Office for Nuclear Regulation

New Reactors Division

Step 4 Assessment of Severe Accidents for the UK Advanced Boiling Water Reactor

Assessment Report: ONR-NR-AR-17-015
Revision 0
December 2017

Published 12/17

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EXECUTIVE SUMMARY

Hitachi-GE Nuclear Energy Ltd is the designer and GDA Requesting Party for the United Kingdom Advanced Boiling Water Reactor (UK ABWR). Hitachi-GE commenced Generic Design Assessment (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 severe accidents.

The scope of the Step 4 assessment is to review the UK ABWR severe accidents safety case and supporting analyses against the expectations of ONR's Safety Assessment Principles (SAPs) and relevant international guidance.

My assessment conclusions are:

- Hitachi-GE has provided suitable severe accident analysis of the UK ABWR, complementing the wider safety case. Relevant severe accident sequences have been identified for the all operating modes, including the reactor at power, the reactor during shutdown and the SFP.
- Hitachi-GE has analysed severe accidents using computer codes which are suitable for modelling the accident phenomena relevant to the UK ABWR.
- Hitachi-GE has identified the importance of the containment in its severe accident management strategy and has identified the relevant challenges to the containment.
- Severe accident analysis has been used to demonstrate the effectiveness of severe accident measures in preventing and/or mitigating accidents.
- Hitachi-GE has provided a clear explanation of how severe accidents would be managed and, for the purposes of GDA, has adequately described strategies and concepts that can be taken forward by the future licensee.
- Hitachi-GE has explained how learning from the accidents at Fukushima Dai-ichi has been used to positively influence the design of the UK ABWR.
- In accordance with UK and international expectations, I consider that Hitachi-GE has presented a thorough report on practical elimination of large or early fission product release for the design.
- Hitachi-GE has demonstrated that the severe accident design features support ALARP claims on the adequacy of the UK ABWR design.
- A severe accident safety case has been presented which is adequate for GDA.

My judgement is based upon the following factors:

- A review of the severe accident analysis performed by Hitachi-GE for the UK ABWR.
- A review of the accident management strategies and concepts that have been provided in GDA.
- A consideration of the severe accident design features of the UK ABWR against UK and international learning following the accident at Fukushima Dai-ichi.

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

Several assessment findings have been identified; these are for a 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.

LIST OF ABBREVIATIONS

AC	Air Conditioning
AE	Accident sequence – large LOCA with injection failure
ALARP	As Low As Reasonably Practicable
ADS	Automatic Depressurisation System
AHEF	Alternative Heat Exchange Facility
AMG	Accident Management Guideline
ANI	Alternative Nitrogen Injection
ATWS	Anticipated Transient Without Scram
BAF	Bottom (of) Active Fuel
B/B	Backup Building
BBG	Backup Building Generator
BDB	Beyond Design Basis
BDBA	Beyond Design Basis Analysis
BiMAC	Basemat-internal Melt Arrest Coolability
BSL	Basic Safety Level
BSO	Basic Safety Objective
BWR	Boiling Water Reactor
CFD	Computational Fluid Dynamics
C&I	Control & Instrumentation
CMSS	Core Melt Stabilisation System
COPS	Containment Overpressure Protection System
CST	Condensate Storage Tank
DAC	Design Acceptance Confirmation
DAG	Diverse Additional Generator
DCH	Direct Containment Heating
DDI	Direct Debris Interaction
DF	Decontamination Factor
DSP	Dryer Separator Pool
DW	Drywell
DWC	Drywell Cooling
EA	Environment Agency
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOP	Emergency Operating Procedure
EPDM	Ethylene Propylene Diene
EPRI	Electric Power Research Institute

FCI	Fuel-Coolant Interaction
FCS	Flammability Control System
FCVS	Filtered Containment Venting System
FLSR	Flooder System of Reactor Building
FLSS	Flooder System of Specific Safety Facility
FMCRD	Fine Motion Control Rod Drive
FPS	Fire Protection System
FPC	Fuel Pool Cooling and Clean-up
GDA	Generic Design Assessment
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
HPCF	High Pressure Core Flooder
HPME	High Pressure Melt Ejection
HPIN	High Pressure Nitrogen Gas Supply System
HVAC	Heating Ventilation and Air Conditioning
HWBS	Hard-Wired Backup System
IAEA	The International Atomic Energy Agency
IVR	In-Vessel Retention
JNES	Japanese Nuclear Energy Safety Organisation
LCO	Limiting Conditions of Operation
LDF	Lower Drywell Flooder
LDW	Lower Drywell
LPFL	Low Pressure Core Flooder
LOCA	Loss of Coolant Accident
LUHS	Loss (of) Ultimate Heat Sink
MAAP	Modular Accident Analysis Program
MCCI	Molten Core Concrete Interactions
MCR	Main Control Room
MDEP	Multi-national Design Evaluation Programme
MSTR	Main Steam Tunnel Room
MUWC	Make Up Water Condensate
NRW	Natural Resources Wales
OECD-NEA	Organisation for Economic Co-operation and Development Nuclear Energy Agency
ONR	Office for Nuclear Regulation
PAR	Passive Autocatalytic Recombiner
PCSR	Pre-construction Safety Report
PCV	Primary Containment Vessel
Pd	Design Pressure

POS	Plant Operating State
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
R/A	Reactor Area
R/B	Reactor Building
RCIC	Reactor Core Isolation Cooling
RCCV	Reinforced Concrete Containment Vessel
RCW	Reactor Building Cooling Water System
RDCF	Remote Depressurisation Control Facility
RGP	Relevant Good Practice
RHR	Residual Heat Removal
RI	Regulatory Issue
RIP	Reactor Internal Pump
RMI	Reflective Metallic Insulation
RO	Regulatory Observation
ROAAM	Risk Oriented Accident Analysis Methodology
RP	Requesting Party
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
RSW	Reactor Building Service Water
RUHS	Reserve Ultimate Heat Sink
SA C&I	Severe Accident Control & Instrumentation
SAPs	Safety Assessment Principles
SAMG	Severe Accident Management Guideline
SBO	Station Blackout
SFAIRP	So Far As Is Reasonably Practicable
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SLCS	Standby Liquid Control System
SoDA	Statement of Design Acceptability
S/P	Suppression Pool
SPCU	Suppression Pool Clean-up
SRV	Safety Relief Valve
SSC	System, Structure (and) Component
SSLC	Safety System Logic and Control
SUPRA	Suppression Pool Retention Analysis Computer Code
TAF	Top (of) Active Fuel

TAG	Technical Assessment Guide
TB	Accident sequence – station blackout
TESG	(MDEP) Technical Expert Sub-Group
TQUV	Accident sequence - transient with loss of feedwater, low pressure and high pressure injection
TSC	Technical Support Contractor
UDW	Upper Drywell
UK	United Kingdom
UK EPR	UK European Pressurised Reactor
US NRC	United States (of America) Nuclear Regulatory Commission
UK ABWR	United Kingdom Advanced Boiling Water Reactor
V/B	Vacuum Breaker
WENRA	Western European Nuclear Regulators' Association
WW	Wetwell

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Annexes

1 INTRODUCTION

1. This assessment report presents my Step 4 Generic Design Assessment (GDA) of Hitachi-GE's UK ABWR reactor design in the Severe Accident area.

1.1 GDA Background

2. Information on the GDA process is provided in a series of documents published on ONR's website (**http://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) from ONR and a Statement of Design Acceptability (SoDA) from 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-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. The objective of the Step 4 assessments is to undertake 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 of 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' (**http://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) and whether the proposed design reduces risks to as low as is 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; and
- 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 intended assessment strategy for GDA Step 4 in the severe accidents area was set out in an assessment plan (Ref. 1).

8. The objective of this Step 4 severe accidents assessment for the UK ABWR is to conduct an in-depth assessment of the severe accidents safety case presented by Hitachi-GE.

9. For the purposes of my assessment I have considered the definition of a severe accident from ONR's SAPs (Ref. 2). Severe accidents are defined as "those fault sequences that could lead either to consequences exceeding the highest off-site radiological doses given in the Basic Safety Level (BSL) of Numerical Target 4 (i.e. 100 mSv, conservatively assessed) or to an unintended relocation of a substantial quantity of radioactive material within the facility which places a demand on the integrity of the remaining physical barriers".

10. The UK ABWR is designed to provide protection against a range faults within the design basis. The protection is intended to ensure that Numerical Target 4 is met for all design basis faults by prevent significant degradation of fuel in the reactor core, or the spent fuel pool (SFP), thus preventing escalation to a severe accident. Hitachi-GE has also carried out analysis of beyond design basis faults with the aim of showing that scenarios, or combinations of events, just outside of the design basis do not escalate to a severe accident. Consideration of design basis faults is covered in ONR's Fault Studies assessment report (Ref. 3).

11. The focus of this assessment is on situations where the design basis protection is assumed to have failed, or the initiating event or sequence is more severe than assumed in the design basis. In these cases the faults could progress to states involving fuel damage. My assessment considers how the accident will be managed and/or mitigated by severe accident measures.

12. Inevitably there is significant overlap between the analysis of severe accidents and the probabilistic safety assessment (PSA), particularly the Level 2 PSA, where there is shared interest in understanding how accidents progress and how progression can be prevented or mitigated. The PSA for the UK ABWR considers a range of severe accident events, including those situations where significant radioactive releases are prevented or limited by design features. These events are of interest to my assessment because they define the assumptions made about the performance of severe accident measures, which need to be substantiated in the severe accident analysis.

13. The PSA also considers, probabilistically, those events where design provisions to prevent or mitigate severe accidents fail or are ineffective, potentially resulting in large releases to the environment. It is the role of the PSA to consider the overall risk profile for the UK ABWR, taking into account the full spectrum of events. Assessment of Hitachi-GE's probabilistic safety case is covered in ONR's PSA assessment report (Ref. 4).

14. In the earlier assessments for Steps 2 and 3 of GDA, the underpinning safety claims and arguments were assessed (Refs.5, 6). The Step 4 assessment has built upon those earlier assessments, looking in greater detail at the evidence that supports claims and arguments made by Hitachi-GE. This has involved review of documentation that:

- demonstrates that the relevant severe accident phenomena have been identified and that computer codes used to model the phenomena are suitable;
- describes the success criteria, in particular in relation to the primary containment, used to determine the effectiveness of severe accident measures;
- summarises the results of severe accident analysis, describes the progression of accidents and demonstrates the effectiveness of accident management measures and strategies;
- from a severe accidents perspective, demonstrates that the design leads to risks which are as low as reasonably practicable (ALARP); and

- presents arguments that large and early releases have been ‘practically eliminated’*.

15. A further key area of interest has been Hitachi-GE’s response to the international learning from the Fukushima Dai-ichi events. This has involved assessment of Hitachi-GE’s case that additional measures have been incorporated into the design of the UK ABWR to enhance plant resilience to beyond design basis events.

16. My assessment focuses on the fundamental features of the design, and the accident management concepts, that need to be established as part of GDA to confirm that the design has adequately addressed severe accidents. I have not set out to request, or to assess, detailed emergency preparedness arrangements or accident management procedures and guidelines; these can only be determined by a future licensee for a specific site.

17. In addition to the technical information contained within the severe accident submissions, my assessment has also considered the adequacy with which the documents are linked together to form a coherent safety case, and how they interface with and support the safety case documentation in other technical areas. Hitachi-GE’s top-level report which summaries the totality of its safety case for the UK ABWR, and ties all the different topic areas together, is the generic pre-construction safety report (PCSR).

1.3 Method

18. My assessment complies with internal guidance on the mechanics of assessment within ONR (Ref. 7)

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* *In this context, the possibility of certain conditions occurring is considered to have been practically eliminated if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise (from IAEA NS-G-1.10)*

2 ASSESSMENT STRATEGY

2.1 Standards and criteria

19. The Safety Assessment Principles (SAPs) (Ref. 2) constitute the regulatory principles against which duty-holders' safety cases are judged, and therefore are the basis for ONR's nuclear safety assessments, including the assessment detailed in this report. The SAPs are supplemented by Technical Assessment Guides (TAGs) which provide additional advice to ONR inspectors on assessing safety case submissions.

2.1.1 Safety Assessment Principles

20. The following are the key SAPs that have informed my interactions with Hitachi-GE during GDA and the assessment presented in this report:

- Severe accidents: FA.15, FA.16, FA.25
- Control and instrumentation of safety-related systems: ESR.1, ESR.7
- Containment: ECV.2, ECV.3;
- Computer codes and calculation methods: AV.1 to AV.8;
- Planning and preparedness: AM.1;
- Numerical Targets (principally Target 9).

2.1.2 Technical Assessment Guides

21. Technical Assessment Guides provide additional guidance to ONR inspectors on the interpretation and application of the SAPs. The following TAGs have informed this severe accident assessment of Hitachi-GE submissions against the SAPs above:

- NS-TAST-GD-005 'Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable)';
- NS-TAST-GD-007 'Severe Accident Analysis'
- NS-TAST-GD-042 'Validation of Computer Codes and Calculation Methods'.

22. In addition to the underlying technical merit of Hitachi-GE's UK ABWR design and analysis, I have considered the adequacy with which the supplied documentation is aggregated together as a safety case. TAG NS-TAST-GD-051(Ref. 8) sets out some key expectations for safety cases against which I have benchmarked Hitachi-GE's submissions:

- All references and supporting information should be identified and be easily accessible.
- There should be a clear trail from claims through the arguments to the evidence that fully supports the conclusions, together with commitments to any future actions.
- A safety case should accurately represent the current status of the facility in all physical, operational and managerial aspects.
- For new facilities or modifications, the safety case should accurately represent the design intent.
- There should be reference from the safety case to important supporting work, such as engineering substantiation. The safety case should be able to act as an entry point for accessing all relevant supporting information on which it is built.

2.1.3 National and international standards and guidance

23. Standards issued by the International Atomic Energy Agency (IAEA) (Ref. 9) and guidance from the Western European Nuclear Regulators Association (WENRA) (Ref. 10) are relevant to the severe accidents assessment of the UK ABWR. The latest

version of the SAPs (Ref. 2) was benchmarked against the extant IAEA and WENRA guidance in 2014, including the specific SAPs identified above for this severe accidents assessment. The general approach adopted in this report has been to assess Hitachi-GE's submissions against the SAPs, and as a result it can be inferred that international guidance is met.

24. There are specific provisions in the WENRA guidance that I will refer to in my assessment. For new reactors, WENRA Objective O3 (Ref. 10) on 'Accidents with Core Melt' is particularly relevant to severe accidents. It sets the expectation that 'large or early' releases are practically eliminated. WENRA has provided further guidance on this Objective, in particular:

- Position 4: Provisions to mitigate core melt and radiological consequences
- Position 5: Practical elimination

25. I have also taken into account new IAEA documentation produced since the SAPs were issued. The IAEA safety requirements (Ref. 9) for the design of nuclear plants set out expectations for the analysis of design extension conditions, which includes severe accidents. The expectation is that the plant shall be designed so that it can be brought into a controlled state and the containment function can be maintained, with the result that the possibility of plant states arising that could lead to an early radioactive release or a large radioactive release is practically eliminated.

26. In line with the international guidance, ONR's SAPs also include an expectation that potential severe accident states have been 'practically eliminated'. To demonstrate practical elimination, the safety case should show either that it is physically impossible for the accident state to occur or that design provisions mean that the state can be considered to be extremely unlikely with a high degree of confidence.

27. The UK, as a Contracting Party to the Convention on Nuclear Safety, also has obligations under the Vienna Declaration on Nuclear Safety (Ref. 11) with respect to the design of new nuclear power plants. These include obligations under the first principle which require that new nuclear power plants are designed consistent with the objectives of preventing accidents. If an accident should occur, the Vienna Declaration requires mitigation of releases of radionuclides causing long-term off site contamination, and avoidance of large or early radioactive releases. The UK's position is that demonstration of practical elimination meets the first principle of the Vienna Declaration (Ref. 12). Future UK reports to the Convention will need to demonstrate how the principle is being applied for new reactors, including the UK ABWR.

