA insights and communicated them to the relevant engineering departments for development of potential ALARP options. These options have then been considered for inclusion in the UK ABWR design or identified for further study or implementation beyond GDA. My assessment has not found any major areas of the plant design for which additional ALARP analysis was needed in GDA, from a PSA point of view, to consider alternative features.

608. My report identifies findings which will enhance the ability of the PSA to provide insights into the risks, and when complete these will need to be reviewed by the licensee for any ALARP implications.

609. My assessment supports the view that the PSA is being adequately used to ensure that the UK ABWR risks are being managed towards ALARP as the UK ABWR design process continues through GDA and into the site specific phase. The PSA has been used to identify ALARP improvements which have been incorporated into the GDA reference design and to identify potential ALARP improvements for further

consideration following GDA. However, further work is needed early during the site specific stage to consider the options identified during GDA and potential new ALARP insights resulting from the updated site specific PSA.

4.2.21 Overall Conclusions from the PSA (A1-5)

4.2.21.1 ASSESSMENT

610. This section presents my conclusions of the GDA review of the UK ABWR PSA when compared against relevant expectations in Table A1-5 of ONR's PSA TAG (Ref. 4). My judgement is based on the significance of the outcomes of my review and their potential impact on the risk profile based on Hitachi-GE's sensitivity analyses, and qualitative or quantitative information from the PSA. I have considered the following:

- the adequacy of the PSA documentation;
- whether it is believed that all aspects of the PSA have been subject to sufficient level of independent review by the Requesting Party (RP);
- whether the PSA has a credible and defensible basis;
- whether the PSA reflects the design of the UK ABWR submitted for GDA;
- the adequacy of the process in place to ensure that the PSA assumptions regarding design and operation of the UK ABWR are captured in the development of future procedures, policies and strategies, design, design modifications, etc;
- the adequacy of the process in place to keep the PSA living;
- whether the PSA has enabled a judgement to be made as to the acceptability of the overall risk of the facility against ONR's SAPs numerical targets; and
- whether the PSA has been effectively used to demonstrate that a balanced design has been achieved and that the risk associated with the design and operation of the UK ABWR is ALARP.

4.2.21.2 STRENGTHS

611. The strengths found during the review of the UK ABWR PSA have been described in all the individual technical sections above as appropriate.

4.2.21.3 FINDINGS

612. In response to RI-ABWR-0002, Hitachi-GE significantly improved its PSA documentation. In some areas of the PSA, improvements are required to add the additional information presented during GDA or enhance the explanation of the modelling approach. Specific shortfalls in these areas have been identified in the previous sections and through assessment findings, as described in Section 2.6.

613. My review of the PCSR has identified that the PSA chapter does not establish explicit links between the PSA and claims made in the wider safety case. During my review, as reported previously, I have seen evidence that the links between the PSA and other areas of the safety case exist and, for most of the cases, there are processes in place to communicate information between the different disciplines in Hitachi-GE. On this basis I am satisfied that the PSA supports the safety case; therefore only a minor shortfall has been raised regarding the need to make this link explicit in the PCSR.

614. During GDA the PSA has not been used to inform the categorisation and classification of SSCs. This was in part due to the parallel development of the fault studies and PSA analysis during GDA. It is my expectation that, as part of the use of PSA to support the detailed design development (as discussed in the previous section); the PSA is used to

confirm the categorisation and classification of SSCs. I consider this as part of normal business for development of the use of PSA during site specific stage.

615. As part of closure of RI-ABWR-0002, my review concluded that Hitachi-GE's QA plan and procedures, in particular the peer review process, were adequate to deliver a submission of sufficient quality to enable a meaningful assessment in GDA. I identified two areas that I followed up during GDA:

- The need for the PSA QA processes and plans, including the peer review process, to be improved upon beyond GDA, with lessons learned during GDA being taken into account.
- The lack of clarity regarding outstanding peer review comments and how they will be addressed in the future.

The outcome of my review and required follow-up is presented in Section 4.2.1.

616. As reported in Section 4.2.2, my review team enquired about Hitachi-GE's system to capture assumptions made in the PSA which could be affected by siting, design and construction, or operational matters (such as procedures, maintenance and testing strategies, training programmes, control room staffing and organisation), and which would need to be reviewed as and when detailed information becomes available. Hitachi-GE has developed a process to capture and transfer assumptions to the site specific stage. My review concluded that this process is adequate (see Section 4.2.2), but follow-up is required beyond GDA to confirm the adequacy of the PSA assumptions when additional information becomes available and ensure these are captured in future design, construction and procedure developments.

617. As reported in Section 4.2.1, Hitachi-GE also created a database that captures the design changes developed during GDA that are not reflected in the PSA. An evaluation of the risk impact of these changes was undertaken during GDA. My review identified that the design reference reflected in the PSA is not the same as the UK ABWR GDA design reference. However, there is clarity regarding the differences and their impact on the risk. The PSA should be updated beyond GDA to adequately reflect the design changes and detailed design information.

618. The results of my review and evaluation of the impact on the risk of the shortfalls identified have helped me to judge whether the PSA has a credible and defensible basis and whether in its current state enables a meaningful comparison against ONR's numerical targets presented in ONR's SAPs (Ref 1). The conclusion is that the PSA developed in response to RI-ABWR-0002 meets most of these expectations. The shortfalls identified during the review are not systemic and, in general, their impact on the PSA overall results is considered to be limited.

619. The review findings are discussed extensively in the sections above for each technical area of the PSA. I have conservatory evaluated the impact on the risk of the shortfalls identified in my review. I have identified that the following shortfalls could have the highest impact on the risk profile of the UK ABWR PSA:

- Inclusion of LUHS events and external flooding hazard.
- Modelling of loss of support systems as a reactor trip instead of manual shutdown.
- The modelling of battery supplies, which is not currently aligned with the design reference.
- Lack of substantiation of claims on systems following BOC / ISLOCA events.
- The location and size of a potential consequential LOCA in case of RPV over-pressurisation.

- The modelling of additional operator actions or other means to control water inventory for on-site water sources and operator actions to ensure sufficient inventory and flow rate is available.
- Inclusion of the updated HRA.
- The potential underestimation of the LOOP frequency during shutdown.

620. My review also identified a number of conservatisms and asymmetries in the PSA. In addition, as reported in Section 4.2.20, Hitachi-GE identified a number of PSA insights to be considered in the detailed design development stage that would reduce the risk further. It is important, for the future use of the PSA, that any conservatisms and asymmetries are removed from the PSA so far as is reasonably practicable.

***Assessment Finding AF-UKABWR-PSA-011**: The licensee shall provide a programme to revise the PSA model ensuring that the planned development of the PSA is adequate to support the intended PSA applications at the appropriate time, including:*

- *Development of the detailed design,*
- *Demonstration of ALARP,*
- *Development of operating rules and technical specifications,*
- *Development of arrangements for examination, maintenance, inspection and testing,*
- *Plant configuration control,*
- *Development of operating and emergency procedures and severe accident management guidelines.*

To achieve this, the licensee is expected to programme resolution of the following PSA modelling shortfalls. These are the asymmetric modelling of systems which contain symmetrically redundant trains of equipment, the inclusion of conservatisms to simplify the modelling and various omissions in the PSA identified by the GDA review. The programme shall ensure that the developments are completed and risk insights available prior to the associated design and operational decisions being taken.

4.2.21.4 CONCLUSION

621. Based on the above, I have concluded that the UK ABWR PSA developed in response to RI-ABWR-0002 and the supporting ROs and RQs broadly meet the expectations of ONR’s PSA TAG (Ref.4).

622. The UK ABWR PSA has a credible and defensible basis. There is clarity regarding the differences between the UK ABWR design reflected in the PSA and the design of the UK ABWR submitted for GDA. Additionally the risk significance of these differences has been shown to be small.

623. The UK ABWR PSA is built on a number of assumptions based on the design documentation available at the time when the PSA was developed. However, this was not sufficiently detailed to provide adequate substantiation during GDA. It is important that adequate substantiation is provided when detailed information becomes available. The PSA needs to be revised to reflect the final detailed design, on-site specific characteristics, and operational matters (such as procedures, testing and maintenance (T&M) schedule, refuelling outage strategy, etc).

624. The UK ABWR PSA submitted in GDA enables a comparison to be made against ONR's SAPs numerical targets. As reported in Section 4.2.20 the identification of PSA insights developed by Hitachi-GE is important to support the demonstration that the UK ABWR risk is being managed towards ALARP and for future design development.

625. The PSA chapter of the PCSR presents an adequate summary of the detailed PSA submissions assessed during GDA Step 4 and provides a route map to the detailed PSA documentation.

4.3 Regulatory Issues

626. Regulatory issues (RIs) are matters that ONR judge to represent a 'significant safety shortfall' in the safety case or design and are the most serious regulatory concerns. RIs are required to be addressed before a DAC can be issued.

627. A summary of relevant RIs that were raised and are now closed related to PSA can be found in Annex 4.

4.4 Regulatory Observations

628. Regulatory observations (ROs) are raised when ONR identifies a potential regulatory shortfall which requires action and new work by the RP for it to be resolved. Each RO can have several associated actions.

629. A summary of ROs related to PSA can be found in Annex 4

4.5 Comparison with Standards, Guidance and Relevant Good Practice

630. In Section 2.2 I have listed the standards and criteria I have used during my assessment to judge whether the UK ABWR PSA submission appropriately addressed regulatory expectations described in RI-ABWR-0002, and supporting references, ROs and RQs, and has been carried out adequately with respect to modern standards.

631. I am able to conclude that the UK ABWR PSA has been carried out adequately with respect to these standards to enable a meaningful GDA assessment to be completed.

4.6 Overseas Regulatory Interface

632. ONR has formal information exchange agreements with a number of international nuclear safety regulators, and collaborates through the work of the International Atomic Energy Agency (IAEA) and the Organisation for Economic Co-operation and Development Nuclear Energy Agency (OECD-NEA). This enables us to utilise overseas regulatory assessments of reactor technologies, where they are relevant to the UK. It also enables the sharing of regulatory assessment findings, which can expedite assessment and helps promote consistency.

633. ONR also represents the UK on the Multinational Design Evaluation Programme (MDEP).This seeks to:

- enhance multilateral co-operation within existing regulatory frameworks;
- encourage multinational convergence of codes, standards and safety goals; and
- implement MDEP products in order to facilitate the licensing of new reactors, including those being developed by Gen IV international Forum.

634. In this assessment, the following information from overseas regulators has been used.

635. Since Step 2 GDA, I have continued to work with international regulators and in particular the U.S. Nuclear Regulatory Commission (NRC) via ONR's participation in

the ABWR MDEP. The ABWR MDEP does not have a PSA subgroup. However, ONR's PSA team has provided input to the severe accidents subgroup. Information regarding severe accident design features and analysis has been exchanged and considered in my review of the UK ABWR level 2 PSA. Further information regarding the interface is reported in more detail in Ref. 42.

4.7 Assessment Findings

636. During my assessment 11 residual matters were identified for a future licensee to take forward in their site specific safety submissions. Details of these are contained in Annex 5.

637. These matters do not undermine the generic safety submission and are primarily concerned with the provision of site specific safety case evidence, which will usually become available as the project progresses through the detailed design, construction and commissioning stages. These items are captured as assessment findings.

638. I have recorded residual matters as assessment findings if one or more of the following apply:

- site specific information is required to resolve this matter;
- resolving this matter depends on licensee design choices;
- the matter raised is related to operator specific features / aspects / choices;
- the resolution of this matter requires licensee choices on organisational matters; or
- to resolve this matter the plant needs to be at some stage of construction / commissioning.

639. Assessment findings are residual matters that must be addressed by the licensee and the progress of this will be monitored by the regulator.

4.8 Minor Shortfalls

640. During my assessment 18 residual matters were identified as minor shortfalls in the safety case, but which are not considered serious enough to require specific action to be taken by the future licensee. Details of these are contained in Annex 6.

641. Residual matters are recorded as a minor shortfall if it does not:

- undermine ONR's confidence in the safety of the generic design;
- impair ONR's ability to understand the risks associated with the generic design;
- require design modifications; and
- require further substantiation to be undertaken.

5 CONCLUSIONS

642. This report presents the findings of my Step 4 PSA assessment of the Hitachi-GE UK ABWR.

643. To conclude, I am satisfied with the claims, arguments and evidence laid down within the PCSR and supporting documentation for PSA. I consider that, from a PSA view point, the Hitachi-GE UK ABWR design is suitable for construction in the UK subject to future permissions and permits beings secured.

644. Several assessment findings (Annex 5) were identified; these are for future licensee to consider and take forward in their site specific safety submissions. These matters do not undermine the generic safety submission and require licensee input/decision.

5.1 Key Findings from the Step 4 Assessment

645. Overall, based on the samples undertaken, I am satisfied that the claims, arguments and evidence laid down within the PCSR, and supporting documentation submitted as part of the GDA process, present an adequate safety case for the generic UK ABWR design in the area of PSA. I consider that, from a PSA view point, the Hitachi-GE UK ABWR design is suitable for construction in the UK subject to future permissions and permits being secured.