2.1.4 Fukushima Dai-ichi learning

28. The 2014 revision of the SAPs was prompted by publication of the Chief Nuclear Inspector's report (Ref. 13) on the implications of the Fukushima Dai-ichi accident for the UK nuclear industry. The 2014 revision also takes into account the revised WENRA reference levels (Ref. 10) and learning from Fukushima Dai-ichi.

29. WENRA has also provided further guidance on Objective O4 (defence-in-depth) for new nuclear plants to address lessons learnt from the Fukushima Dai-ichi accident. This is also relevant to my assessment.

30. Although assessment against the SAPs should capture the key elements of Fukushima Dai-ichi learning, I have also considered Hitachi-GE's submissions against the specific UK and international learning from:

- HM Chief Inspector of Nuclear Installations - Interim Report (Ref. 14);
- HM Chief Inspector of Nuclear Installations - Final Report (Ref. 13);

- European Council "Stress Tests" for UK nuclear power plants - National Final Report (Ref. 15); and
- The Fukushima Dai-ichi Accident Report by the IAEA Director General (Ref. 16).

31. It should be noted that my assessment against the Fukushima Dai-ichi learning has been restricted to those matters which are relevant to a Requesting Party; matters which would fall under the responsibility of a future licensee, or national agencies and bodies, do not form part of my assessment for GDA.

2.2 Use of Technical Support Contractors

32. It is usual in GDA for ONR to use Technical Support Contractors (TSCs) to enable access to independent advice and experience, analysis techniques and models.

33. During Steps 2 and 3, Gesellschaft für Anlagen und Reaktorsicherheit (GRS) carried out three reviews which have informed my assessment activities in Step 4:

- Summary of International Good Practice and International Requirements on Severe Accident Analyses (Ref. 17)
- Review of Hitachi-GE's Topic Report Regarding Severe Accident Analyses (Ref. 18)
- Best Practices for Independent Confirmatory Severe Accident Analyses for ABWR Plant Design (Ref. 19)

34. To supplement ONR's internal capability, a contract was placed with GRS for a fault studies/severe accidents specialist to work as an integral part of GDA Step 4 assessment team under my supervision.

35. AMEC Foster Wheeler provided independent confirmatory analysis of selected severe accident sequences using the MELCOR model. The output from this work (Ref. 20) provided advice to ONR on the adequacy of specific aspects of Hitachi-GE's UK ABWR severe accident analysis.

36. AMEC Foster Wheeler also carried out a review of Hitachi-GE's submissions on containment performance. This review (Ref. 21) was principally to support the assessment of a Regulatory Observation raised in the PSA area, however the findings of this review have also informed my assessment of the robustness of the containment to severe accident challenges.

2.3 Integration with other assessment topics

37. 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. There are interfaces with a number of topics, in particular:

- Fault Studies has considered protection requirements for design basis and beyond design basis faults. It is failure of this protection that could potentially result in progression of a fault to a severe accident. An understanding of the initiating faults and protection systems is essential to understand how a severe accident might occur. There has also been close cooperation with Fault Studies on Hitachi-GE's proposals for managing steam generated from the spent fuel pool (SFP) during design basis faults and the related arrangements for managing hydrogen if the fault develops into a severe accident. More generally, I have considered the effectiveness of Hitachi-GE's hydrogen management measures for both severe accidents and design basis faults.

- PSA and severe accident areas are complimentary and this has been reflected in my interactions with the PSA assessor. Hitachi-GE has used Level 1 PSA to identify the representative plant damage states that have then been considered in the severe accident analysis. Selection of accident sequences has been considered by the PSA assessor, as have matters relating to selection of representative release categories and source terms. There has also been common interest in containment performance analysis in relation to determining the Level 2 PSA release categories and the demonstration that the UK ABWR containment is robust against severe accident challenges.
- Level 3 PSA provides analysis of radiological consequences and conditional probabilities for severe accident sequences. Results from the Level 3 PSA provide risks for comparison with ONR's Numerical Targets. The PSA outputs inform ONR's judgements on the adequacy of severe accident measures.
- The Reactor Chemistry assessor has taken the lead on chemistry matters relating to accident phenomena and the performance of severe accident measures.
- The requirements for severe accident C&I has been jointly considered with the C&I assessor.
- The engineering substantiation of Hitachi-GE's severe accident mechanical systems has been covered by the Mechanical Engineering assessor.

38. I have provided input to assessment of PCSR Chapter 22: Emergency Preparedness (Ref. 22) on matters relating to severe accident measures and procedures insofar as they are defined for the generic site.

39. I have collaborated with other disciplines to assess Hitachi-GE's response to learning from the Fukushima Dai-ichi accident.

2.4 Out of scope items

40. In GDA I have not considered potential severe accidents involving fuel route cask handling operations within the reactor building and all operations once the spent fuel has left the reactor building.

41. Any severe accidents that are specific to a two-unit site have been excluded from this assessment.

42. A detailed assessment of technical specifications, operating /emergency procedures and accident management arrangements is generally out of scope for GDA. However, in the severe accidents area I have considered Hitachi-GE's accident management concepts based on preliminary procedures and guidelines provided to ONR for the purposes of GDA.

43. The detailed design of the back-up building (B/B), included the location, layout and arrangement, is considered to be site-specific and therefore out of scope. However, the severe accident functional requirements for the B/B is considered in my assessment.

44. The arrangement of mobile equipment and associated water sources is considered to be out of scope. The minimum functional requirements of the mobile equipment are however considered.

3 REQUESTING PARTY'S SAFETY CASE

3.1 Safety case documentation and structure

45. Hitachi-GE has identified the Generic Pre-Construction Safety Report (Generic PCSR) as the key documentation and submission that outlines its top level claim that the "UK ABWR constructed on a generic site within the United Kingdom (UK), can be operated safely under all operating and fault conditions." (Ref. 23).

46. Chapter 26 of the PCSR (Ref. 24) covers Severe Accident Analysis. Chapter 26 also covers beyond design basis analysis (BDBA) – this aspect is considered as part of the Fault Studies assessment (Ref. 3).

47. There is, inevitably, a large amount of information in other PCSR chapters which is linked to Chapter 26. Notably this includes (but is not limited to):

- Chapter 5: General Design Aspects (Ref. 25)
- Chapter 9: General Description of the Unit (Facility) (Ref. 26)
- Chapter 10: Civil Works and Structures (Ref. 27)
- Chapter 13: Engineered Safety Features (Ref. 28)
- Chapter 14: Control and Instrumentation (Ref. 29)
- Chapter 16: Auxiliary Systems (Ref. 30)
- Chapter 22: Emergency Preparedness (Ref. 22)
- Chapter 23: Reactor Chemistry (Ref. 31)
- Chapter 24: Design Basis Analysis (Ref. 32)
- Chapter 25: Probabilistic Safety Assessment (Ref. 33)
- Chapter 28: ALARP Evaluation (Ref. 34)

48. While the PCSR is clearly a vital and fundamental part of the UK ABWR safety case, it is only providing a summary of lower level safety case documents. Sitting beneath the PCSR (and referenced from it) are a large number of Topic Reports and Basis of Safety Case Reports. It is these references (and supporting references from these reports) which have been the main area for assessment during GDA Step 4 and provide the technical basis for most of the regulatory judgements on the UK ABWR, more so than the PCSR.

49. For severe accidents, the following reports have been central to my assessment

- Topic Report on Severe Accident Phenomena and Severe Accident Analysis (Ref. 35)
- Severe Accident Safety Case for Shutdown Reactor and SFP (Ref. 36)
- Demonstration of Practical Elimination of Early or Large Fission Product Release for UK ABWR (Ref. 37)
- Basic Requirement Specification of Severe Accident Management Measures (Ref. 38)
- Basis of Safety Cases on Severe Accident Mechanical Systems (Ref. 39)
- Containment Venting Strategy in UK ABWR (Ref. 40)
- An ALARP Evaluation on Methods/Technologies for the Mitigation of Molten Core Concrete Interactions for the UK ABWR (Ref. 41)
- Flammable Gas control and Supporting Analysis in UK ABWR (Ref. 42)
- ALARP Discussion on Flammable Gas Control (Ref. 43)
- Containment Performance Analysis Report in UK ABWR (Ref. 44)
- Applicability of the HM Chief Inspector's Recommendations and ONR's Stress Test Findings to UK ABWR Design (Ref. 45)
- Accident Management Guideline (After Core Damaged) for UK ABWR (Ref. 46)
- Consideration of Fuel Coolant Interactions for UK ABWR (Ref. 47)

50. Ref. 35 is Hitachi-GE's principal submission for severe accidents. It is also the repository of Hitachi-GE's severe accident analysis for the reactor at power.

51. A number of other reports have been referenced by Hitachi-GE and submitted to ONR in the course of GDA Step 4. These have been referenced as appropriate in Section 4 of this assessment report.

3.2 Safety case submissions addressing regulatory observations

52. During Step 3 of GDA, I identified significant gaps in Hitachi-GE's severe accident safety case. I raised a Regulatory Observation (RO): RO-ABWR-0023 (Ref. 48), setting out my expectations for the additional work required to address the shortfalls. The majority of submissions received during Step 4, and covered in my assessment in Section 4 of this report, collectively support Hitachi-GE's response to this RO.

53. I also raised RO-ABWR-0039 (Ref. 49) in GDA Step 3. This required Hitachi-GE to address international learning from the accident at Fukushima Dai-ichi. Hitachi-GE provided a principal submission (Ref. 45) which considered each individual learning point, with reference provided as required to other supporting safety case documents. I consider this further in Section 4.8 of my report.

54. RO-ABWR-0046 (Ref. 50) was raised by ONR in response to identified shortfalls in Hitachi-GE's evidence on the identification of primary containment failure modes and the limiting failure envelope. This was raised by the PSA inspector, but this is also relevant to my assessment. Hitachi-GE's principal submission in response to this RO is Ref. 44, which is a key supporting document to PCSR Chapter 26.

3.3 Key design features of the UK ABWR

55. The UK ABWR design is described in detail across multiple chapters of the PCSR and this is not repeated here. However, there are some key features relevant to severe accidents which are highlighted as background for information and convenience.

3.3.1 Reactor building

56. The reactor building (R/B) houses the reactor pressure vessel (RPV), the reinforced concrete containment vessel (RCCV) that constitutes the primary containment vessel (PCV), major portions of the reactor steam supply system, parts of the steam tunnel, the refuelling area, emergency core cooling systems (ECCS), heating ventilation and air conditioning (HVAC) systems and additional supporting systems.

57. The SFP is located adjacent to the RCCV inside the secondary containment of the R/B, but outside and above the RCCV. In order to access the reactor core for refuelling, it is necessary to open the PCV and the RPV and flood the reactor well and the dryer separator pool (DSP) up to the SFP water level. Then, the pools can be connected by removing the SFP gate.

3.3.2 Backup building

58. The UK ABWR has a backup building (B/B), remote from the R/B and main control room (MCR), which provides a means of delivering functions to support severe accident management. The B/B provides an alternative location for operating severe accident systems and monitoring accident conditions within the plant. The B/B systems are powered by two air-cooled diesel generators located in the B/B, which are independent from the emergency diesel generators (EDGs) used to support fundamental safety functions for design basis accidents.

3.3.3 Containment

59. A simplified representation of the UK ABWR containment is shown in Figure 1.

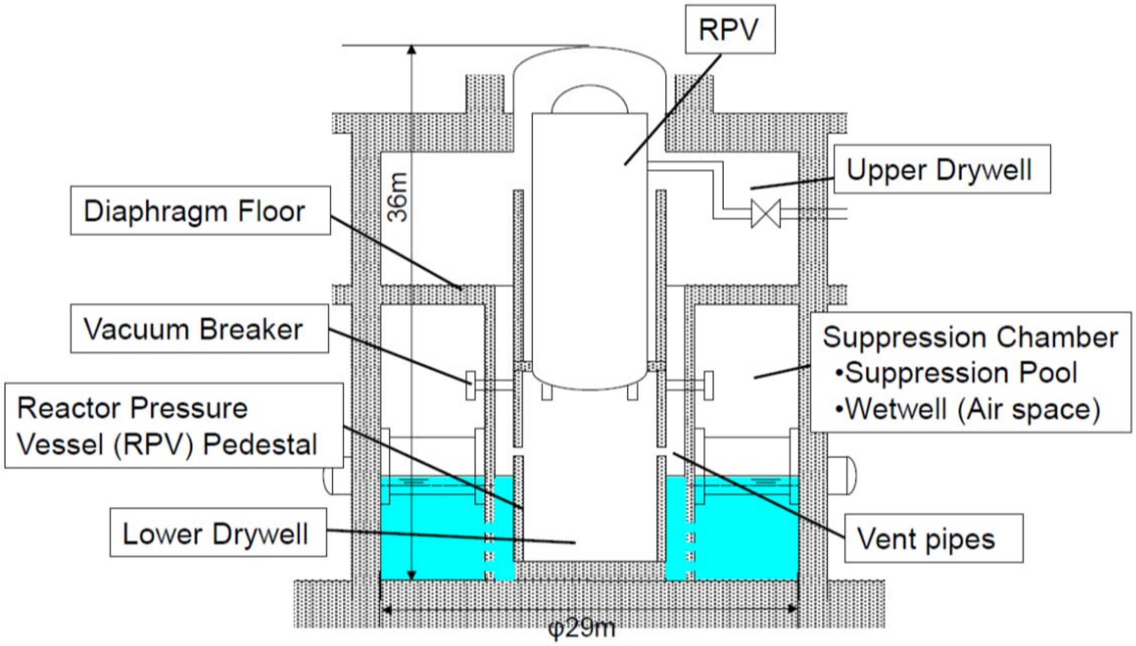

Figure 1: Cross-Section of the UK ABWR RCCV

60. The PCV is provided by the RCCV, which incorporates a leak-tight steel liner, and the PCV head. The containment has a capability for rapid closure or isolation of pipes and ducts that penetrate the PCV boundary in order to maintain leak tightness. The PCV is designed to remain leak tight (within design limits) where severe accident measures operate as designed to mitigate the effects of an accident. Under certain severe accident conditions the containment pressure may need to be reduced to maintain PCV integrity; this is achieved by venting to atmosphere. The PCV design pressure (Pd) is 310 kPa (gauge), which I refer to in this report as 1×Pd. In severe accidents the ultimate failure pressure of the PCV is claimed to be in excess of 2×Pd.

61. The PCV is separated into a drywell (DW) and wetwell (WW). The DW and WW are separated by the diaphragm floor and pedestal wall, the latter providing support to the RPV. The DW, which houses the RPV and associated pipework, consists of upper and lower air volumes connected by vertical drywell connecting vents. The WW comprises a single volume with a gas space and a water suppression pool (S/P). The S/P is used to rapidly condense steam during controlled RPV depressurisation or relief via the safety relief valves (SRVs). Steam from the DW, either from a LOCA break inside the DW, or arising from a failure of the RPV in a severe accident, is also routed to the S/P through horizontal vent pipes.

62. Vacuum breakers (V/Bs) connect the lower drywell (LDW) to the WW gas space to prevent an excess pressure in the WW relative to the DW. The (eight) V/Bs open and close passively, with their position indicated in the Main Control Room (MCR). The V/Bs are located at the end of a pipe through the pedestal wall, well within the WW gas space.

63. Under accident conditions containment sprays in the WW airspace and upper DW provide a means for temperature control and also provide scrubbing of fission products from the PCV atmosphere. The S/P also provides scrubbing of gases and steam which pass from the DW to the WW via the vent pipes and also from the RPV via the SRVs.

In accident conditions the suppression pool (S/P) pH will be controlled to maximise the effectiveness of scrubbing and limit the release of iodine.

64. The PCV head at the top of the DW is secured to the primary containment by a bolted flange. The head is removed during shutdown to permit access to the RPV for refuelling/defuelling. During a severe accident the reactor well above the PCV head can be flooded to minimise the possibility of over-temperature failure of the head and provide additional scrubbing if a release from this location were to occur.