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214 Topic Report on Use of PSA in ALARP Assessment - Current Status and Future Applications. UKABWR GA91-9201-0001-00232 - AE-GD-0803. Revision 1. Hitachi-GE, February 2017. TRIM 2017/77609.

215 PSA GDA Inspection Scope and Record, ONR, May 2017. TRIM 2017/226710.

216 Demonstration of Practical Elimination of Early or Large Fission Product Release for UK ABWR, UK ABWR GA91-9201-0003-02179 - AE-GD-0992 Revision 0, Hitachi-GE, June 2017. TRIM 2017/254924.

217 Comprehensive list of GDA for requirements & assumptions to be transferred to operating regime, UK ABWR GA91-9201-0003-02055 – XD-GD-0049, Revision 1, September 2017. TRIM 2017/368149.

218 Topic Report on Safety Relief Valve Diversity, UK ABWR GA91-9201-0001-00270 - SE-GD-0601, Revision 0, May 2017. TRIM 2017/173427.

Annex 1

Safety Assessment Principles

SAP No	SAP Title	Description	Interpretation	Comment
FA. 10	**Fault analysis : PSA** Need for PSA	Suitable and sufficient PSA should be performed as part of the fault analysis and design development and analysis.	This principle sets the framework and requirements for a PSA study. The overriding aim of the PSA assessment is to assist ONR judgements on the safety of the facility and whether the risks of its operation are being made as low as reasonably practicable.	Addressed in Section 4 & 5 of this report. RI-ABWR-0002 was issued in Step 3. In response a full scope PSA was submitted and RI-ABWR-0002 closed in Step 4. Section 4.2.20 discusses the need for suitable and sufficient PSA to support demonstration that the risks are ALARP. This assessment report concludes that the PSA is suitable and sufficient for GDA, hence the SAP is met. Assessment findings are raised where shortfalls are required to be addressed after GDA.
FA. 11	**Fault analysis : PSA** Validity	PSA should reflect the current design and operation of the facility or site.	This principle establishes the need for each aspect of the PSA to be directly related to existing facility information, facility documentation or the analysts' assumptions in the absence of such information. The PSA should be documented in such a way as to allow this principle to be met.	Addressed in Section 4.2.1 Whilst the PSA does not reflect the final GDA design reference, the gap is known and understood. Therefore, this SAP is met.
FA. 12	**Fault analysis : PSA** Scope and extent	PSA should cover all significant sources of radioactivity, all permitted operating states and all relevant initiating faults.	In order to meet this principle the scope of the PSA should cover all sources of radioactivity at the facility (eg fuel ponds, fuel handling facilities, waste storage tanks, radioactive sources, reactor core, etc), all types of initiating faults (eg internal faults, internal hazards, external hazards) and all operational modes (eg nominal full power, low power, shutdown, start-up, refuelling, maintenance outages).	Addressed in Section 4.2.1 RI-ABWR-0002 was issued in Step 3. In response a full scope PSA was submitted and RI-ABWR-0002 closed in Step 4. The scope of the PSA is adequate for GDA and the SAP is met.
FA. 13	**Fault analysis : PSA** Adequate representation	The PSA model should provide an adequate representation of the facility and/or site.	The aim of this principle is to ensure the technical adequacy of the PSA. Inspectors should review PSA models, data and results to be satisfied that the PSA has a robust technical basis and thus provides a credible picture of the contributors to the risk from the facility.	Section 4 of this report is almost entirely devoted to this SAP. A number of assessment findings have been raised in various areas, and no GDA Issues have been raised. It can be concluded that this SAP is generally met, however further work is needed following GDA.

<table>
<tr><td>FA. 14</td><td>Fault analysis : PSA
Use of PSA</td><td>PSA should be used to inform the design process and help ensure the safe operation of the site and its facilities.</td><td>The aim of this principle is to establish the expectations on what uses the duty-holders should make of the PSA to support decision-making and on how the supporting analyses should be undertaken.</td><td>There is evidence that the PSA has been used in the design process, including the development of potential ALARP options to be considered or implemented within and following GDA.
Many of the assessment findings raised are aimed at ensuring the PSA is developed sufficiently to aid operational safety decisions in the future, and to demonstrate that risks are ALARP.</td></tr>
<tr><td>NT. 1</td><td>Numerical targets and legal limits
Assessment against targets</td><td>Safety cases should be assessed against the SAPs numerical targets for normal operational, design basis fault and radiological accident risks to people on and off the site.

Target 7: Individual risk to people off the site from accidents

Target 8: Frequency dose targets for accidents on an individual facility – any person off the site

Target 9: Total risk of 100 or more fatalities</td><td>Target 7: BSL 10^{-4}/yr BSO 10^{-6}/yr

Target 8:
<table><tr><td></td><td>BSL</td><td>BSO</td></tr><tr><td>Off-site dose 0.1-1 mSv</td><td>1</td><td>10^{-2}</td></tr><tr><td>Off-site dose 1-10 mSv</td><td>10^{-1}</td><td>10^{-3}</td></tr><tr><td>Off-site dose 10-100 mSv</td><td>10^{-2}</td><td>10^{-4}</td></tr><tr><td>Off-site dose 100-1000 mSv</td><td>10^{-3}</td><td>10^{-5}</td></tr><tr><td>Off-site dose >1000 mSv</td><td>10^{-4}</td><td>10^{-6}</td></tr></table>
Target 9: BSL 10^{-5}/yr BSO 10^{-7}/yr</td><td>Addressed in Section 3 of this report.
The results produced by Hitachi-GE meets the BSOs for Target 7 and dose bands 1 to 4 of Target 8.
Target 9 and Target 8 dose band 5 results are above the BSOs, but well below the BSLs.
The PSA related elements of NT1 are met for GDA, however further work is required following GDA to demonstrate that the risks are ALARP.</td></tr>
<tr><td>NT. 2</td><td>Numerical targets and legal limits
Time at risk</td><td>There should be sufficient control of radiological hazards at all times.</td><td>Sufficient protection based on engineering and operational features.
Avoidance of high point in time risks that would exceed BSLs if evaluated as continuous risks.</td><td>Addressed in Section 2 and Section 4.2.15.
There are times during shutdown when the point in time risks will be elevated. The PSA has been used to develop, and contains explicit modelling of, the outage and T&M schedule. The PSA is not aligned with the final GDA outage schedule, and this is identified as an assessment finding, however the final outage schedule is expected to result in a reduction in risk.
Evaluation of point in time risks as if they were continuous risks has not been provided by Hitachi-GE and full scope PSA (including hazards) has not been conducted for all plant operating states. However the results presented give me confidence that the point at time risks will not exceed BSLs.</td></tr>
</table>