65. The main component of the secondary containment is the R/B reinforced concrete building structure that forms the external weather envelope. The secondary containment boundary encloses the RCCV primary containment above the concrete basemat. The secondary containment encloses all penetrations through the PCV and all those systems external to the PCV that may become a potential source of radioactive release after an accident, including the main steam tunnel.

66. The secondary containment is not designed to provide a leak-tight barrier. Under normal conditions the secondary containment is maintained at a slightly negative pressure by the reactor area HVAC system. Following a release of radioactivity into the secondary containment, the standby gas treatment system (SGTS) is designed to provide this function. The SGTS comprises two trains which exhaust to the stack through filters.

3.3.4 The reactor

67. The reactor core is moderated and cooled by un-borated water which is allowed to boil in the core during normal operation. A separator above the core outlet directs water to the core shroud where is it mixed with feedwater and circulated upwards through the core by ten reactor internal pumps (RIPs). Steam from the separator is dried and then passes from the RPV to the turbines through four steam lines.

68. The reactor core consists of 872 fuel assemblies and 205 control rods. Control rods enter from underneath of the core, connected to fine motion control rod drive (FMCRDs) through penetrations in the bottom of the RPV. Individual fuel assemblies comprise a 10 by 10 grid of fuel pins housed within a zirconium channel box which provides partial separation of flows between adjacent assemblies. Fuel pins comprise uranium dioxide pellets with zirconium cladding. Fuel assemblies are supported above the lower plenum by the core support plate.

69. Reactor protection is delivered by the Class 1 safety system logic and control system (SSLC) which hydraulically inserts control rods to scram the reactor. Alternative systems are provided either to insert the control rods using alternative hydraulic actuation, or by injecting boron into the core using the standby liquid control system (SLCS). These alternative systems are delivered by the diverse hard-wired backup system (HWBS). The SLCS injects a neutron absorbing solution of sodium penta-borate into the core to provide sufficient negative reactivity to shut down the reactor in a safe manner from full power operation to cold shutdown conditions if control rod insertion is not achieved.

3.3.5 Emergency core cooling

70. The emergency core cooling system (ECCS) provides the primary means for fuel cooling for design basis faults. The ECCS consists of three independent divisions, each with functions for high pressure and low pressure water injection into the RPV in the event of a reactor fault. For both of Divisions II and III of the ECCS high pressure injection is provided by the high pressure core flooder (HPCF) system. In Division I of the ECCS, the high pressure water injection function is provided by the reactor core isolation cooling system (RCIC). For each of the three divisions, low pressure injection

is provided by the low pressure flooder system (LPFL). Each division of the ECCS can be powered by one EDG.

71. The purpose the HPCF is to inject water into the core when the RPV is in a pressurised state, although it can also operate at low pressures. The automatic depressurisation system (ADS) provides depressurisation of the RPV by actuation of the SRVs. This allows the plant to achieve a safe, stable low pressure state. The LPFL provides injection at relatively low pressures for cases where the RPV is in a depressurised state, for example due to a large break or where the RPV has been depressurised by the ADS. The ECCS also provides residual heat removal (RHR) functions for the RPV and can cool the S/P. Two trains can also be set up for S/P cooling during faults.

72. All ECCS modes are designed to deliver adequate core cooling, ensuring that the water level in the core is maintained at an appropriate level above top of active fuel (TAF). Coolant for injection is drawn from either the condensate storage tank (CST), external to the R/B, or the S/P.

73. The RCIC uses a turbine-integrated pump driven by decay heat steam to inject water into the RPV and maintain water level. The RCIC operates automatically to maintain RPV water level and requires only battery power (which lasts for up to 24 hours). Exhaust steam from the RCIC is condensed in the S/P, leading to a rise in PCV pressure and temperature if RHR functions are unavailable for S/P cooling in a beyond design basis event.

3.3.6 Severe accident in-vessel cooling

74. A severe accident giving rise to significant core damage would normally develop as a result of failure or degradation of the ECCS functions. In this case, an alternative active low pressure injection system, the flooder system of specific safety facility (FLSS), is provided to prevent and/or mitigate fuel damage by providing post-scram fuel cooling. In the event of core melt and relocation to the core support plate, the FLSS is designed to arrest further melt progression.

75. Operation of the FLSS requires that the RPV is depressurised to a low pressure state by the operators, either using the ADS or the remote depressurisation control facility (RDCF). The RDCF, which is controlled by the HWBS, is diverse from the ADS and can be operated from either the MCR or the B/B.

76. The FLSS consists of two trains of two pumps with a dedicated water source, individual piping and the necessary valves. Instrumentation and control is through the HWBS. The FLSS is also designed to provide cooling water to the PCV spray header, the LDW, the reactor well (for cooling the PCV head and flange) and the SFP. The FLSS pumps are located in the B/B and can be operated either from the MCR or the B/B. On-site water storage and fuel supplies are provided for seven day operation without external supplies.

77. The flooder system of reactor building (FLSR) is a mobile system which replicates the FLSS injection functions. It uses a mobile pump and power truck which would be normally stored on-site and has specific connections to FLSS injection piping. In a severe accident, the FLSR could potentially be deployed in about 8 hours and if necessary could provide cooling following termination of the RCIC, or otherwise if the FLSS was unavailable. On-site water storage and fuel supplies for the FLSR are provided for seven day operation without external supplies.

78. Hitachi-GE has also identified the continued operation of the control rod drive (CRD) purge water pumps as a high pressure injection measure if available. The CRD purge water pumps take suction from the CST and are connected to the Class 1 emergency

power supply. The pumps are capable of injecting into the RPV against the full operational reactor pressure.

79. Hitachi-GE has identified further possibilities for low pressure RPV injection from Class 3 systems, including from the fire protection system (FPS) and the make-up water condensate (MUWC) system.

80. In the absence of in-vessel cooling, the UK ABWR does not include any design provision for in-vessel retention (IVR) of core debris. Instead, the strategy for the UK ABWR is to manage core debris in the LDW of the containment. Operators would attempt to manually depressurise the reactor before RPV failure using the ADS or RDCF. The objective is to avoid high pressure melt ejection (HPME) and therefore mitigate challenges from direct containment heating (DCH) and rapid steam generation from fuel-coolant interaction (FCI).

3.3.7 Severe accident ex-vessel cooling

81. RPV failure would result in a release of a molten mixture of fuel, zirconium, steel and control rod material into the primary containment; this material is known as corium. The concrete floor of the LDW is designed as a spreading area for corium released from the RPV. The intention is that the size of the spreading area is sufficient to allow the corium to be cooled by overlying water, thereby minimising molten core concrete interactions (MCCI). The LDW floor includes a concrete layer of at least 1 metre thickness, constructed of basaltic concrete to minimise generation of non-condensable gases formed by MCCI. This is intended to prevent contact of corium with the PCV liner.

82. The severe accident strategy for the UK ABWR is to pre-flood the LDW if RPV failure is considered likely. This is intended to be achieved by manual activation of water injection into the LDW using the FLSS or FLSR. If injection is unavailable, then the Lower Drywell Flooder (LDF), comprising ten fusible (thermally actuated) plug valves, activate passively to flood the LDW with water from the S/P. The design intent is that containment structures and components are designed to withstand the effects of steam explosions associated with ex-vessel FCI.

3.3.8 Cooling for shutdown and spent fuel pool severe accidents

83. The FLSS and FLSR are designed to provide fuel cooling for accidents involving the reactor at shutdown or the SFP. There are also a number of other systems which are able to provide water make-up for fuel cooling in a severe accident, although these are not designated as severe accident systems in the PCSR. The following systems can potentially provide low pressure make-up / injection (subject to the availability of power):

- fuel pool cooling and clean-up (FPC) system;
- MUWC system;
- suppression pool clean-up (SPCU) system;
- FPS for injection into the RPV.

3.3.9 Containment heat removal

84. In a severe accident, heat would accumulate in the S/P by release of steam through the SRVs. Alternatively, if the RPV has failed then steam would be generated in the LDW from water overlying the corium; this would also result in heat-up of the S/P due to transfer of steam through the vent pipes.

85. The RHR functions of the ECCS include the S/P cooling mode which is able to provide long term heat removal from the containment in severe accidents. Heat from the RHR

is rejected to the closed loop reactor building cooling water (RCW) system which itself rejects heat to the reactor building service water (RSW) system in the heat exchanger building. The RSW takes its water from a water intake pit, passes it through the RCW heat exchangers and then discharges it to a water discharge pit. A conceptual design for a reserve ultimate heat sink (RUHS) has also been proposed for GDA.

86. The RHR, RCW and RSW are active systems which require electrical power to, and availability of, one of the Class 1 distribution systems. Diverse power supplies, independent from the EDGs, are provided by a Diverse Additional Generator (DAG) and large mobile power truck; however both of these require one of the Class 1 distribution systems to be available to deliver ECCS functions. The mobile alternative heat exchange facility (AHEF) is also available to provide an alternative heat sink to the RHR (by-passing the RCW and RSW) if one ECCS division can be powered.

87. If the RHR function is unavailable, a severe accident will result in an increase in the energy stored in the PCV. In this case, containment heat removal and pressure control is achieved through venting of steam and gases from the PCV to atmosphere through the filtered containment venting system (FCVS). The preferred route is to vent from the WW as this has the benefit of fission product scrubbing by the S/P. The FCVS incorporates filters for the further removal of elemental iodine and particulates prior to discharge to atmosphere through the stack. A hardened, unfiltered venting route is also available if necessary, although venting through the FCVS would be the preferred option.

88. The FCVS also incorporates a containment overpressure protection system (COPS) intended to ensure that pressure is relieved from the PCV before the integrity of the containment is challenged. The COPS is a passive system which uses bursting disks to release steam and gases from the WW through the FCVS.

3.3.10 Hydrogen management

89. During at-power operation, the PCV of the UK ABWR is inerted with nitrogen to limit oxygen concentration to no greater than 4%. Suppressing the level of oxygen limits the potential for combustion of any flammable gases which might accumulate in the PCV in fault or accident conditions. This is a notable design feature of the UK ABWR relative to PWR designs which operate with an air-filled containment.

90. The UK ABWR also includes measures for mitigation of hydrogen in the primary and secondary containments. During GDA Hitachi-GE introduced a design change to the primary containment Flammability Control System (FCS) of the Reference Design, to replace active recombiners (which require electrical power) with passive autocatalytic recombiners (PARs). A PAR is a passive device that recombines hydrogen and oxygen gases into water using a metal catalyst. They are self-starting when gas concentrations exceed minimum levels and operate without the need for power.

91. The overall provision of flammable gas control measures includes:

- PARs are located in the PCV for design basis accidents to prevent the build-up of hydrogen and oxygen which could occur in faults and accidents due to radiolysis of water.
- PCV venting is used to manage hydrogen concentration in the PCV during severe accidents.
- Alternative Nitrogen Injection (ANI), a severe accident provision delivered by mobile equipment, is available post-venting to supporting re-inerting of the PCV
- PARs and the SGTS are used for PCV leakage into the R/B during reactor severe accidents.
- The R/B blowout panel and door are used for severe accidents involving the shutdown reactor or the SFP.

3.3.11 Categorisation and classification

92. Hitachi-GE's approach to categorisation and classification is described in Chapter 5 of the PCSR (Ref. 25). According to this approach, severe accident systems are categorised as Category B because they make a significant contribution to nuclear safety for beyond design basis accidents. They are either Class 2 or Class 3 depending on whether the system is the principal or secondary means of delivering the Category B function.

3.3.12 Accident management strategy

93. The strategy for managing a reactor severe accident is to provide fuel cooling using any available means of low pressure injection and, in the absence of RHR containment cooling, release steam from the containment by venting through a filtered route. For the UK ABWR, effective fuel cooling (even with boiling) can be delivered under low pressure conditions.

94. For the reactor during shutdown with the RPV head removed and for the SFP, the strategy is to provide low-pressure make-up to offset losses due to boil-off or from drain-down. Where necessary, any steam and hydrogen discharged into the R/B would be released to atmosphere to mitigate the risk associated with a hydrogen explosion.

95. A severe accident control and instrumentation (SA C&I) system is provided for control of severe accident systems and for monitoring of plant conditions using accident qualified equipment. The SA C&I system can be operated from the MCR or the B/B.

4 ONR STEP 4 ASSESSMENT

4.1 Overview of assessment strategy

96. Here I set out the overall strategy for my assessment of Hitachi-GE's severe accident safety case. I derive my assessment expectations from ONR SAPs, TAGs and relevant international guidance – see Section 2.1.

- **Section 4.2: Severe accident scenarios** – In this section I have assessed the objectives of Hitachi-GE's severe accident analysis and considered how this compliments Hitachi-GE's safety case for design basis and beyond design basis faults, and the PSA.
- **Section 4.3: Severe accident analysis codes** – Here I present my assessment of the overall suitability of the computer codes used by Hitachi-GE's to model severe accidents.
- **Section 4.4: Severe accident phenomena** – In this section I present my assessment of whether Hitachi-GE has identified the relevant severe accident phenomena and whether these phenomena have been suitably modelled using Hitachi-GE's analysis codes.
- **Section 4.5: Containment performance** – Hitachi-GE identifies that maintaining containment integrity is a key element of the severe accident safety case. In this section I present my assessment of whether Hitachi-GE has correctly identified and characterised challenges to the containment and whether suitable containment success criteria have been defined to support the severe accident analysis.
- **Section 4.6: Severe accident analysis** – In this section I have assessed Hitachi-GE's severe accident analysis to establish whether accident progression has been adequately modelled and whether the effectiveness of the UK ABWR severe accident measures has been demonstrated.
- **Section 4.7: Severe accident management** – In this section I have considered whether Hitachi-GE has identified suitable concepts that can be taken forward by the future licensee when developing site-specific procedures. I have also considered whether the specification for Hitachi-GE's SA C&I system supports the accident management concept.
- **Section 4.8: Lessons learnt from the Fukushima Dai-ichi accident** - I have considered Hitachi-GE's overall response to events at Fukushima Dai-ichi and how learning from this has been reflected generally in the design of the UK ABWR.
- **Section 4.9: Practical elimination** – In this section I present my assessment of Hitachi-GE's case that large and early releases have been practically eliminated. This has considered UK and international expectations on practical elimination for new reactors.
- **Section 4.10: Demonstration that risks are ALARP** – In this section I present my assessment of Hitachi-GE's claims that the provision of the UK ABWR severe accident measures contributes to an overall design which reduces risk ALARP.
- **Section 4.11: Safety case documentation** – In this section I present my assessment of the overall adequacy of the PCSR and supporting documentation against my expectations for a severe accidents safety case.
- **Section 4.12: Overseas regulatory interface** – In this section I explain how my participation in the MDEP UK ABWR Working Group has supported my assessment activities in GDA.

4.2 Severe accident scenarios

4.2.1 Background

97. Hitachi-GE has provided analysis of severe accident sequences to support both the Level 2 PSA and the severe accident safety case. Accident sequences for the following operating modes are analysed:

- reactor at power (Ref. 35);
- reactor during shutdown (Ref. 36); and
- SFP (Ref. 36)

98. The selection of accident sequences resulting in fuel damage is the output of a systematic Level 1 PSA approach. During Step 4 Hitachi-GE has refined and expanded the list of representative accident sequences in response to challenges from ONR's PSA assessor and these changes have been reflected in updates to Hitachi-GE's severe accident analysis. The final list of sequences includes those initiated by low frequency events outside of the design basis and those due to events within the design basis but where design basis measures fail.

99. For the reactor at power, Hitachi-GE has identified twenty-one representative plant damage states (Ref. 51) based on results from the Level 1 PSA. The plant damage states cover a range of sequences initiated by LOCA and intact circuit faults. The damage states include all Level 1 PSA outcomes, including sequences that result in early containment failure. The grouping of Level 1 PSA accident classes into plant damage states is based on the key parameters at the time of core damage that would influence the subsequent containment response and potential for radioactive release. The progression of the plant damage states has been analysed in the Level 2 PSA for the reactor at power to determine whether either:

- there are no challenges to the containment and no release of radioactivity;
- the containment remains intact but a release occurs due to venting; or
- the containment fails, resulting in an uncontrolled radioactive release.