Annex 2

Technical Assessment Guide

TAG Ref	TAG Title
NS-TAST-GD-005 Revision 8	Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable)
NS-TAST-GD-030 Revision 5	Probabilistic Safety Analysis
NS-TAST-GD-045, Revision 3	Radiological Analysis - Fault Conditions

Annex 3

National and International Standards and Guidance

National and International Standards and Guidance
IAEA standards and guidance: • IAEA Standards and Guidance. Safety of Nuclear Power Plants: Design Safety Standard. Specific Safety Requirements SSR-2/1, 2016. • Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants Safety Standard. Specific Safety Guide SSG-3, 2010. • Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants Safety Standard. Specific Safety Guide SSG-4, 2010. • Living Probabilistic Safety Assessment (LPSA). IAEA-TECDOC-1106, 1999. • A Framework for a QA programme for PSA. IAEA-TECDOC-1101, 1999. Safety Assessment and Verification for Nuclear Power Plants. Safety Guide. International Atomic Energy Agency (IAEA) Safety Standards Series No. NS-G-1.2, IAEA, Vienna, 2001, TRIM 2007/29995
Western European Nuclear Regulators' Association. Reactor Harmonization Group. WENRA Reactor Reference Safety Levels, 2014.
ASME standards and guidance: • ASME/ANS RA-Sa-2009 (and Addenda ASME/ANS RA-S-2008), 2009. Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications. • Advanced Light Water Reactor PRA Standard. ANS/ASME JCNRM RA-S 1-5 [DRAFT]. • Requirements for Low Power and Shutdown PRA. ANS-58.22-2014, 2014
NUREG/CR-6850 (EPRI TR-1011989) Fire PRA Methodology for Nuclear Power Facilities, Electric Power Research Institute and U.S. Nuclear Regulatory Commission, September 2005.
EPRI standards and guidance: • EPRI 1019259, Fire Probabilistic Risk Assessment Methods Enhancements Supplement 1 to NUREG/CR-6850 • EPRI 1011989, Technical Report, September 2010. • EPRI 1019194, Guideline for Performance of Internal Flooding Analysis Probabilistic Risk Assessment, Final Report, December 2009

Annex 4

Regulatory Issues / Observations

RI / RO Ref	RI / RO Title	Description	Date Closed	Report Section Reference
RI-ABWR-0002	Probabilistic Safety Analysis: Project Plan and Delivery	The objective of this RI is to state ONR's expectations with respect to Hitachi-GE developing and delivering a suitable and sufficient probabilistic safety analysis for the UK ABWR fault analysis as part of the GDA submission.	23/02/17	4.2.1
RO-ABWR-0013	Probabilistic Safety Analysis: Project Plan and Delivery	The objective of this RO is to state ONR's expectations related to the development and delivery of the PSA for the UK ABWR as part of the GDA submission and to gain early confidence that Hitachi-GE will be able to deliver a full scope, modern PSA within the GDA timeframes. This RO was escalated to RI-ABWR-0002.	02/09/15	4.2.1
RO-ABWR-0040	PSA: Identification of Applicable Internal Hazards	The objective of this RO is to state ONR's expectations related to the identification of internal hazards for the UK ABWR PSA and to request Hitachi-GE to respond to the shortfalls identified by ONR's review.	09/06/17	4.2.10
RO-ABWR-0041	PSA: Identification of Applicable External Hazards	The objective of this RO is to state ONR's expectations related to the identification of external and internal hazards for the PSA and request Hitachi-GE to respond to the shortfalls identified by ONR's review.	09/06/17	4.2.13
RO-ABWR-0042	PSA internal initiating events at power	The main objective of this RO is to state ONR's expectations related to the identification and grouping of initiating events for the UK ABWR PSA and request Hitachi-GE to respond to the shortfalls identified by ONR's review.	22/05/17	4.2.4
RO-ABWR-0046	UK ABWR Containment Performance Analyses for Severe Accidents	The objective of this RO is to state ONR's expectations related to the containment performance characterisation and its documentation.	25/06/17	4.2.18
RO-ABWR-0048	Level 2 PSA methodology	The objective of this RO is to state ONR's expectations related to the UK ABWR level 2 PSA and request Hitachi-GE to respond to the shortfalls identified by ONR's review.	01/06/17	4.2.18
RO-ABWR-0053	Level 1 and level 2 PSA for internal events during operation at power - System Analyses	The main objective of this RO is to state ONR's expectations related to the system analyses for the UK ABWR PSA (for internal initiating events during operation at power) and request Hitachi-GE to address the shortfalls identified by ONR's review.	02/06/17	4.2.7
RO-ABWR-0076	PSA ALARP Demonstration and Optioneering	The objective of this RO is to state ONR's expectations related to the demonstration that the risk calculated by the UK ABWR PSA is as low as is reasonably practicable (ALARP).	14/07/17	4.2.20

Annex 5

Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-UKABWR-PSA-001	Because of the importance, and the regulatory expectation, of using the PSA to risk inform the design, demonstrate that numerical targets are met and that risks are reduced to ALARP, the licensee shall: 1. Develop processes and procedures to ensure that the PSA is kept living and is aligned with the design reference. Implementation of this process should ensure that differences between the PSA and the final GDA design reference are adequately addressed. 2. Develop an overall programme which ensures that the shortfalls and future PSA development needs presented in this assessment report (summarised in Annex 7) are included in the plans for the site specific PSA, such that risk insights are able to be identified and utilised to inform associated design and operational decision making. 3. Develop processes and procedures to ensure the PSA assumptions are captured in future design, construction and procedure development. This process should also ensure that the PSA model and documentation is updated to reflect any changes to assumptions as more detailed information becomes available.	General Expectations – Approaches and Methodologies (A1-1.1), PSA Scope (A1-1.2), Freeze Date (A1-1.3)
AF-UKABWR-PSA-002	Because of the risk significance of the assumptions made concerning outage, maintenance and test unavailabilities, and the lack of information available at GDA to substantiate these assumptions, the licensee shall ensure that the basis for the modelling and assumptions concerning outage, maintenance and test unavailabilities of systems and components (including standby) used in the PSA, is justified and aligned with the technical specifications and maintenance programmes, or alternative values / strategies justified.	Unavailabilities Due to Testing and Maintenance (A1-2.6.3)
AF-UKABWR-PSA-003	Because of the potential risk significance of intersystem common cause failures, the licensee shall use the PSA to identify intersystem common cause failure effects for the UK ABWR following on from the work in GDA. The results shall be used to inform the incorporation of appropriate defences and, where appropriate, intersystem common cause failures should be included explicitly in the model.	Common Cause Failures (CCFs) (A1-2.6.4)