100. Hitachi-GE has considered 'mitigated' and 'unmitigated' accident sequences and analysed these using the MAAP severe accident analysis code. Hitachi-GE states that unmitigated sequences for all plant damage states are used to determine available time margins, for example to RPV failure or containment failure. The sequences provide an input to Hitachi-GE's Level 2 PSA, which considers outcomes where severe accident measures function as intended and also those unlikely cases where the severe accident measures fail. For certain PSA reactor fault sequences, Hitachi-GE has only considered unmitigated accidents. For these cases, Hitachi-GE argues that additional analyses are not necessary as the containment either fails before core damage or, for the purposes of the PSA, it is assumed that the mitigation would not be effective in preventing containment failure. However, Hitachi-GE has included these sequences in its consideration of practical elimination (see Section 4.9).

101. For the mitigated cases, Hitachi-GE claims that the proposed severe accident measures would be effective in preventing accident escalation beyond the initial plant damage states and/or providing mitigation of consequences. Severe accident analysis has been presented to support this claim. Hitachi-GE implies that the selected mitigated cases that have been considered are sufficient for demonstrating the effectiveness of engineered severe accident features (for the relevant reactor sequences).

4.2.2 Assessment

102. I have considered Hitachi-GE's high level approach to severe accident analysis and how this fits in with the overall UK ABWR safety case. My objective is to consider the scope and use of Hitachi-GE's severe accident analysis against the expectations set out in FA.15, FA.16 and FA.25 of the SAPs. In addition to setting out expectations for severe accident analysis, these SAPs also cover how the analysis should complement the wider safety case, including DBA and PSA.

103. Hitachi-GE has provided DBA in PCSR Chapter 24 (Ref. 32) as part of its demonstration that high level safety functions for core cooling, heat removal and reactivity control provide protection of the plant against design basis faults. This analysis has been reviewed in the Fault Studies assessment report (Ref. 3). Failure of design basis measures could potentially result in progression of a fault to a severe accident. The Level 1 PSA has been used to identify sequences where design basis protection fails, or where design basis measures would be ineffective, leading to fuel damage states involving degradation of fuel and release of radioactivity. Hitachi-GE has carried out severe accident analysis for these damage states.

104. Hitachi-GE has also carried out beyond design basis analysis (BDBA) (Ref. 52) which sets out to demonstrate that the UK ABWR design is robust against a range of events just outside of the design basis. Hitachi-GE uses best-estimate approaches to show such events would not lead to a 'cliff-edge' that would result in significant fuel degradation. The assessment of Hitachi-GE's selection of beyond design basis faults and the BDBA is within the scope of the Fault Studies assessment report (Ref. 3) and so I do not comment on the detail here. However, for the purposes of GDA, I am satisfied that the use of BDBA to demonstrate the robustness of the plant against beyond design basis faults complements Hitachi-GE's consideration of severe accidents involving core melt. As such, I consider that Hitachi-GE's approach is consistent with the expectations in SAP FA.15 and FA.25 on the need to demonstrate the absence of cliff edge effects. I also consider that Hitachi-GE's use of BDBA alongside severe accident analysis is consistent with the consideration of design extension conditions as set out in Requirement 20 of IAEA SSR-2/1 (Ref. 9).

105. Hitachi-GE's initial submissions only considered severe accidents arising from faults initiated with the reactor at power. During Step 3 of GDA I challenged Hitachi-GE to provide a severe accident safety case for all operating modes, including the shutdown reactor and the SFP. I am also satisfied for the purposes of GDA that the scenarios selected for the reactor at shutdown and the SFP allow an adequate investigation and understanding of the relevant severe accident challenges and phenomena for these operating modes.

106. In my Step 4 Assessment Plan (Ref. 1) I stated that the selection of severe accident sequences would be considered as part of the scope of the PSA assessment (Ref. 4). The focus of the assessment in my report is on how the accident state or scenario can be controlled and/or mitigated. Hitachi-GE's analysis of mitigated cases is used to demonstrate the effectiveness of the proposed severe accident measures. I assess the details of Hitachi-GE's analysis in Section 4.6.

107. Hitachi-GE has identified a number plant damage states which result either in containment failure before core melt or immediately after. In these cases Hitachi-GE assumes, for the purposes of the PSA, that the severe accident measures would not prevent containment failure. Hitachi-GE has also analysed unmitigated accident sequences and included these in the PSA for situations where severe accident measures are assumed to fail. For these sequences, the focus is on prevention of the plant damage states by robust plant design and a demonstration that risks are reduced to ALARP by design features which prevent entry into the damage states. These unmitigated sequences are captured in the Level 3 PSA (Ref. 53) and the associated

risks to people are reflected in the overall risk profile for the UK ABWR. The probabilistic treatment of these sequences is considered in the PSA assessment report (Ref. 4).

108. Having considered Hitachi-GE's overall approach to severe accident analysis, I am satisfied that this meets my expectations for GDA, in particular:

- Hitachi-GE's severe accident analysis compliments the DBA and PSA, reflecting the expectations in FA.15 and FA.25 of the SAPs.
- Hitachi-GE's use of beyond design basis analysis and severe accident analysis is consistent with expectations in FA.15 and FA.25 SAPs on consideration of cliff-edge effects. Hitachi-GE's approach is also consistent with international expectations on consideration of design extension conditions.
- Hitachi-GE's selection of severe accident sequences for the reactor at power, shutdown and SFP is as a result of a systematic approach based on Level 1 and 2 PSA. This approach is consistent with the expectations of FA.15 of the SAPs.
- Hitachi-GE's severe accident analysis has been used to consider how accident states or scenarios will be controlled and/or mitigated, meeting the expectations of SAP FA.16.

4.3 Severe accident analysis codes

4.3.1 Background

109. Hitachi-GE has used a number of industry-standard computer codes for its severe accident analysis. Standard engineering packages have also been used to support Hitachi-GE's containment performance analysis. The relevant computer codes are:

- MAAP 4.07 is Hitachi-GE's main analysis tool for the thermal hydraulics analysis of reactor accident sequences.
- SUPRA is used to pre-calculate decontamination factors for the S/P for use in MAAP.
- MAAP 5.03 is used to model severe accidents in the SFP and for MCCI sensitivity analyses.
- GOTHIC 8.1 is used to model hydrogen behaviour in the R/B.
- JASMINE v.3 is used to determine the intensity of a steam explosion due to ex-vessel FCI.
- AUTODYN is used to model propagation of steam explosion pressure waves.
- ABAQUS is used for modelling of the RCCV structural and metal components in response to severe accident loads.
- STAR-CCM+ is used for Computational Fluid Dynamics (CFD) analysis of high pressure corium ejection into the LDW in support of containment performance analysis.

4.3.2 Assessment overview

110. In this section I focus on the adequacy of the overall verification and validation status of Hitachi-GE's severe accident codes.

111. My assessment of Hitachi-GE's modelling approach has been against SAPs AV.1 to AV.8. Hitachi-GE's methods for severe accident analysis are used both to support its severe accidents safety case and to underpin the PSA. As such I have coordinated my assessment of the adequacy of the models with ONR's PSA assessor, who has also considered the PSA-specific expectations in NS-TAST-GD-030 (Ref. 8).

112. Regarding AV.4 (quality assurance), I note that all of the computer models used by Hitachi-GE are developed and maintained by reputable external organisations and

most are in widespread use internationally. It is my judgement that the quality arrangements are likely to be commensurate with the expectations for beyond design basis analysis and I have chosen not to investigate quality assurance aspects further. Furthermore, I have chosen not to carry out any assessment of Hitachi-GE's procedures for the development, maintenance and application of datasets. I take assurance from the fact that Hitachi-GE's general approach has been considered by other ONR colleagues in relation to design basis analysis and found to be satisfactory (Ref. 3).

113. SAPs AV.7 (data collection through life) and AV.8 (update and review) will primarily be of relevance to the licensee. I would expect the licensee to learn from future developments in severe accident research and improved understanding of plant accidents, and to revise analysis for the UK ABWR as required after GDA.

4.3.3 MAAP

114. Hitachi-GE has produced a MAAP (Modular Accident Analysis Program) model for severe accident analysis of the UK ABWR reactor (Ref. 35). MAAP is a computer code licensed by the Electric Power Research Institute (EPRI). MAAP is a recognised code for severe accident analysis of light water reactors and has been subject to much international scrutiny. As MAAP 4.07 is the main calculation tool supporting Hitachi-GE's severe accidents and Level 2 PSA safety case, it has been subject to specific assessment by ONR.

115. MAAP user documentation is publicly available and has been supplemented by Hitachi-GE submissions on the MAAP physics models (Ref. 54) and phenomenological uncertainties (Ref. 55). Hitachi-GE claims that MAAP 4.07 is validated for BWR-type reactors (Ref. 54) and has cited several examples for the validation to show that MAAP is capable of simulating characteristics and severe accident phenomena of BWRs. Hitachi-GE reports in Ref. 54 that measurements from the accidents at the Fukushima Dai-ichi BWR reactors have been re-produced using MAAP. In addition, Hitachi-GE presents a comparison (Ref. 54) of selected unmitigated sequences between its MAAP results for the UK ABWR and results obtained with MELCOR for the Japanese ABWR by the Japanese Nuclear Energy Safety Organisation (JNES). This evidence provides me with assurance that the MAAP code is suitable for simulating BWR-type reactors (SAP AV.1) and that the relevant processes are being modelled (SAP AV.2). Overall, I am satisfied that Hitachi-GE's submissions meet the relevant expectations of SAP AV.5 regarding provision of documentation.

116. ONR has challenged Hitachi-GE in RQ-ABWR-0192 to justify why it has used MAAP 4.07 instead of the more recent MAAP version 5. I have considered Hitachi-GE's response in conjunction with ONR's PSA assessment (Ref. 4), which has considered advice from ONR's TSC supporting the PSA assessment. As a result, I am satisfied that relevant accident phenomena have been captured in Hitachi-GE's MAAP 4.07 analysis.

117. Hitachi-GE has provided a copy of the MAAP input deck developed for the UK ABWR (Ref. 56). The input deck was subject to review as part of the PSA assessment (Ref. 4). The targeted review of key input parameters, which are known to have a potential impact on analysis results, found no errors or inconsistencies, and found that the initial and boundary conditions for the sequences were correctly modelled. Notwithstanding this finding, I have identified specific areas where Hitachi-GE's UK ABWR MAAP representation does not adequately represent the actual plant design, for example in relation to nodalisation of regions in the UDW and also the MCCI modelling of the pedestal wall; I discuss these aspects later in my assessment.

118. Hitachi-GE has performed sensitivity analysis as a key part of the investigation of phenomenological uncertainties. Based on a review of international studies performed

by the US NRC, EPRI and others, 13 areas related to the severe accidents modelling were identified and analysed (Ref. 55):

- core melt progression and hydrogen generation;
- fission product release from core;
- CsI re-evaporation;
- time of vessel failure;
- re-criticality during in-vessel recovery;
- DCH;
- ex-vessel FCI;
- MCCI;
- containment failure location;
- containment failure area;
- high temperature failure of drywell;
- S/P decontamination factors; and
- main steam line creep rupture.

119. Hitachi-GE has discussed how uncertainty in these areas could affect accident progression and outcomes. Hitachi-GE has also discussed additional sensitivities as part of the PSA Level 2 modelling (Ref. 57). Based on Hitachi-GE's submissions, I consider that the sensitivity of MAAP results to key assumptions is adequately understood. This addresses my expectations for SAP AV.6. I recognise that knowledge of severe accident phenomena is continuing to develop and I would expect a future licensee to update its severe accident safety case to reflect new insights as they become available. I would expect that on-going and future investigations into the accident progression in Fukushima Dai-ichi will also produce relevant data and insights that can be considered for the UK ABWR design by the future licensee.

120. To further support my assessment of MAAP 4.07 for accidents involving the reactor, Amec Foster Wheeler has performed independent confirmatory analyses of selected accident sequences (Ref. 20) using the MELCOR code. MELCOR (Ref. 58) is a fully integrated, engineering-level computer code, developed by Sandia National Laboratories for the United States Nuclear Regulatory Commission. It is widely used by regulators around the world. MELCOR treats a broad spectrum of phenomena which may be encountered during the progression of a severe accident. I consider MELCOR to be an alternative, but equally respected, code that is the obvious benchmark for MAAP.

121. The strategy for the independent confirmatory analysis work was informed by a study of best-practice for confirmatory analysis (Ref. 19). The objective was to gain confidence in Hitachi-GE's severe accident analysis, through confirmatory analysis of a small sample of severe accident fault sequences. The MELCOR model was developed from existing MELCOR models of the Fukushima Units 1 & 3 BWRs (Ref. 59 & 60), modified by Amec Foster Wheeler using Hitachi-GE's source documentation, drawings and specifications to reflect the UK ABWR design. This work has provided useful insights into strengths and weaknesses of MAAP to inform my assessment.

122. In general, I have observed broad agreement between the results from Hitachi-GE's MAAP severe accident analysis and the MELCOR independent confirmatory analysis for the sequences considered. Nominally, the timelines of the MELCOR transients agree reasonably well with those predicted by MAAP, but it is noted that the modelling is very different in some parts of the code and that there are a number of cancelling effects. However, given that these analyses were based on models developed independently and using codes with an entirely separate developmental pedigree, this good agreement does provide me with confidence that the Hitachi-GE analysis is, in general, reasonable and appropriate. The positive findings from this work support my conclusion that Hitachi-GE's model is representative of the UK ABWR design and provides adequate modelling of the relevant phenomena (SAPs AV.1 and AV.2). I refer

to specific findings from the independent confirmatory analysis in later sections of my report. I have used these findings to support judgements on the adequacy of Hitachi-GE's safety case.

123. For severe accidents involving the SFP, Hitachi-GE has used MAAP 5.03 as this version is capable of simulating the relevant heat transfer processes. One additional feature included in MAAP 5.03 is the ability to simulate the effect of water sprays to provide cooling to fuel assemblies in the SFP. Hitachi-GE has used this feature of the code to consider the potential benefits of sprays in limiting radioactive release after the onset of fuel damage (Ref. 36). ONR's Fuel and Core assessor has carried out a specific review of the SFP spray model in MAAP 5.03 and has judged that the physical models applied are appropriate (Ref. 61).

124. In summary, I am satisfied that Hitachi-GE's MAAP model adequately represents the extant design as well as relevant processes and phenomena and is fit for purpose to support Hitachi-GE's safety case. This position is supported by findings from independent confirmatory analysis. I therefore conclude that Hitachi-GE's use of MAAP meets my GDA expectations for the AV series of SAPs.

4.3.4 GOTHIC

125. GOTHIC is a general-purpose thermal-hydraulics computer code for multi-component multi-phase flow developed by Zachry Nuclear Engineering's Analysis Division sponsored by EPRI. GOTHIC is used to support Hitachi-GE's severe accident hydrogen management safety case. Specifically, the code is used to model the flow of hydrogen in the R/B following leakage from the PCV (for accidents initiated at power) and hydrogen released directly into the R/B (for accidents during shutdown and from the SFP).

126. Hitachi-GE claims that GOTHIC is suitable for the simulation of multi-component, multi-phase flow and heat transfer in the R/B. Hitachi-GE points out that the development of the code follows quality assurance procedures endorsed by the US NRC. The physical models of GOTHIC have been validated against test data for relevant phenomena in the containment of nuclear reactors and test data for hydrogen behaviour in containments are part of its validation matrix (Ref. 62). The results of a NEA benchmark in 2007 indicate that a previous version of GOTHIC was capable of adequately predicting and reproducing test results on hydrogen behaviour (Ref. 63). This provides me with assurance that GOTHIC considers the relevant phenomena for the application being considered.