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-UKABWR-PSA-004	Because of the risk significance of internal hazards, and the dependency on site specific design and layout, The licensee shall provide a revised systematic prioritisation of all internal hazards, including combined internal hazards, for all sources of radioactivity on-site that is representative of the site specific design and layout and consistent with the internal hazards deterministic safety case. The prioritisation shall include demonstration that the risk associated with all the screened out internal hazards would be insignificant compared to the UK ABWR total risk. The licensee shall then provide a revised PSA for internal hazards on the basis of the prioritisation performed.	Level 1 PSA: Analysis of Hazards – Prioritisation of Internal Hazards (A1-2.7-1)
AF-UKABWR-PSA-005	Because of the risk significance of external hazards and the dependency on site specific characteristics, design and layout, the licensee shall provide a revised systematic prioritisation of external hazards. The prioritisation shall consider all sources of radioactivity on-site and the specific characteristics of the site. The analysis should address external hazards that could be correlated. The licensee shall provide a demonstration that the risk associated with all the external hazards screened out would be insignificant compared to the total risk. The licensee shall then provide a revised PSA for external hazards on the basis of the prioritisation performed.	Level 1 PSA: Analysis of Hazards – Prioritisation of External Hazards (A1-2.7)
AF-UKABWR-PSA-006	Because of the potential risk significance of loss of ultimate heat sink events, the licensee shall consider loss of ultimate heat sink initiating events (including biological fouling) and external flooding initiating events within the site specific PSA, or adequately justify their exclusion. The analysis shall take site specific heat sink design and expected operator actions into account. The licensee shall use the analysis to identify any relevant PSA insights to aid improvement of the design or operation of the UK ABWR.	Level 1 PSA: Analysis of Hazards – Prioritisation of External Hazards (A1-2.7)
AF-UKABWR-PSA-007	Because of the risk significance of internal fire and internal flooding events in the GDA PSA, the licensee shall provide revised internal fire and internal flood PSAs for shutdown and spent fuel pool operations which are consistent in detail and scope to the at power analysis. The revised PSAs shall reflect the site specific design, operation and maintenance of the UK ABWR and take any relevant shortfalls identified by the GDA review into account.	Level 1 PSA: Low Power and Shutdown Modes (A1-2.8) and Level 1 PSA: Spent Fuel Pool and Fuel Route PSA

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-UKABWR-PSA-008	Because of the large difference identified in the monte carlo generated mean and the point estimate results in the GDA PSA, review and explanation is needed to confirm confidence in the numerical results. Therefore the licensee shall review the uncertainty analysis for core damage frequency and large release frequency, to identify the cause for the significant difference in the monte carlo generated mean and the point estimate results and, if appropriate, the licensee shall put in place measures to resolve the cause of the significant difference.	Level 1 PSA: Uncertainty Analyses, Quantification and Interpretation of the Level 1 PSA Results (A1-2.9)
AF-UKABWR-PSA-009	Because of the site specific nature of the level 3 PSA and the shortfalls identified in the GDA review, the licensee shall provide a revised level 3 PSA model and documentation, as part of the development of the site specific PSA, which takes into consideration the following: • Justification for the decontamination factors applied to the barriers to fission product release. This shall including those for the standby gas treatment system. • Updating the population data to reflect the most recent census, when reasonably practical to do so. This is needed to provide a more realistic assessment of dose uptake. • Consideration and justification for the expected increase in notional fatalities projected to the end of station life. The use of the most recent census data will assist this. • Model multiple release phases to more realistically model spent fuel pool fault sequences, or use and justify an alternate method for comparison against SAPs Target 7. • Revise the method for comparison to SAPs Target 9 to release frequency multiplied by conditional probability of exceeding 100 fatalities.	Level 3 PSA (A1-4)

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-UKABWR-PSA-010	Because of the ongoing regulatory expectation to demonstrate that the risks are being managed ALARP, the licensee shall develop and implement processes and procedures to ensure that PSA insights are systematically identified, prioritised and considered as part of design development. This shall take into account the shortfalls identified by the GDA review in Section 4.2.20 of this assessment report. These describe risk reduction options identified but intended for implementation beyond GDA, and shortfalls that when resolved may alter the identification and sentencing of ALARP options. The process shall ensure that: • The ALARP options identified in GDA submissions for implementation or consideration beyond GDA have been adequately considered and sentenced by the licensee. This shall be done at the appropriate time to ensure the PSA insights from these options are available to risk inform the appropriate aspects of the detailed design. • The PSA is sufficiently technically developed to support this process, with any relevant shortfalls and insights identified by ONR during GDA being considered and implemented, as appropriate. These shortfalls are identified in Section 4 of this assessment report.	Use of PSA in ALARP Demonstration

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-UKABWR-PSA-011	Because of the importance and regulatory expectation of using the PSA to risk inform design and operation of the UK ABWR, the licensee shall provide a programme to revise the PSA model ensuring that the planned development of the PSA is adequate to support the intended PSA applications at the appropriate time, including: • Development of the detailed design, • Demonstration of ALARP, • Development of operating rules and technical specifications, • Development of arrangements for examination, maintenance, inspection and testing, • Plant configuration control, • Development of operating and emergency procedures and severe accident management guidelines. To achieve this, the licensee is expected to programme resolution of the following PSA modelling shortfalls. These are the asymmetric modelling of systems which contain symmetrically redundant trains of equipment, the inclusion of conservatisms to simplify the modelling and various omissions in the PSA identified by the GDA review. The programme shall ensure that the developments are completed and risk insights available prior to the associated design and operational decisions being taken.	Overall Conclusions from the PSA (A1-5)