4.3.5 Other codes

127. Hitachi-GE's modelling of steam explosions (Ref. 47) uses the JASMINE code (Ref. 64) developed by the Japan Atomic Energy Agency specifically for this purpose. JASMINE is based on mechanistic models for pre-mixing of corium particles with water and the mechanical energy release caused by propagation of a shock-wave and subsequent fragmentation of the particles. As an input, JASMINE uses corium jet sizes predicted by MAAP. The energy release is used as an input to the AUTODYN code which calculates propagation of the resulting pressure wave in the LDW. Hitachi-GE claims that the JASMINE code has been validated against experimental data (Ref. 47) and is suitable for predicting the energy release from a steam explosion in situations likely to be encountered in the UK ABWR. AUTODYN is a recognised engineering code used in a range of applications.

128. ONR's Fuel and Core inspector has carried out a targeted review of JASMINE (Ref. 61) on the basis that this is likely to present the main source of uncertainty in the analysis of ex-vessel steam explosions. The conclusion (Ref. 61) is that JASMINE provides a reasonable representation of the physical processes likely to occur and is

suitable to support the assessment of steam explosion energy release. Uncertainties associated with AUTODYN are considered to be less significant and so the code has not been subject to specific review. A factor in my decision is Hitachi-GE's evidence which shows that the load on the pedestal wall would be significantly below that required to cause failure of the pedestal wall. I give further consideration to Hitachi-GE's analysis of steam explosions in Section 4.5.5.

129. SUPRA is a mechanistic suppression pool scrubbing model originally based on extensive studies by EPRI. Through response to RO-ABWR-0066 (Ref. 65), Hitachi-GE claims that SUPRA adequately models removal of fission products in the S/P. The SUPRA code has been considered in the Reactor Chemistry assessment report (Ref. 66) and I have not considered this further in my review.

130. STAR-CCM+ is a commercially available multi-physics simulation code with a wide variety of applications outside of nuclear engineering. The code has been used by Hitachi-GE to provide CFD analysis of the dispersion of corium particles in the PCV following RPV failure at high pressure, resulting in HPME. The UK ABWR includes design provisions to depressurise the reactor before RPV failure to prevent HPME. I therefore judge that the adequacy of STAR-CCM+ is unlikely to have a significant impact on nuclear safety.

131. ABAQUS is a commercially available structural analysis code which has been used by Hitachi-GE to analyse containment performance under severe accident loads. This has been considered by ONR's Civil Engineering and Structural Integrity assessors (Refs. 68, 69).

4.4 Severe accident phenomena

4.4.1 Background

132. Hitachi-GE considers that appropriate understanding of severe accident phenomena and progression is important to prevent and/or mitigate a severe accident in the UK ABWR (Ref. 24).

133. For the reactor, Hitachi-GE identifies and discusses three specific phases of accident progression and presents analysis of the phenomena using MAAP. These are discussed for the reactor at power (Ref. 35) and during shutdown (Ref. 36). For the reactor, Hitachi-GE identifies the following phases and phenomena as being relevant:

- in-core phase (prior to core support plate failure):
  - core melt and relocation; and
  - re-criticality.
- lower plenum phase:
  - in-vessel steam explosion; and
  - RPV failure.
- ex-vessel phase:
  - ex-vessel steam explosion;
  - steam generation due to FCI;
  - DCH;
  - direct debris interaction
  - MCCI;
  - production and distribution of hydrogen; and
  - over-pressure and over-temperature.

134. Accident phenomena and progression for the SFP are discussed in Ref. 36. Hitachi-GE identifies that the heat transfer processes following fuel uncovery and hydrogen generation and combustion are the key phenomena.

135. FA.15 of the SAPs states that the severe accident analysis should be based on an adequate understanding of the severe accident phenomena and accident progression. The objective of this part of my assessment is therefore to consider whether Hitachi-GE has considered the relevant severe accident phenomena that characterise accident progression for the UK ABWR.

4.4.2 In-core phase

4.4.2.1 Melt progression

136. Hitachi-GE describes the progression of core degradation following loss of core cooling. In the short term, steam cooling of the upper parts of the fuelled core may continue to be effective even if the water level drops below TAF. However, if cooling cannot be restored then fuel melting is expected to commence before the level reaches bottom of active fuel (BAF). For the purposes of Severe Accident Management Guidelines (SAMGs), Hitachi-GE considers that entry into a severe accident condition would be indicated by an increase in radiation levels in the PCV and/or detection of hydrogen in the PCV. Both of these would be indicators of fuel damage.

137. In the absence of core cooling, Hitachi-GE's safety case (Ref. 35) identifies that progression of core damage will be characterised by:

- candling of melted fuel, causing fuel channel blockage;
- hydrogen generation due to oxidation of fuel cladding;
- relocation of degraded fuel to the core support plate at the bottom of the core;
- failure of the core support plate.

138. The rate and extent of candling of fuel, and thus blockage of fuel channels, is a key factor in the progression of core melt. Candling of fuel in a channel results in blockage of the channel and an absence of steam cooling above the blockage. Hydrogen is generated from the exothermic oxidation of core materials and produced at a rate dependent on temperature and the availability of water/steam. Unlike in the case of a PWR, individual fuel assemblies of an ABWR are enclosed in zirconium channel boxes which, at least initially, limit cross-flows of water and steam between adjacent assemblies. As degradation progresses, melted fuel, zirconium and control rod material, known as 'corium', eventually collects on the core support plate. The core support plate itself may fail in the absence of cooling, marking a key stage in the progression of the accident.

139. Hitachi-GE's treatment of these phenomena has been investigated by my TSC as part of a comparative study of the models in MAAP and MELCOR (Ref. 20). Whilst the relevant phenomena are considered in both codes, my TSC has identified that there are some significant differences in how the phenomena are modelled. My TSC considers that the biggest fundamental difference in the approach to modelling core degradation between MELCOR and MAAP is in the candling and blockage models, with the result that MAAP tends to predict more channel blockage. This has a consequential effect on debris temperature and hydrogen generation during core melt. Despite the differences, my TSC has advised (Ref. 20) that both codes provide a credible representation of the core degradation phase. Despite the different approaches, I take assurance from the confirmatory analysis which supports the MAAP predictions for the time to core support plate failure. I am therefore satisfied that Hitachi-GE's description of phenomena and modelling of core melt is reasonable, and that this provides a suitable basis to inform the severe accident analysis.

4.4.2.2 Re-criticality

140. During core melt, boron carbide in the control rods would form a eutectic with the outer steel casing of the rods, resulting in melting and relocation of the material at temperatures below that required to melt the fuel cladding. In the absence of sufficient water coolant, the core would be under-moderated and would remain sub-critical. However, if the core were to be re-flooded before melting of the cladding, then there would be the potential for parts of the intact core to return to criticality.

141. If re-criticality were to occur during re-flooding, I am less concerned about a core-disruptive event, such as an unmitigated prompt criticality event. However, I regard the establishment of locally critical regions in the core as a theoretical possibility. Such regions could easily reach power levels far exceeding the capacity of the vent systems or the RHR. Persistent re-criticality could therefore potentially challenge PCV integrity, resulting in an unfiltered containment failure with a partially destroyed (but cooled) core. This is not considered as an outcome in Hitachi-GE's PSA.

142. Hitachi-GE's severe accident analysis and PSA assume that re-criticality following initiation of core melt will not occur and that heat generation will be limited to decay heat levels. Hitachi-GE claims that some boron would be retained in the core following melting of control rods, which I agree seems plausible. Hitachi-GE refers to work carried out for an earlier BWR design which shows that retention of only a small fraction (5%) of the boron in the core would be sufficient to maintain sub-criticality. Hitachi-GE's argument assumes a homogeneous distribution of the remaining boron which is unlikely to be representative. Whilst I accept the principle of Hitachi-GE's argument, I am not convinced that this evidence alone supports a definitive conclusion that re-criticality of the intact core cannot occur.

143. In addition, Hitachi-GE claims that the potential for re-criticality during re-flooding would be restricted to a limited 'time-window' between melting of control rods and melting of fuel. Re-criticality could only occur during this time window if water injection was recovered and sufficient water level was reached in the core in that period. Hitachi-GE claims that during re-flood, fuel cladding would likely become brittle and shatter, resulting in collapse of the fuel and mitigation of any criticality risk. Neither MAAP nor MELCOR include models for brittle fracture of cladding caused by the thermal shock on re-flood, but it seems plausible that fuel collapse into a non-critical geometry could occur. I am satisfied that Hitachi-GE has identified and considered the potential for re-criticality and that is reflected in the proposed mitigation strategy, which is to inject boron into the core using the SLCS. I consider this further in Section 4.6 as part of my assessment of Hitachi-GE's re-flooding analysis.

144. Hitachi-GE also claims that re-criticality of a water moderated fuel debris bed (formed following core melt) is unlikely and provides a supporting reference to criticality calculations. Whilst I have not considered the evidence in detail, I am satisfied with Hitachi-GE's argument that re-criticality in a debris bed would be unlikely because of the non-ideal geometry and material distribution in the debris.

4.4.3 Lower plenum phase

4.4.3.1 Corium behaviour in the RPV lower head

145. Following failure of the core support plate, corium would re-locate into the lower plenum and undergo fuel-coolant interaction (FCI) with water residing in the RPV below the core support plate. Hitachi-GE claims (Ref. 35) that the risk of an in-vessel steam explosion (resulting in so-called α-mode PCV failure) during lower plenum FCI is negligible. This is because of the non-ideal conditions for an explosion and the presence of structures in the lower plenum which would dissipate energy if a steam

explosion did occur. I am satisfied that Hitachi-GE's conclusion is consistent with current scientific understanding of this topic (Ref. 70).

146. In Appendix F of Ref. 35, Hitachi-GE has presented a calculation that the debris bed in the lower plenum would be coolable in-vessel if RPV injection using the FLSS or FLSR is available after core support plate failure. However, the PSA does not claim any credit for establishing debris coolability following timely coolant injection after the core support plate has failed and it is assumed by Hitachi-GE that from this point an accident will progress to RPV failure. Furthermore, Hitachi-GE does not claim an in-vessel retention (IVR) strategy, which would be difficult to justify for the UK ABWR due to the penetrations in the bottom of the RPV lower head. Instead, Hitachi-GE's case is that if the RPV fails, the released corium can be managed ex-vessel.

4.4.3.2 RPV failure modes

147. A failure of the RPV in the lower head region is expected in the absence of debris cooling. This represents a key stage in the accident progression, leading to the possibility of subsequent challenges to the containment. Hitachi-GE identifies a number of potential RPV failure locations and modes (Ref. 35), which it claims are included in the MAAP model of the UK ABWR:

- melting of control rod guide tubes, instrument tubes and the bottom drain line due to corium ingress;
- ejection of control rod guide tubes, instrument tubes and the bottom drain line due to stress-induced weld failure;
- jet ablation of the RPV wall due to corium release from the core support plate;
- over-heating of the RPV wall by the high heat conductivity metal layer, over-lying the corium pool; and
- creep rupture of RPV lower head.

148. Hitachi-GE's analysis shows that ejection of a control rod guide tube is the dominant RPV failure mode for the UK ABWR, resulting in a failure of about 0.2 m diameter. I am aware that larger failures of the lower RPV head could occur, however given the design of the UK ABWR with multiple control rod drive penetrations at the bottom of the lower head, the findings of Hitachi-GE's analysis appear to be reasonable. Hitachi-GE claims that smaller penetrations such as instrument lines and the RPV bottom drain line are not vulnerable to early failure, in advance of control rod guide tube failure. For both ex-vessel FCI and MCCI effects, Hitachi-GE argues that initial corium release rates for the bottom drain line would be much lower and these effects would therefore be bounded by control rod guide tube failure. I consider these arguments to be reasonable.

4.4.4 Ex-vessel phase

149. If RPV failure is anticipated, Hitachi-GE's accident management strategy is to pre-flood the LDW cavity using low pressure FLSS or FLSR injection. Even if pre-flooding is not carried out, significant quantities of water may potentially be present in the LDW at the time of RPV failure, for example from S/P overflow. When the RPV fails, corium would be released into the water pool. Hitachi-GE identifies that there is the potential for a steam explosion, but argues that this is unlikely because of the limited volume of water and the need for an external trigger to disrupt the stable vapour film at the surface of the corium particles. Despite this, Hitachi-GE has investigated the magnitude and effects of potential steam explosions in detail using JASMINE and AUTODYN (Ref. 47) and has also considered the likelihood of a steam explosion using the risk oriented accident analysis methodology (ROAAM) to support the PSA (Ref. 35).

150. Even in the absence of a steam explosion, Hitachi-GE identifies the potential for rapid pressurisation of the PCV due to FCI and considers the effects using MAAP.

151. Hitachi-GE also identifies DCH to be an additional challenge if the RPV fails at high pressure. The accident management strategy is to depressurise the reactor before RPV failure. However, if failure at high pressure occurs, Hitachi-GE considers the potential heating of the containment atmosphere by corium debris and the effect of direct debris interaction on containment structures.

152. MCCI is recognised by Hitachi-GE as an important phenomenon for the UK ABWR. Corium released from the RPV into the LDW would result in MCCI, giving rise to ablation of the concrete on the LDW floor and the pedestal wall. Non-condensable gases, including hydrogen, would also be generated. This could occur at an enhanced rate if the LDW is dry (potentially leading to containment failure), but the design intent is that cooling of the corium by water injected into the LDW, or from the LDF, would mitigate the progression of MCCI.

153. Hitachi-GE uses MAAP to model MCCI. The model in MAAP 4.07 pre-dates a phase of significant improvements that were implemented in the MAAP 5 versions. Hitachi-GE has provided arguments in Ref. 55 that inputs for the simplified coolability model in MAAP 4.07 have been specified in an appropriate manner, based on results from the latest models in MAAP version 5. The confirmatory MELCOR analysis (Ref. 20) has provided me with assurance that Hitachi-GE's treatment of MCCI using MAAP 4.07 is reasonable.

4.4.5 Phenomena for accidents involving the SFP

154. For severe accidents involving the SFP, Hitachi-GE identifies that the key phenomenon to be modelled is the degradation of fuel once the SFP water level has dropped below TAF. The key aspect is the rate of heat-up of the fuel and the effect this has on release of fission products as the fuel degrades. For the purposes of PSA source terms, Hitachi-GE assumes that fission products released from the fuel are not retained in the secondary containment (due to opening of the blowout panel) and are released to the environment.

155. For the SFP, Hitachi-GE uses MAAP 5.03 to determine the progression of fuel damage (and resulting fission product release) after the water level drops below TAF. MAAP version 5 introduced specific modelling capabilities for fuel in SFPs which was not included in earlier versions. The SFP capability, which was developed in response to the Fukushima Dai-chi accident, includes the expected features such as fuel damage and melt progression, air and steam oxidation, hydrogen production, and fission product release. Following drain-down or boil-off, passive cooling of low-rated fuel assemblies by air might be effective. However there is considerable uncertainty in this mode of cooling and Hitachi-GE has chosen not to take any credit for this in its safety case. Hitachi-GE argues that MCCI is not a concern for the SFP due to the relatively low levels of decay heat in the stored fuel assemblies. It is also argued that local criticality, caused by the melting of fuel rack fixed neutron absorbers to less optimum configurations, is unlikely. I am satisfied with these arguments and am content that Hitachi-GE has identified the relevant severe accident phenomena for the SFP. I am also satisfied that Hitachi-GE's use of MAAP 5.03 to model these phenomena is appropriate.

4.4.6 Hydrogen

156. For reactor severe accidents, Hitachi-GE identifies that hydrogen would be generated by steam oxidation of fuel cladding and steel components in the RPV, and also ex-vessel due to MCCI. Hydrogen would be released into the PCV, which is designed to be leak-tight. Hitachi-GE does not consider combustion in the PCV because the containment is normally inerted with nitrogen when the reactor is at power. This means that there would be insufficient oxygen in the PCV to support combustion, even if hydrogen concentrations rise to significant levels. However, Hitachi-GE recognises the

potential for leakage of hydrogen from the PCV into the R/B, where combustion could occur in the presence of air.

157. For severe accidents involving the reactor during shutdown reactor with the PCV head removed, hydrogen could be released directly into the R/B. Hydrogen could also be released into the R/B during a SFP severe accident due to oxidation of uncovered fuel cladding. For both these cases, Hitachi-GE identifies the potential for hydrogen combustion in the R/B.

158. For both PCV leakage and releases directly into the R/B, Hitachi-GE has used the GOTHIC code to model the transport of hydrogen. Overall, I am satisfied that Hitachi-GE has identified the relevant hydrogen phenomena and used a suitable model to consider the potential for combustion.

4.4.7 Radionuclide behaviour

159. The assessment of this aspect has been led by the Reactor Chemistry discipline, principally against Hitachi-GE's responses to two Regulatory Observations: RO-ABWR-0043 (Ref. 71) and RO-ABWR-0066 (Ref. 72). The first of these covers the effect of water pH control on the retention of iodine in the S/P. The second extends this to a wider consideration of chemical and physical behaviour or radionuclides in severe accidents. In response to these Regulatory Observations, Hitachi-GE has identified (Ref. 65) the 'mechanisms' relevant to severe accidents and explained how these have been considered in the severe accident analysis. The following mechanisms have been considered by Hitachi-GE:

- quantity and speciation of radioactivity release from fuel;
- effect of containment spray operation;
- retention of radionuclides on containment wall surfaces;
- dissolution and scrubbing of radionuclides in the S/P;
- retention of radionuclides in the S/P;
- retention of radionuclides in the primary containment;
- retention and leakage of radionuclides in the R/B; and
- retention and decay of radionuclides within the FCVS.

160. ONR's assessment of the response to the Regulatory Observations can be found in the Reactor Chemistry assessment report (Ref. 66).

4.4.8 Assessment summary

161. Reflecting expectations in FA.15 of the SAPs, I have reviewed Hitachi-GE's consideration of severe accident phenomena and how this supports the severe accident analysis. As already noted, phenomena relating to the chemical and physical behaviour or radionuclides have been considered as part of ONR's reactor chemistry assessment (Ref. 66). However, within the scope of my assessment, I am satisfied that Hitachi-GE has:

- demonstrated an adequate understanding of the relevant severe accident phenomena and described these in the safety case;
- described how the phenomena affect the accident progression; and
- considered accident phenomena using suitable computer codes.

4.5 Containment performance

4.5.1 Background

162. Hitachi-GE's concept for severe accidents is that the plant should not experience primary containment failure. The primary containment for the reactor is provided by the

PCV. The PCSR (Ref. 24) summarises the acceptance criteria used by Hitachi-GE to demonstrate that containment integrity is maintained. Hitachi-GE claims that challenges to containment integrity are prevented by specifying an appropriate design envelope and by providing severe accident mitigation measures to prevent the design envelope being exceeded. Hitachi-GE claims that:

- Failures due to over-pressure and over-temperature are prevented by appropriate design of the PCV boundary and provision of measures that are designed to ensure that:
  - conditions in the PCV are maintained below failure criteria by features such as the LDF, COPS and PCV sprays to control temperature and pressure below failure criteria; and
  - PCV failures due to DCH (and rapid pressurisation due to ex-vessel FCI) are prevented by ensuring that the RPV can be depressurised to below 2 MPa before RPV failure.
- Hydrogen concentrations can be maintained below flammable limits and therefore there would be no challenges to the containment from hydrogen combustion.
- The pedestal wall will withstand pressure waves should an ex-vessel FCI steam explosion occur, thus preventing containment failure.
- Concrete ablation due to MCCI is limited by flooding of LDW such that collapse of the pedestal wall, leading to gross containment failure, does not occur.
- Corium does not come into contact with the PCV liner due to the layer of concrete in the base of the LDW and the concrete pedestal wall.

163. If severe accident measures are effective then Hitachi-GE identifies that the principal challenges to the PCV would be from over-pressure and over-temperature. Hitachi-GE claims that the ultimate capacity of the containment is in excess of 2×Pd. For high temperatures, Hitachi-GE claims that integrity of critical components would be maintained up to about 300°C. Hitachi-GE argues that the assumed failure criteria are conservative and that the margins, in particular to over-pressure, are substantial. Hitachi-GE claims that the design incorporates measures to prevent conditions exceeding the stated ultimate capacity – I consider the effectiveness of these measures in Section 4.6 of my report.

4.5.2 Assessment of containment concept

4.5.2.1 Primary containment concept

164. The importance of providing a robust containment is reflected in UK and international expectations, in particular:

- SAP ECV.2, which states that containment systems should be designed to minimise releases, including under accident conditions;
- WENRA guidance (Ref. 10) which emphasises the importance of maintaining the integrity of the containment barrier throughout the course of a core melt accident; and
- the IAEA Director General's report on the Fukushima Dai-ichi accident (Ref. 16), which re-states the need for a reliable confinement function for beyond design basis accidents.

165. I consider that Hitachi-GE's concept for the containment of the reactor, which is that containment failure should not occur in a severe accident, is consistent with these expectations.

166. I also consider the high level expectation of SAP ECV.3, which states that the primary means of confining radioactive materials should be through the provision of passive sealed containment systems and intrinsic safety features, in preference to the use of

active dynamic systems and components. The UK ABWR PCV is designed to provide a leak-tight containment, which retains radioactivity in a severe accident. Hitachi-GE's concept does rely on a means for long-term heat removal from the containment, which preferentially would be to restore closed-loop cooling to the S/P and the DW using active systems. However, heat removal can also be achieved by venting excess pressure from the containment to atmosphere, which would be through filters to minimise the radiological consequences. If necessary, venting can be achieved by passive opening of the COPS bursting disks. I conclude that UK ABWR primary containment concept is consistent with the expectations for a passive containment system in SAP ECV.3.

167. The R/B of the UK ABWR provides a secondary containment function for the PCV, although Hitachi-GE does not make any severe accident performance claims.

4.5.2.2 Reactor building

168. Under normal operation the R/B is maintained at negative pressure by the R/A HVAC system. In fault or accident conditions the R/A HVAC system would be automatically isolated and the SGTS, a Class 2 system, would be started. The SGTS trains are fitted with filters that are designed to be effective against particulate releases. Hitachi-GE does not make any claims on the containment function of the R/B during shutdown or SFP severe accidents.

169. Severe accidents during reactor shutdown (with the PCV head removed) or from the SFP would most likely be initiated by design basis loss of cooling faults, or LOCAs leading to drain-down. These faults could lead to boiling of water in the reactor and/or SFP, which would result in generation of steam in the R/B. In the design basis safety case (Ref. 32), Hitachi-GE has identified that the steam generation rate would exceed the capacity of the SGTS, whilst the R/A HVAC is assumed to be automatically tripped. Following challenge from ONR's Fault Studies inspector, Hitachi-GE has carried out an optioneering study to consider design solutions for removing steam from the R/B. Hitachi-GE's ALARP solution (Ref. 73) is to release steam from a blowout panel located on the upper wall of the R/B.

170. Should a loss of cooling lead to boiling in the shutdown reactor or SFP, Hitachi-GE argues that, provided fuel remains covered, there will be no fuel damage and any release of radioactivity through the blowout panel will be limited. In this situation, Hitachi-GE argues that the radiological consequences would meet ONR's Numerical Target 4 for design basis faults. Hitachi-GE's ALARP case is considered in detail in ONR's Fault Studies assessment report (Ref. 3). However, as noted in Ref. 3, Hitachi-GE's ALARP arguments are closely linked with the proposed severe accident management strategies.

171. If cooling or make-up cannot be restored, the design basis faults would eventually lead to uncovery of fuel in the shutdown reactor or the SFP, resulting in a severe accident due to fuel damage. Hitachi-GE has identified that a deflagration of hydrogen generated due to steam oxidation of fuel cladding could present a challenge to the R/B structure and important SSCs located within the R/B. Hitachi-GE's strategy for managing hydrogen in the R/B is to use the blowout panel and a large equipment door at ground level in the R/B to promote natural ventilation, thereby preventing the build-up of flammable concentrations in the R/B. I consider the effectiveness of this measure for hydrogen control in Section 4.6.

172. A consequence of the blowout panel being open in a severe accident is that this would allow radioactivity to pass direct from the R/B to the atmosphere. Hitachi-GE's ALARP arguments for the blowout panel consider the benefits and dis-benefits of the blowout panel for both design basis steam generation and severe accident management. In the context of R/B containment for severe accidents, Hitachi-GE argues that:

- The blowout panel (and equipment door) has a positive benefit in contributing to Hitachi-GE's hydrogen management strategy for the R/B, by prevent hydrogen deflagration if measures to prevent fuel uncovery have failed. If a hydrogen deflagration were to occur, this could impair the integrity of the R/B structure and damage SSCs required for long term accident management.
- Even without the blowout panel, the R/B structure is not designed to be leak-tight.

173. Severe accidents during shutdown or in the SFP would be a progression of slow developing faults. For shutdown faults I recognise that these would only occur in time-limited modes of operation, ie refuelling outages. In both cases, prevention of a severe accident could be achieved through simple measures, involving continuous addition of relatively small quantities of water. This could be delivered by any one of a number of diverse systems. A further important factor is Hitachi-GE's claim that uncovery of fuel in the SFP or shutdown reactor has been practically eliminated, which I consider in more detail in Section 4.9. Overall, I am satisfied that Hitachi-GE's arguments provide a valid basis to support the containment concept for the extant design.

174. ECV.7 of the SAPs states that appropriate sampling and monitoring systems should be provided outside the containment to detect, locate, quantify and monitor for leakages or escapes of radioactive material from the containment boundaries. Given the obvious potential for releases from the blowout panel in a severe accident, I would expect that a future licensee to give specific consideration to the monitoring and sampling of releases from the blowout panel as part of the development of accident management arrangements. I am satisfied that this can be developed by the future licensee as part of normal business.

4.5.3 Assessment overview of PCV failure modes

175. There is an expectation in SAP ECV.3 that the performance of the containment during severe accidents should be defined. Hitachi-GE's original submissions in this area did not meet my expectations, either in terms of a clear understanding of how the primary containment could fail in a severe accident, or why the design was robust against such challenges. During Step 3 of GDA I raised RO-ABWR-0023 Action 5.3, requiring Hitachi-GE to describe containment failure modes under severe accident conditions. This was supplemented by RO-ABWR-0046 (Ref. 50), a cross-cutting RO raised by the ONR's PSA assessor. This RO required Hitachi-GE to characterise the challenges to the primary containment. The principal objective of this RO related to PSA expectations for the definition of an appropriate 'best-estimate' containment failure envelope. However, much of the scope of this RO is relevant to the severe accident safety case and my assessment of containment performance is based on Hitachi-GE's submissions in response to this RO.

176. Hitachi-GE's principal submission in response to RO-ABWR-0046 is Ref. 44. This provides a comprehensive review of the relevant failure modes and the success criteria which need to be met to ensure containment integrity. Hitachi-GE uses these success criteria in its severe accident analysis to demonstrate the effectiveness of the UK ABWR severe accident measures. I consider whether the severe accident measures are effective in preventing containment failure in Section 4.6.

177. I note that through the PSA, Hitachi-GE has identified accident sequences where the containment could potentially fail, for example due to multiple failures of design basis protection systems, or due to failures of severe accident measures. For these cases Hitachi-GE has considered the possibility of containment failure and analysed the radiological consequences of releases against ONR's Numerical Targets. This aspect is considered in ONR's PSA assessment report (Ref. 4). The focus of my assessment in this section is on the success criteria used to show that severe accident measures, when available, are effective in preventing containment failure.

178. My assessment activities in this area have been coordinated with a number of other disciplines, including PSA, Civil Engineering and Structural Integrity. To support the assessment of RO-ABWR-0046, ONR contracted Amec Foster Wheeler to carry out an in-depth review of Hitachi-GE's containment performance submissions. The findings from this review (Ref. 21) inform my assessment.

4.5.4 Pressure and temperature failure modes

4.5.4.1 Over-pressure failure

179. Hitachi-GE has used a structural model to determine the response of the RCCV structure and components to beyond design basis pressures and temperatures (Ref. 44). The model is based on the extant UK ABWR design, but uses generic BWR or J-ABWR calculations where the site-specific properties are not yet known. ONR's assessment of the withstand capability of the RCCV structure can be found in the Civil Engineering assessment report (Ref. 68). As part of the response to RO-ABWR-0046 , Hitachi-GE has also investigated the failure of PCV penetrations such as access and equipment hatches and the PCV head using engineering calculations in combination with experimental data for specific components such as seals (Ref. 44). Hitachi-GE's analysis has been reviewed by Amec Foster Wheeler (Ref. 21), in support of ONR's PSA assessment (Ref. 4). The conclusion from the PSA assessment is that the assumed PCV failure mode of 2×Pd at the drywell head flange is adequately bounding, although on a best-estimate basis the failure pressure would be expected to be higher. On this basis, I am satisfied that the assumed success criteria for PCV pressure (2×Pd) is adequate for assessing the effectiveness of severe accident measures.

4.5.4.2 Over-temperature failure

180. Hitachi-GE claims that over-temperature failures of seals and gaskets at low pressure would result in small leak areas (equivalent to about 0.3% to 2% of the area for PCV head flange overpressure failure) and therefore over-temperature is not likely to be the primary failure mode at low pressure. The temperature resistance of gaskets and seals has been subject to specific investigations by Hitachi-GE (Ref. 75) as part of Fukushima learning. As a result, the materials have been changed to EPDM rubber as this was identified as having a good combination of characteristics and superior to silicone rubber previously used in Japanese ABWR plants. Hitachi-GE claims that temperatures of 300°C are adequately bounding for flange gaskets and seals; this is confirmed by Amec Foster Wheeler's review (Ref. 21).

181. Addressing Fukushima Dai-ichi learning, Hitachi-GE has identified a mitigation action for further protecting the limiting PCV failure location at the PCV head flange by reactor well flooding, using the FLSS or FSLR. By cooling the PCV head structure and by reducing the differential pressure over the flange, Hitachi-GE claims this measure potentially increases the resilience to over-pressure failures at that location by at least 0.2×Pd. One concern with flooding the reactor well with cold water was that this could lead to a thermal shock of the PCV head structure, causing it to fail. In response to RO-ABWR-0059 (Ref. 76), Hitachi-GE has improved the design of the PCV head to ensure that it is robust against thermal shock. This has been the subject of assessment by ONR's Structural Integrity inspector (Ref. 69).

4.5.4.3 Negative pressure

182. The PCV will typically be at a higher pressure than the R/B during an accident. However, in the long-term phase, the depletion of non-condensable gases due to venting and condensation of steam during containment cooling could potentially lead to a negative pressure in the PCV. Venting from the WW can easily deplete the PCV inventory of non-condensable gases to a point where the negative pressure tolerance could be exceeded if the remaining steam in the PCV condenses. This could

potentially result in a breach of the PCV boundary leading to release of fission products when the pressure returns to above atmospheric.

183. Whilst the containment is designed for a pressure difference of -0.04×Pd, Hitachi-GE has shown that the PCV is robust against negative pressure differences of about -0.25×Pd (Ref. 44). Amec Foster Wheeler has reviewed Hitachi-GE's analysis of the negative pressure tolerance and did not identify any significant concerns (Ref. 21).

184. Hitachi-GE claims that generation of negative pressures requires an erroneous mode of operation of the RHR, the spurious operation of the spray systems (in particular for the WW) for an extended period of time, together with a failure to introduce nitrogen into the containment prior to closure of the vent line.

185. Hitachi-GE has considered the spurious operation of both DW and WW spray systems without previous PCV venting. Hitachi-GE claims that if the spray systems are turned off 30 minutes after the pressure in the DW or the WW drops to the environmental pressure, the resulting negative pressure in the PCV will remain within the design value (Ref. 44).

186. Hitachi-GE has outlined the procedure that operators would follow during severe accident scenarios to prevent negative pressure containment failure (Ref. 40). The procedure clearly establishes criteria for stopping PCV spray in order to prevent negative pressure failure. I accept that this specific risk can be effectively controlled by adequate procedures and relevant training of the operators.

187. The PCV is also robust against negative pressures due to quasi-static pressure differences and hydrodynamic loads that may occur during severe accidents. The V/Bs ensure that the differential gas pressure between the WW and DW does not exceed 3.4 kPa (Ref. 44). The vent pipes also ensure that the pressure in the DW cannot exceed that in the WW by more than the water level height difference (Ref. 77). I am therefore satisfied other potential mechanisms for negative pressure failure are mitigated by passive features of the PCV design.