Annex 6

Minor Shortfalls

Minor Shortfall Number	Minor Shortfall	Report Section Reference
MS-UKABWR-PSA-001	Lessons learned during GDA and the peer review process should be reflected in the PSA QA plan and processes.	General Expectations – Approaches and Methodologies (A1-1.1), PSA Scope (A1-1.2), Freeze Date (A1-1.3)
MS-UKABWR-PSA-002	Shortfalls identified related to PSA assumptions should be reviewed and resolved in the PSA model and documentation. The PSA assumptions list and the PSA documentation should be consistent, with timely propagation of new and closed assumptions.	General Expectations – Assumptions in the PSA (A1-1.5)
MS-UKABWR-PSA-003	The descriptions of gates in the PSA models and the documentation of the fault tree gates should be improved.	General Expectations – Computer Codes and Inputs (A1-1.4)
MS-UKABWR-PSA-004	Any findings raised by ONR relating to the verification and validation of the CAFTA software should be addressed.	General Expectations – Computer Codes and Inputs (A1-1.4)
MS-UKABWR-PSA-005	Consideration should be made if an update of the thermal-hydraulic and/or severe accident analyses to achieve more 'best estimate' results is needed in future updates of the PSA. Any update to the thermal-hydraulic or severe accident analyses to achieve more 'best estimate' results should be reflected in the PSA.	General Expectations – Computer Codes and Inputs (A1-1.4)
MS-UKABWR-PSA-006	A more detailed justification of the difference in results between SAFER, SHEX and MAAP that are relevant to the PSA should be developed and included in the PSA documentation.	General Expectations – Computer Codes and Inputs (A1-1.4)
MS-UKABWR-PSA-007	The licensee should enhance the documentation of the derivation of success criteria to include all the analysis cases in a single document with a clear identification of inputs, systems available, actuation times, and the resulting RPV and containment conditions. The documentation should also be improved to provide clear identification of the minimum equipment requirements and performance for success and how the success criteria bound all potential actuations which could contribute to more severe conditions.	Level 1 PSA: Accident Sequence Development – Determination of Success Criteria (A1-2.2)

Minor Shortfall Number	Minor Shortfall	Report Section Reference
MS-UKABWR-PSA-008	The GDA review has identified the PSA documentation of the accident sequence analysis should be improved to provide more detailed information and justification of the event tree logic adopted, including the following: • Sufficiently detailed explanation of the gate structures to provide assurance that the flag files are free from errors and reasonably represent the intended sequence logic. In addition, it may be more beneficial in the long term to consider replacing the flag structures with actual event tree and/or fault tree logic. • The functional description of the event tree nodes and its applicability to each event tree branch. • Systematic and clear demonstration for the event tree success sequence end point (24 hours). • Detailed description of sequence dependencies. • In some cases more detailed justification for the accident sequences should be provided in the documentation. The documentation of LOOP / SBO event tree modelling remains an example of lack of clarity regarding the justification for the fault tree modelling.	Level 1 PSA: Accident Sequence Development – Event Sequence Modelling (A1 2.3)
MS-UKABWR-PSA-009	The documentation of the system analyses has been identified by the GDA review as an area which could benefit from enhancement, including: • Description of system operating and shutdown modes including abnormal operations. • Detailed information regarding the success criteria of front line and support systems. • Expansion of the discussions of system secondary functions along with the basis for why these functions are or are not included in the PSA. • Detailed description of the system boundary and interface between systems. • Extension of the documentation on circular logic to all relevant systems. • Modelling of bypass of trip and sensor signals and how maintenance will impact system availability.	Level 1 PSA: System Analysis (A1-2.4)
MS-UKABWR-PSA-010	The GDA review identified that there was not a clear auditable trail for the identification of component failure modes included in the PSA.	Random Component failure Probabilities (A1-2.6.2)
MS-UKABWR-PSA-011	The seismic PSA documentation should be extended to provide additional information and justification on: • Extrapolation of the seismic hazard curve beyond 1.2g PGA. • SSC status in the seismic equipment list. • Additional discussion on the shutdown seismic event tree accident sequence modelling.	Level 1 PSA: Analysis of Hazards – Seismic Analysis (A1-2.7-4)
MS-UKABWR-PSA-012	The SFP PSA documentation should be improved in the areas noted in Section 4.2.16.	Level 1 PSA: Spent Fuel Pool and Fuel Route PSA

Minor Shortfall Number	Minor Shortfall	Report Section Reference
MS-UKABWR-PSA-013	Additional model development should be considered to improve the model quantification process and run time.	Level 1 PSA: Uncertainty Analyses, Quantification and Interpretation of the Level 1 PSA Results (A1-2.9)
MS-UKABWR-PSA-014	The presentation of the PSA results should be reviewed to ensure that a complete description of the results is provided, including individual sequence contribution to PDS, release categories and total LRF and potential design and operation vulnerabilities are clearly identified.	Level 1 PSA: Uncertainty Analyses, Quantification and Interpretation of the Level 1 PSA Results (A1-2.9)
MS-UKABWR-PSA-015	The large release definition should be reconsidered to determine if an enhanced definition could provide better alignment to the SAP Target 9 criterion.	Level 1 PSA: Uncertainty Analyses, Quantification and Interpretation of the Level 1 PSA Results (A1-2.9)
MS-UKABWR-PSA-016	The level 2 PSA documentation should be improved in the areas noted in Section 4.2.18.	Level 2 PSA (A1-3)
MS-UKABWR-PSA-017	The level 3 PSA documentation should be updated to include additional information provided in RQ responses, including RQ-ABWR-1104,	Level 3 PSA (A1-4)
MS-UKABWR-PSA-018	The PCSR should be updated to present the links between the PSA and other areas of the safety case.	Overall Conclusions from the PSA (A1-5)

Annex 7

Summary Information to Support Assessment Finding AF-UKABWR-PSA-001

Report Section Reference or PSA Topic Area	Summary of Shortfalls Identified within the Step 4 GDA PSA Assessment Report
Level 1 PSA: Identification and Grouping of Initiating Events (A1-2.1)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for initiating events: • The initiating events identified as missing in Section 4.2.4 of this assessment report. • A revised loss of support system initiating event group that adequately represents the plant response following a support system CCF (complete or partial) initiating event, and explicitly includes any relevant operator actions. • A documented comparison between the initiating events considered in the deterministic fault schedule and the initiating events in the PSA. Any differences identified should be justified.
Level 1 PSA: Accident Sequence Development – Determination of Success Criteria (A1-2.2)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for accident sequence analysis: • Clear identification and justification of the minimum equipment requirements and performance for each success criterion, including addressing the specific shortfalls identified in Section 4.2.5 of this assessment report. • The impact of potential actuations that could contribute to more severe conditions, including potential system malfunctions. • Adequately accounting for the impact of physical and environmental conditions that arise during the evolution of LOCA, BOC and ISLOCA accidents. • Confirmation of any requirement for containment heat removal following LOCA outside containment. • The expected life of class 2 DC batteries and any other measure included in the design to ensure core cooling following depletion of DC batteries.
Level 1 PSA: Accident Sequence Development – Event Sequence Modelling (A1 2.3)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for event sequence modelling: • Dependencies noted as missing or unclear in Section 4.2.6 of this assessment report. • The accident sequence shortfalls identified in Section 4.2.6 of this assessment report (such as, multiple cycles of vacuum breakers, containment isolation failure, loss of RPV level instrumentation, late containment failure). • ‘Best estimate’ representation of ATWS sequences. • Late containment failures and the impact on systems. • The location and size of a potential consequential LOCA in case of RPV over-pressurisation. • Adequate reflection of the on-site inventories and any operator actions or other means to control water inventory and ensure sufficient inventory and flow rate is available.