4.5.5 Steam explosions

188. With regards to ex-vessel FCI following RPV failure at low pressure, my assessment considers two types of potential challenges to PCV integrity. The first challenge would be from a steam explosion, resulting in a pressure wave and potential damage to LDW structures and components.

189. Hitachi-GE assumes that containment integrity would be lost if the RPV pedestal wall suffered catastrophic structural failure. The pedestal wall comprises inner and outer steel plates, each a few centimetres thick, separated by about 2 metres of concrete (Ref. 78). Hitachi-GE assumes that pedestal wall failure would occur if stresses in the inner steel plate exceed the yield stress. Hitachi-GE's safety case for MCCI states that the pedestal wall retains its load bearing capacity even if the inner steel plate is degraded over its whole circumference. Therefore, in reality, a steam explosion would need to damage the outer steel plate, in addition to the inner plate, before pedestal wall collapse could occur. Hitachi-GE does not take credit for the contribution of the concrete to the structural strength of the RPV support.

190. Hitachi-GE has modelled the energy release from a steam explosion using the JASMINE code and the effect of the energy release on the pedestal wall using AUTODYN (Ref. 47). Hitachi-GE assumes a representative, but conservative case where the LDW is pre-flooded with subcooled water up to the vent pipe return lines (a height of seven metres) at the time of RPV failure (Ref. 35). The calculations are based on a corium release from a control rod drive penetration. I consider these to be reasonable assumptions.

191. Hitachi-GE's insights from experiments indicate that even with higher RPV pressures at failure and larger RPV failure areas, ex-vessel FCI energies are still within one order of magnitude of the 27 MJ energy release obtained by Hitachi-GE using JASMINE (Ref. 47). Based on its AUTODYN model, Hitachi-GE argues that a central explosion of 1500 MJ energy would be needed for a structural failure of the pedestal wall (Ref. 35). I therefore accept Hitachi-GE's arguments that an energy release sufficient to cause structural failure of the pedestal wall, and consequently the containment, is unlikely to occur.

192. Hitachi-GE's results show that the stresses at the inner steel plate of the pedestal wall reach only about 16% of the yield stresses (Ref. 47). I challenged Hitachi-GE in RQ-ABWR-1236 to consider whether an energy release centred closer to the periphery of the LDW could impose higher loads on the pedestal wall, for example from a corium release from a control rod drive tube on the extremity of the RPV. I consider that applying typical scaling laws would suggest much higher loads on the region of the pedestal wall closest to the source. In response (Ref. 79), Hitachi-GE argues that, based on the blast dynamics modelled by AUTODYN, localised loads would be a factor of two higher than that for an explosion source at the centre of the LDW, but that that there would still be sufficient margins to failure of the inner plate. Given the likely conservatisms in the energy release predicted by JASMINE, I judge that Hitachi-GE's response to this challenge is adequate.

193. In summary, I am content that Hitachi-GE has provided an adequate demonstration that the UK ABWR pedestal wall is likely to be robust against the effects of ex-vessel steam explosion pressure waves. I note that this position is consistent with the findings of the NRC staff, who in their review of the ABWR (Ref. 80) concluded that the reactor pedestal would be capable of withstanding the best-estimate loads associated with an ex-vessel steam explosion.

4.5.6 Hydrogen combustion

194. In Ref. 42, Hitachi-GE claims that the flammable gas control measures specified for the UK ABWR limit gas concentrations to below flammable limits and therefore prevent combustion. For the primary containment, these measures include a nitrogen-inerted PCV during normal operation. Hitachi-GE states that by preventing combustion, damage to containment structures and SSCs is prevented, thereby minimising the potential for radioactive release to the environment. Hitachi-GE states that combustion is considered to be prevented when the hydrogen concentration is under 4% by volume or the oxygen concentration is under 5%, based on Ref. 81. However, Hitachi-GE's analysis (Ref. 42) also refers to the flammability limits determined by Shapiro and Moffette (Ref. 81) which take into account steam concentration. Hitachi-GE uses these criteria for consideration of hydrogen combustion in the primary containment and in the R/B.

195. Hitachi-GE's analyses consider the potential for combustion based on the flammability effects of hydrogen. In Ref. 42 Hitachi-GE has considered the effect of carbon monoxide on the flammability of mixtures with hydrogen. Based on the findings of Ref. 82, Hitachi-GE concludes that this has the effect of increasing the flammability limits and therefore makes the mixtures containing carbon monoxide less flammable.

196. Hitachi-GE quotes experimental data which indicates that detonation (which could place damaging loads on the containment structures and SSCs) would only occur at concentrations well above the flammable limits. The conditions that could lead to detonation are recognised as an area of uncertainty. However, by limiting the gas concentrations in containment to below flammable limits, Hitachi-GE claims that the adverse effects of hydrogen (and carbon monoxide) combustion can be avoided.

197. Requirement 58 of IAEA SSR-2/1 (Ref. 9) states that there should be measures to control the concentrations of hydrogen, oxygen and other substances in the containment atmosphere in accident conditions so as to prevent deflagration or detonation loads that could challenge the integrity of the containment. WENRA Position 4 (Ref. 10) also sets the expectation that there shall be appropriate provisions to prevent the damage of the containment due to combustion of hydrogen. The objective of these provisions is to retain an engineered barrier against accidental off-site releases. I am satisfied that Hitachi-GE's objective to prevent flammable concentrations is consistent with these expectations. I also accept that Hitachi-GE has identified relevant criteria for considering whether combustion of hydrogen (and carbon monoxide) could occur, and therefore whether there would be challenges to the containment.

4.5.7 Ablation of the pedestal wall

198. Hitachi-GE claims that cooling of corium by flooding the LDW would significantly reduce ablation and that possible MCCI challenges to the pedestal wall would in reality only occur for unmitigated accident sequences where the LDW remains dry. However, it is important that there is an adequate understanding of the success criteria for this failure mode so that judgements can be made on how accidents might progress.

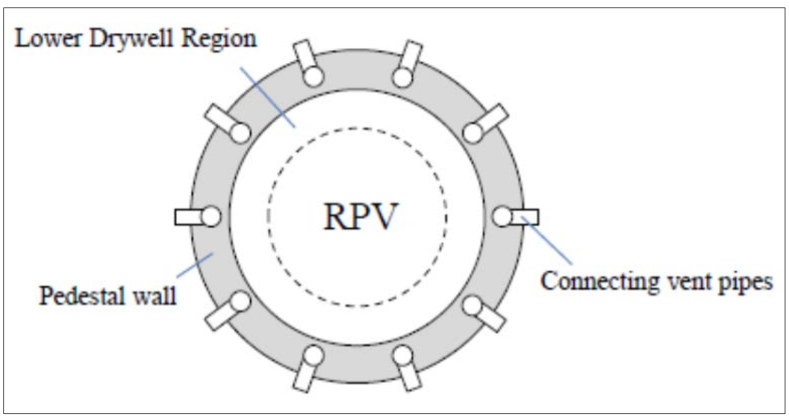

Figure 2: Section through pedestal wall

199. A horizontal cross-section though the concrete pedestal wall is shown in Figure 2. Corium spreading on the LDW floor would initially impinge the inner steel plate of the pedestal wall. Hitachi-GE's results show that it typically fails within less than 30 minutes after contact with the corium. Hitachi-GE assumes that the loss of the inner steel plate does not affect the overall integrity of the pedestal wall, although this is not discussed explicitly in the safety case. After the inner steel plate fails, the bulk concrete of the pedestal wall is ablated. The concrete of the pedestal wall is assumed to have no load-bearing function in the Civil Engineering safety case. Hitachi-GE assumes that the load-bearing function of the pedestal wall is maintained by outer steel plate until ablation has progressed radially through 90% of the thickness of the concrete. At this point, Hitachi-GE claims that the temperature rise of the outer steel plate of the pedestal wall cause the plate to lose its load-bearing capacity, resulting in collapse of the pedestal wall and failure of the containment due to loss of support to the RPV. After consultation with ONR's Civil Engineering inspector, I consider that Hitachi-GE submissions (Ref. 44 & 83) do not provide evidence to justify why integrity is maintained up to this point.

200. Figure 2 also shows the vent pipes which are positioned in the pedestal wall about 25 cm from the inner steel plate. Modelling of the behaviour of the corium and MCCI in such geometries is beyond the capability of existing severe accident codes. Hitachi-GE argues that it is not necessary to consider the presence of these pipes because the

vent pipes contain water and if they were breached there would be a flow of water from the S/P into the LDW. Hitachi-GE claims this would locally arrest any further radial MCCI through the pedestal wall. There would also be the additional benefit of flooding to the whole of the LDW, effectively limiting further MCCI in the LDW (Ref. 67). I am content with Hitachi-GE's argument that melt progression in these areas is likely to be arrested by the supply of cooling water into the vent pipes from the S/P. However, Hitachi-GE has not provided sufficient evidence that there would be no adverse effects of vent pipe break-through on integrity of the pedestal wall. In particular, if corium reaches the lowest S/P horizontal vent, it could potentially increase the heat load to the outer steel plate and other features of the pedestal wall. During interactions, Hitachi-GE has provided a qualitative argument that the outer steel plate would not fail if these additional loads are considered, but no evidence has been provided.

201. In conclusion, I am satisfied for GDA that Hitachi-GE has presented reasoned arguments in relation to pedestal wall failure and the effects of the vent pipes. I also note Hitachi-GE's claims (which I consider in Section 4.6) that there are significant margins to pedestal wall failure for mitigated scenarios where the corium is cooled. However, I am not satisfied the assumed failure point of the pedestal wall (assumed to be when ablation has progressed through 90% of the wall thickness) has been adequately justified in GDA. I therefore make the following Assessment Finding:

- AF-ABWR-SA-01: Failure of the pedestal wall has been identified by Hitachi-GE as a potential challenge to the containment in a severe accident. In GDA, Hitachi-GE has not presented detailed design calculations to justify the failure criterion for the pedestal wall when subject to molten core-concrete interaction. The licensee shall substantiate the failure criterion for the pedestal wall in severe accidents, including specific consideration of challenges to the pedestal wall structure from molten core material which may break though into the pedestal wall vent pipes.

4.5.8 Direct corium contact

202. Hitachi-GE identifies corium contact with the PCV liner as a potential challenge to containment integrity. For low pressure RPV failures, Hitachi-GE claims that particulate debris would be retained within the LDW by the pedestal wall and concrete floor. As a result, debris would not come into contact with the PCV liner. For high pressure RPV failures, Hitachi-GE identifies the potential for transport of debris into the LDW access tunnels. Transport of debris from the LDW to other parts of the containment could occur, but the transport path would be restricted by the connecting vents. Hitachi-GE has considered the effects of direct debris interaction on containment integrity. I consider Hitachi-GE's modelling of these effects in Section 4.6. However, I am satisfied that the relevant challenges from particulate debris have been considered.

203. For corium released onto the LDW floor, the potential for axial ablation of the concrete, leading to contact with the PCV liner, is not specifically discussed in PCSR Chapter 26 (Ref. 24). However, this is identified as a potential challenge in Hitachi-GE's containment performance analysis submission (Ref. 44) and results for axial ablation are included in Hitachi-GE's severe accident analysis (Ref. 35). I am therefore satisfied that this challenge has been considered.

4.5.9 S/P bypass

204. An important feature of the UK ABWR PCV is the S/P which is designed to provide pressure suppression by condensing steam generated during accidents. The S/P also provides scrubbing of steam and gases and acts as an additional barrier for releases from the containment vent. Pressure suppression and scrubbing would be impaired if there was a bypass of the S/P and steam were to pass directly into the WW gas space. Hitachi-GE's acceptance criteria do not include S/P bypass. This could potentially

result in additional challenges to the containment, so I also consider the significance of this in my assessment.

205. The tightness of V/Bs under severe accident conditions is important to Hitachi-GE's overall severe accidents safety case, even if this is not prominently acknowledged in the GDA submissions. Hitachi-GE considers that the V/Bs are robust against severe accident conditions, although no formal safety case claims are made. Hitachi-GE has however stated that the V/Bs are designed to withstand a pressure difference between LDW and WW of at least 220 kPa (Ref. 79) and that the V/B seals are tolerant of temperatures up to 300°C (Ref. 84). In response to challenge in RQ-ABWR-1385, Hitachi-GE has also considered if there could be benefits from placing V/Bs in other locations that might be less vulnerable to severe accident loads, but concludes that these would not be practical (Ref. 84).

206. For RPV failure at low pressure, Hitachi-GE has presented analysis to show that the V/Bs will not be challenged by radiative heat transfer from corium frozen to structures below the RPV. In addition, Hitachi-GE argues that corium in the LDW will be cooled by water from the LDF and that the steam will maintain LDW temperatures significantly below that required to fail the V/Bs. I would also expect that the V/B seals would be protected to a great extent against high LDW temperatures because the valves are located in the WW at the ends of pipes several metres in length. I would also expect the seals to be cooled by steam in the WW. A further consideration is that even if the seals were to degrade, the bypass flows would be relatively small and would be unlikely to prevent steam condensation via the vents. I am therefore satisfied V/B seal failure due to over-temperature is unlikely to have a significant effect on accident progression.

207. For structural failure of the valves, I consider that this would be expected to occur at much higher temperatures than that required for seal failure. In the case of HPME, Hitachi-GE has performed CFD simulations (Appendix D of Ref. 44) showing that only a small amount of corium debris is expected to be deposited in the V/B penetrations and that none would reach the V/B valves which are located at the far end of the penetration pipes. Whilst this analysis will be subject to uncertainties, I accept that failure of the valves would be unlikely.

208. Overall, I am satisfied that the V/Bs are likely to be sufficiently robust for severe accident challenges based on the information provided in GDA. However, Hitachi-GE has not presented severe accident claims for the V/Bs in its safety case. I would expect relevant claims to be defined so that equipment selected in the final design can be appropriately qualified. I have therefore raised the following Assessment Finding:

- AF-ABWR-SA-02: Hitachi-GE has assumed that the vacuum breakers would be robust against severe accident conditions, thus preventing suppression pool bypass. However, specific safety case claims or performance requirements for the vacuum breakers in severe accident conditions have not been identified in the GDA safety case documentation. The licensee shall identify the requirements placed on the vacuum breakers by the severe accident safety case and demonstrate that these can be met by the final design.

I am aware that the ends of the access tunnels have movable joints to accommodate expansion. I believe that these could provide a potential S/P bypass route if the tunnels are exposed to severe temperature loads, but I am not aware that Hitachi-GE has given specific consideration to this. I would expect that the potential vulnerability of these joints, particularly those adjacent to the LDW, be considered during detail design to ensure that the potential for bypass is minimised. I am satisfied with the position for GDA, but my expectation is that this should be examined by a future licensee in the final design.

4.6 Severe accident analysis

4.6.1 Background

209. As part of the PSA, Hitachi-GE has identified a range of at-power severe accident sequences and in Ref. 35 has analysed the progression of these using MAAP. These include:

- unmitigated sequences where severe accident measures are assumed to be unavailable, or are ineffective due to the timing or nature of the accident; and
- mitigated sequences where the benefit of severe accident measures, both passive and active, is considered.

210. In Ref. 35 Hitachi-GE summarises its unmitigated analyses in terms of the timings for key stages in the accident progression, in particular :

- core damage;
- core support plate failure;
- RPV failure;
- over-temperature/pressure failure of the PCV, and
- failure of the pedestal wall due to MCCI (leading to assumed catastrophic containment failure).

211. Hitachi-GE identifies the following representative reactor at power sequences for demonstrating the effectiveness of severe accident measures:

- Large LOCA resulting in reactor trip, but with core injection failure (sequence 'AE'†).
- Transient resulting in reactor trip, with failure of feedwater, high pressure injection and low pressure injection (sequence 'TQUV'†).
- Transient resulting in reactor trip, but with failure of feedwater, high pressure injection and failure of reactor depressurisation (sequence 'TQUX'†).
- Station blackout resulting in loss AC supplies to electrically-driven ECCS functions (sequence 'TB'†).
- Medium/small LOCA with high pressure injection, but with reactor depressurisation failure (sequence 'S12UX'†).