Report Section Reference or PSA Topic Area	Summary of Shortfalls Identified within the Step 4 GDA PSA Assessment Report
Level 1 PSA: System Analysis (A1-2.4)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for system analysis: • Failure modes identified as missing in in Section 4.2.7 of this assessment report (such as, structural failures, failure of the alarms or indications, CCFs, flow diversion valves, re-start basic events, spurious opening of SRVs, control of RPV water level, electrical system). • Substantiation of the success criteria of HVAC, taking ambient conditions into account. This shall include explicitly stating the maximum temperature limit for all rooms containing SSCs claimed in the PSA and the associated HVAC success criteria. • The impact of environmental conditions, other than loss of room cooling, on systems operability. This shall include environmental conditions after containment failure or high energy pipe breaks outside containment, heating of the water supply in the suppression pool, debris that could plug screens/filters both internal and external to the plant. • Systems dependencies that were not included in the PSA due to loss of information on support systems (such as, HWBS, SAuxP, FLSR, FLSS support systems). • Revised system analysis which is detailed such that the use of supercomponents has been avoided or minimised. • All the relevant systems are modelled (HPIN, long term nitrogen makeup). • Identification of any additional IEs. • Substantiation for the digital C&I reliability data including CCFs
Level 1 PSA: Human Reliability Analysis (A1-2.5)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for HRA: • Alignment with the final GDA HRA. • Human failure events, dependencies or information noted as missing in Section 4.2.8 of this assessment report. • Comprehensive treatment of dependencies between the level 1 and level 2 PSA, taking note of the shortfalls identified in Section 4.2.8 of this assessment report.
Initiating Event Frequencies (A1-2.6.1)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for initiating event data: • Update LOCA initiating event frequency to reflect all the relevant sizes of LOCAs. • Update LOOP frequency, conditional LOOP and LOOP recovery probabilities for all operating states (at power, shutdown, low power) to reflect the site specific characteristics.
Random Component failure Probabilities (A1-2.6.2)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for component data: • Potential optimism in testable check valve data. • Variation in water quality between systems. • Structural failures of active components. • Discovery of latent failures. • A justification of SRV data used (active vs passive).

Report Section Reference or PSA Topic Area	Summary of Shortfalls Identified within the Step 4 GDA PSA Assessment Report
Common Cause Failures (CCFs) (A1-2.6.4)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for CCFs: • Improvement of the sensitivity study on CRD CCF to consider components other than the CRD drives. • Consideration of RVI line CCF. • Substantiation of digital C&I CCF data. • Detailed justification, including consideration of sub-components, of diversity between EDGs and BBGs.
Level 1 PSA: Analysis of Hazards – Prioritisation of Internal Hazards (A1-2.7-1)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for internal hazards: • The PSA internal hazards prioritisation documentation should be updated to explicitly include consideration of hazards within the PCV and identify if any additional hazards are required to be considered in the PSA. • The comparison of the probabilistic and deterministic internal hazards analyses should be reviewed following GDA, to take any late developments in the deterministic internal hazards analyses into account along with any relevant PSA or deterministic hazard assessment findings. • Substantiation should be provided for the claim that HVACs are designed such that hazardous materials located outside of the MCR compartment cannot prevent the delivery of the fundamental safety functions. • Hazard impacts on sources of radioactivity other than the reactor and the SFP should be considered fully.
Level 1 PSA: Analysis of Hazards – Analysis of Internal Fires (A1-2.7-2)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for internal fire PSA: • The technical shortfalls described in Section 4.2.11 of this report. • Consider the physical impact of a fire on a protected cable in the room of its termination. This shall determine whether the exposed cable and/or the equipment needs fire protection. • The main control room fire analysis modelling properly reflects the plant design. The licensee shall ensure that a fire in the main control room will not cause spurious pump motor or motor operated valve actuations with fire induced bypass of the valve protective torque and limit switches. • Include the circulating water building and equipment within the global plant analysis boundary during the site specific analysis. • The internal fire at power refinement study has been reviewed and any new assumptions are recorded in the PSA assumptions list (or equivalent database that the licensee puts in place).
Level 1 PSA: Analysis of Hazards – Analysis of Internal Flooding (A1-2.7-3)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for internal flood PSA: • The analyses reflect the drainage design. • The flooding scenarios have been improved to ensure that conservatism is avoided in risk significant flooding scenarios. • The revised grouping of internal flooding initiating events is not unduly conservative. • Relevant steam release paths to ensure that the survivability of safety equipment claimed in the PSA is justified, particularly where the safety equipment clamed is within the same division as the steam release.

Report Section Reference or PSA Topic Area	Summary of Shortfalls Identified within the Step 4 GDA PSA Assessment Report
Level 1 PSA: Analysis of Hazards – Prioritisation of External Hazards (A1-2.7)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for external hazards: • Substantiation is required for the claim that HVAC vents, steel doors and the buildings to prevent energetic tornado missiles from entering buildings containing SSCs. • The tornado data should be made site specific and take historical data into account. • Substantiation is required for the claim that FLSR will be available following external hazard events.
Level 1 PSA: Analysis of Hazards – Seismic Analysis (A1-2.7-4)	The licensee shall update the seismic PSA, specifically including update of the seismic fragility analysis to site specific data, and to adequately reflect site specific design considerations. As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for seismic PSA: • Expanding the scope to include all operational states and sources of radioactivity in a fully probabilistic manner. • Consideration of crane collapse in all POSs. • Additional clarification and justification of why the impact of seismic-induced failure of the reactor building does not lead directly to core damage. • Consider claiming class 3 systems following a seismic event. • Considering consequential hazards that might be caused by a seismic event and, if appropriate, include them in the PSA. This should include revised seismic - fire or flooding interaction analysis, consideration of seismic-induced failure of masonry block walls and seismic-induced external dam failure using site specific information during the detailed design phase. • Extension of the seismic specific HRA to reflect site specific characteristics and procedures. • Shortfalls identified concerning the seismic fragility analysis in Section 4.2.14.
Level 1 PSA: Low Power and Shutdown Modes (A1-2.8)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for shutdown PSA: • Revise and extend the list initiating events (including low power initiating events) to consider the site specific procedures and T&M schedule (eg consideration of rod withdrawal error at low power)and additional HRA-B initiating events. • Justification for the size and location of the consequential LOCA due to RPV overpressure where SRVs fail to open. This should include substantiated characterisation of failure of SRVs to open. • Consideration of consequential and environmental impacts of BOCs and LOCAs. • Analyse the impact of water injection after the fuel assemblies are uncovered with a view to providing clear guidance to the operators on the best course of mitigation action. • Consider operator action to control of the RPV water level to conserve feed water stocks. • Consider injection routes not modelled in the PSA, but identified as available in the 'Use of PSA in ALARP' topic report. • Refinement of conservative assumptions on water level and status of the RPV head. • Recognition of the long time periods available for some shutdown sequences, and use the PSA to risk inform the response available for these faults. • Provide an evaluation of point in time risks.

Report Section Reference or PSA Topic Area	Summary of Shortfalls Identified within the Step 4 GDA PSA Assessment Report
Level 1 PSA: Spent Fuel Pool and Fuel Route PSA	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for SFP and fuel route PSA: • Consider the possibility of a cask drop event causing a SFP leak beyond the ability of the plant to make-up, supported by revised structural analysis of cask drop events to address the shortfalls and uncertainties identified in GDA. • Consider the conservatisms noted in Section 4.2.16 on long duration sequences, fire protection system makeup and the definition of the fuel damage criterion. • Address improvements identified in GDA to reduce the frequency of occurrence of operator induced loss of FPC. • Represent the hydrogen management strategy, including required operator actions and the impact of hydrogen accumulation in severe accident sequences. • Explicitly consider loss of inventory in POS E.
Level 1 PSA: Uncertainty Analyses, Quantification and Interpretation of the Level 1 PSA Results (A1-2.9)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised uncertainty analyses: • Addressing the shortfalls identified by ONR and specialist TSC reviews of sensitivity studies, as identified in Section 4.2.17. • Review of any relevant international guidance, as identified in Section 4.2.17. • Additional consideration of ECCS suction strainer reliability. • Additional consideration of severe accident phenomenology.
Level 2 PSA (A1-3)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for level 2 PSA: • Addressing the severe accident phenomena that were identified as not adequately treated or omitted in Section 4.2.18 of this assessment report (such as, reactivity, hydrogen combustion in the reactor building, bypass of the suppression pool, omitted conditions in PDS AC, the potential for creep rupture of the MSL, length of de-inerted period). • Consider the omitted, unjustified or simplified treatment of failure modes and SSCs in the CET identified Section 4.2.18 of this assessment report (such as, latent, incident failures of containment, explicit modelling of operator cues, loss of level indication, S/P pH control system, SRV re-closure, HPIN). • Address areas of conservatisms or optimism noted in Section 4.2.18 of this assessment report (such as, timing of injection to prevent RPV breach, claims on high pressure injection, drywell venting, suppression pool bypass modelling). • Improve identification of environmental effects related to core damage progression and containment leakage or failure and explicit inclusion in PSA. • Identify assumptions related to the systems survivability and effectivity and operator actions during severe accident progression, and other degraded conditions. • Revise the release category groups in the level 2 PSA in line with the outcome of the containment performance analysis so that they accurately reflect the timing and magnitude of the release. • During the site specific development of the PSA, the conservatisms in the level 2 PSA should be reduced to make the assessment both more realistic and more useful to decision-makers.

Report Section Reference or PSA Topic Area	Summary of Shortfalls Identified within the Step 4 GDA PSA Assessment Report
Level 2 PSA: Containment Performance Analysis (A1-3.3)	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised Containment Performance Analysis: • Consideration of a 'best estimate' pressure-temperature envelope for drywell head flange; • Update RCCV models to reflect site specific design, including consideration of thermal deformation where the steel liner interfaces with the concrete containment. • Inclusions of a complete list of containment challenges that is extended to cover missing scenarios (high containment water level more than 17.15m) and negative containment pressure (failure to open of the vacuum breakers). • Revised and substantiated analysis of MCCI that addresses the shortfalls identified by the PSA and severe accident analysis (consideration of challenges due to high temperatures on V/B, load-bearing capacity of the pedestal wall, modelling of vent pipes, impact of pedestal wall collapse on the PCV)
Level 3 PSA (A1-4) – Non Reactor Faults and Low Consequence Faults	As part of the development of the site specific PSA, the following shortfalls identified by the GDA review should be taken into account in the revised PSA models and documentation for non-reactor and low consequence faults: • All non-reactor faults identified in the DBA FMEAs for non-reactor SSCs shall be explicitly included in the PSA as an initiating event or explicitly considered as part of an initiating event group in the PSA, with the PSA documentation identifying the link between the PSA and DBA. • The PSA should consider a range of sizes of breaks on the turbine system, in a range of locations. • The PSA should probabilistically consider the protection available for each non-reactor fault.
Level 1 PSA: Accident Sequence Development – Event Sequence Modelling (A1 2.3) And Level 2 PSA (A1-3)	As part of the development of the site specific PSA the licensee shall ensure that the PSA is aligned with the site specific operating procedure, emergency operating procedures (EOPs) and severe accident management guidelines (SAMGs), and they are risk informed using the PSA. This shall include taking into account the shortfalls identified by the GDA review: • Address specific shortfalls identified by the GDA review (such as, including priority to use water sources, control of water level in the RPV, crew intervention during ATWS, operator prioritisation of low pressure system injection systems). • Align the POSs in the PSA with the site specific operating states and ensure that the link with the site specific technical specifications is clear. • Ensure that the link between the level 2 PSA CETs and the SAMGs is transparent and explicitly documented. • Any additional initiating events as a result are identified and included in the PSA.

Annex 8

Plant Operating States applied in the PSA

Plant Operating State (POS)	Description
S	Transition to reactor cold shutdown
A	Transition to reactor disassembly and reactor well gate open with Division 2 in maintenance
B-1	Full water level in reactor well and gate open with Division 2 in maintenance
B-2	Full water level in reactor well and gate open with Divisions 1 and 3 in maintenance
C	Transition to closed condition of PCV/RPV heads with Divisions 1 and 3 in maintenance
D	Preparation of plant start up
E	Full core off-loaded to the spent fuel pool
F	Reactor at power or low power states

Note: Plant Operating State Descriptions taken from the PSA Summary Report (Ref. 11)