212. Hitachi-GE's main objective for its mitigated analyses is to demonstrate that severe accident measures can either stop the progression of core degradation before RPV failure occurs, or that containment failure after RPV failure can be prevented. Hitachi-GE claims that relevant combinations of mitigation measures are effective in ensuring that the containment success criteria (Section 4.5) are met. The effect of following severe accident measures have been analysed using MAAP:

- RPV depressurisation via RDCF;
- RPV injection via FLSS or FLSR;
- LDW flooding via FLSS or FLSR;
- LDW flooding by the passive LDF;
- UDW spray via FLSS or FLSR;
- restoration of containment heat removal using the RHR system; and
- PCV over-pressure protection by manual venting or the COPS.

213. The objective of Hitachi-GE's severe accident safety case for the shutdown reactor is to demonstrate that if fuel damage occurred, the fuel should be cooled to mitigate fission product release to the environment (Ref. 36). Hitachi-GE claims that in-vessel cooling will prevent RPV failure during shutdown, even if core damage occurs and

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† *Hitachi-GE's accident sequence nomenclature from the PSA, Ref.51.*

injection is initiated after failure of the core support plate (Ref. 24). If no cooling is initiated before RPV failure then Hitachi-GE claims that the LDF provides adequate cooling to ex-vessel corium in the LDW. In these cases the decay heat levels are much lower than for accidents initiated from at-power states and therefore the time to core damage is more than 30 hours (Ref. 36).

214. Similarly, the objective of Hitachi-GE's severe accident safety case for the SFP is to demonstrate that damaged fuel can be cooled to mitigate fission product release. Due to the large volumes of water present in the SFP, Hitachi-GE claims that in the worst case (with a full core off-load), fuel damage would occur after about 160 hours (for boil-off faults) and about 60 hours for small LOCAs (Ref. 36). Hitachi-GE claims that fission product release from damaged fuel can be mitigated by water sprays fed by the FLSS or FLSR.

4.6.2 Assessment overview

215. SAP FA.16 (Ref. 2) sets the expectation that severe accident analysis should be used in the consideration of further risk-reducing measures, beyond those derived from engineering analysis, DBA and PSA. Furthermore, there is an expectation that the severe accident analysis should complement accident management strategies and procedures (insofar as they are relevant to GDA) and support the PSA.

216. In this section of my report I present my assessment of Hitachi-GE's analysis of the accident phenomena and against the proposed containment success criteria (see Section 4.5). To support this assessment objective, the scope of my review covers Hitachi-GE's analyses of:

- unmitigated sequences to determine whether these adequately characterise the relevant accident progression and containment challenges.
- mitigated accident sequences to consider whether there is adequate substantiation of the effectiveness of the proposed accident management strategies and engineered severe accident features.

217. I have chosen to target my assessment on accidents for the reactor at power, where accident progression is more complex and likely to be subject to more uncertainty in terms of timings and success criteria.

218. For my assessment I have selected a small sample of severe accident sequences for detailed review. Firstly, I selected the TQUV sequence (transient followed by failure of high and low pressure injection) as a representative scenario where the RPV fails at low pressure. Hitachi-GE's analysis of this sequence includes a number of mitigated cases, which set out to demonstrate the effectiveness the engineered severe accident measures. The mitigated TQUV sequence has also been identified by Hitachi-GE as the representative sequence which defines the containment vent release category in the Level 3 PSA.

219. I have also selected the TB sequence (SBO with operation of RCIC for 8 hours, followed by failure of RPV depressurisation and therefore failure of LP injection). For the unmitigated case the RPV is assumed to fail at high pressure, resulting in challenges to the containment from rapid over-pressurisation. Hitachi-GE's analysis also considers the effectiveness of mitigation, including depressurisation of the reactor to a low pressure state before RPV failure.

220. These two accident sequences were also investigated by my TSC (Ref. 20) as part of the independent confirmatory analysis using MELCOR. I use insights from this work to inform my assessment.

4.6.3 Transient with failure of feedwater and ECCS injection

4.6.3.1 Unmitigated case

221. The TQUV plant damage state is characterised by a transient with successful scram, but with failure of feedwater and high and low pressure ECCS injection systems. RPV water level reduces as the remaining water in the core boils and is released to the S/P through the SRVs. This scenario assumes that the RPV is depressurised successfully when the RPV water level drops to 20% above BAF, which is in accordance with Hitachi-GE's proposed accident management guidelines (Ref. 46). Hitachi-GE's analysis shows that depressurisation rapidly reduces RPV pressure to well below 2 MPa (Hitachi-GE's criterion for avoiding high pressure melt ejection). This leaves the RPV in a low pressure state, but without core cooling.

222. Due to the absence of cooling, the core melts and re-locates to the core support plate and then into the lower plenum. Hitachi-GE's MAAP model predicts RPV failure about 7 hours into the accident from a failure at a control rod drive tube penetration. As a result, corium flows out of the RPV and spreads on the floor of the LDW. After evaporating any residual water, the corium starts to attack the concrete of the LDW floor and the pedestal wall. After about 16 hours, failure of the PCV head flange occurs due to over-pressurisation. In the absence of LDW flooding, MCCI continues unabated and after about 20 hours results in breach of the PCV boundary.

223. The confirmatory MELCOR calculations (Ref. 20) for the key event timings for in-vessel phenomena show reasonable agreement with Hitachi-GE's MAAP results. However, to an extent this agreement masks important differences in approaches between MAAP and MELCOR, notably in the treatment of core blockage (see Section 4.4). Accident progression with corium in the lower plenum is also subject to a number of uncertainties. Hitachi-GE's MAAP model assumes that the corium pool in the lower plenum is initially cooled by the overlying water. As this evaporates, the corium heats up and the RPV fails at a control rod drive tube penetration. The confirmatory analysis has identified differences in the treatment of lower plenum pool behaviour and pool heat transfer models between MAAP and MELCOR. Furthermore, corium chemistry effects such as eutectic steel-corium interactions in the lower plenum have significant phenomenological uncertainty and are not considered in either model. However, this analysis does broadly support the RPV failure timing predicted by MAAP. I am satisfied that the uncertainties in RPV failure timing would be unlikely to give rise to cliff-edge effects in Hitachi-GE's analysis, for example with regard to ex-vessel progression and the subsequent effectiveness of severe accident management measures.

224. I have examined the basis for Hitachi-GE's analysis of containment temperature loads in some detail. One particular concern was that Hitachi-GE's predictions of containment temperature might not take account of all heat loads during in-vessel melt progression. In response to my challenges, Hitachi-GE has provided additional evidence to support the safety case:

- A sensitivity study with a refined nodalisation (Appendix C of Ref. 44) confirms that temperatures in the PCV head gas space would be about 250°C, which is below the failure temperature of 300°C.
- Reflective metallic insulation (RMI) around the RPV and particularly the RPV head is robust against expected severe accident loads and so would be expected to remain in place (Ref. 111). The importance of the RMI for Hitachi-GE's safety case is also considered in ONR's assessment of design basis faults (Ref. 3).
- High temperatures in the RPV do not lead to failures of the reactor pressure boundary above the core, e.g. at the RHR suction line or at the RPV head spray line (Ref. 88).

- SRVs remain open and the SRV tailpipes remain intact during the core degradation phase despite high heat loads from hot gases inside the RPV (Ref. 89).

225. Based on the additional evidence provided by Hitachi-GE, I am satisfied that containment failure due to over-temperature is unlikely to occur before RPV failure.

226. After RPV failure, the superheated molten corium quickly relocates to the LDW floor where it is assumed to spread uniformly until constrained, initially, by the pedestal wall. For the unmitigated sequence, Hitachi-GE assumes the unavailability of LDW flooding. Hitachi-GE's prediction of PCV over-pressure failure at 17 hours is broadly in line with the time of 14 hours predicted by MELCOR. Whilst there are some differences in the amounts of hydrogen generated ex-vessel, the MELCOR analysis provides confidence that Hitachi-GE's modelling of accident progression ex-vessel is reasonable.

227. Hitachi-GE's analysis shows that after RPV failure the bulk LDW temperatures could exceed the containment failure criterion of 300°C. Over-temperature failure of the access tunnel hatch seals might therefore be expected at some point after RPV failure. However, I would expect heat losses to the containment structures, the restricted convective flows in the tunnels and thermal inertia of the hatches to delay failure of the hatch seals. These effects are not captured in the MAAP model. However, as discussed in Section 4.5, seal leakage due to over-temperature would be significantly less than the limiting PCV head over-pressure failure. I therefore am satisfied that for this unmitigated case, failure due to over-pressure is likely to be the primary challenge and should provide the basis for a bounding source term.

228. In the absence of LDW flooding, MAAP ultimately predicts breach of the PCV to occur due to MCCI (after more than 20 hours). Hitachi-GE argues that breach of the vent pipes before this point would flood the LDW and terminate the MCCI before this time. In Section 4.5 I have identified that there is some uncertainty regarding the effect of vent pipe break-through on the integrity of the pedestal wall. However, for this unmitigated case Hitachi-GE does identify that the pedestal wall will ultimately fail. Given that containment failure due to over-pressure occurs well before pedestal wall failure, I am satisfied that this uncertainty on the exact timing of pedestal wall failure would not have a significant effect on the accident progression.

229. In summary, I am satisfied that the analysis of the unmitigated TQUV sequence adequately describes accident progression for this case. Based on independent confirmatory analysis, the progression of core degradation, RPV failure and PCV response appear credible. As discussed in Section 4.5, there remains some uncertainty with regards to the progression of MCCI through the pedestal wall, but I am satisfied that, for the purposes of GDA, this does not affect Hitachi-GE's analysis of this unmitigated case.

4.6.3.2 RPV re-flooding

230. The initial phase of this sequence is the same as the unmitigated case, with core damage starting after 40 minutes due to absence of the Class 1 ECCS. One hour into the accident, alternative core injection using the Class 2 FLSS is assumed to be started with a minimum flow rate of 90 m^3/h. This is the equivalent of about 10% of the FLSS injection capacity of one train against low RPV pressure (Ref. 35). With this low injection rate the RPV is re-flooded and the water level recovers to TAF within about one hour. Hitachi-GE's analysis shows that further core degradation is stopped before the core support plate fails and RPV failure does not occur.

231. With regard to the effectiveness of re-flooding, Hitachi-GE's results show that hydrogen generation in the core starts at about the time of core injection initiation (Ref. 35). Whilst hydrogen generation effectively stops within 1 hour, which indicates

that the core has been re-flooded, peak core temperatures remain high. This is likely to be a consequence of MAAPs blockage model, which facilitates the establishment of a pool of molten corium despite RPV flooding. My TSC has also investigated the re-flooding behaviour for this sequence in some detail. MELCOR predicts that fuel assemblies will melt and relocate within the core, establishing blockages that impede re-flooding. However, within 1 hour after start of re-flooding, the core will be quenched and cooled. No pool of hot corium is predicted and the core support plate also remains intact.

232. Hitachi-GE has assumed a conservative RPV injection flow rate based on performance requirements for the FLSS in severe accidents. The FLSS injection delay time is not based on specific consideration of the timing of core injection, but appears to be chosen so as to reach a specific core degradation state. Based on the actual specifications of FLSS pumps (Ref. 30) and design basis injection assumptions (Ref. 32), substantially higher core injection rates would be expected in such a situation. Hitachi-GE has not investigated the effects or implications of higher injection rates, nor potential variations in the timing of core injection. However, the results clearly show that core injection can be effective in principle using a conservative scenario. I therefore accept Hitachi-GE's analysis as sufficient for GDA but would expect additional investigations by a future licensee for a better understanding of core re-flood strategy options.

233. One important consideration for my assessment of Hitachi-GE's safety case is the potential for re-criticality, which is particularly relevant for a re-flooding scenario. I have discussed this phenomenon in Section 4.4. Hitachi-GE has used the TQUV sequence as a limiting case and has claimed that there is only a brief (8 minute) time window where a re-criticality could theoretically occur (Ref. 35). Simplistically, Hitachi-GE assumes that this is the time between melting of control rods and melting of fuel.

234. The MELCOR confirmatory analysis has modelled the melting of control rods and fuel and suggests that the TQUV case considered by Hitachi-GE may not be bounding as this case has high decay heat which acts to minimise the time between control rod and fuel cladding collapse. Furthermore, MELCOR analysis confirms that the actual spatial progression of control rod and core melt through the core is complex and that there may be much longer periods where at least part of the core could have melted control rods with fuel rods largely intact. The timing and rate of re-flooding are also likely to be important, but these factors have not been explored in detail by Hitachi-GE. Whilst Hitachi-GE has identified the potential for re-criticality during re-flooding, I am not convinced that the full extent of this possibility has been explored, or that the identified 8 minute window is necessarily representative. I consider that a better understanding of the potential for re-criticality would support the development of SAMGs by the future licensee. I cover this is an Assessment Finding (AF-ABWR-SA-03) below.

235. As a counter-measure against possible re-criticality, Hitachi-GE proposes that the SLCS should be used by the operators to inject boron into the core during re-flooding. The SLCS is designed to inject boron into the core, sufficient to achieve cold shutdown, in the event of a design-basis ATWS event. I note that operation of the SLCS is part of the SAMGs provided in GDA, although the system is not a designated severe accident system and there are no formal severe accident claims. The SLCS is however a Class 2 system and is supported by the BBGs. I am satisfied this system could form part of the future licensees' strategy for minimising the risk of re-criticality during re-flooding.

236. I conclude that Hitachi-GE's evidence does not adequately demonstrate that the range of conditions leading to for re-criticality have been identified. Further analysis of the potential for re-criticality would be required to inform the development of SAMGs by the future licensee. Whilst the SLCS has been identified by Hitachi-GE as a possible mitigation measure, severe accident requirements for this system have not been

established in GDA. I am satisfied that these gaps could be addressed by the future licensee and I therefore raise the following Assessment Finding:

- AF-ABWR-SA-03: Hitachi-GE has identified the theoretical possibility of re-criticality in a severe accident during re-flooding of the reactor pressure vessel, resulting in potential challenges to the primary containment. Hitachi-GE has presented limited analysis of the conditions which could give rise to re-criticality. To inform site-specific accident management guidelines, the licensee shall perform sufficient additional analysis to identify the range of conditions that could lead to a possible re-criticality. For the conditions which could potentially result in re-criticality, the licensee shall consider the requirements for any design provisions which could reduce the risk of re-criticality so far as is reasonably practicable.

4.6.3.3 Ex-vessel corium cooling

237. Hitachi-GE claims that corium released into the LDW will be cooled by water from pre-flooding of the LDW, or by passive opening of the LDF after RPV failure. I focus my assessment on the effectiveness of the LDF, which I consider to be a more limiting case than cooling by the FLSS and FLSR active systems (which are able to deliver higher flows).

238. In the absence of a sufficiently deep water pool in the LDW prior to RPV failure, the thermal radiation from the corium will open the fusible plugs of the LDF valves (designed to open at 260°C). Hitachi-GE's analysis (Ref. 35) shows that corium can be cooled by flooding through all ten LDF valves, thereby limiting the rate of MCCI. In response to RQ-ABWR-1468 (Ref. 90), Hitachi-GE has performed an additional sensitivity analysis for this scenario where only two of the 10 LDF fusible plugs are assumed to open. The MAAP results show that the reduced flow rate has no noticeable impact on MCCI ablation depth.

239. Whilst the independent confirmatory calculations by my TSC (Ref. 20) predict radial ablation of the pedestal wall due to MCCI to be effectively stopped by the water pool above the corium, the axial ablation depths are almost five times greater than the MAAP predictions. However, the rate of axial ablation is low and this would not breach the concrete of the LDW floor or fail the pedestal wall. I therefore accept that Hitachi-GE's analysis is sufficient to demonstrate the effectiveness of the LDF in preventing containment failure due to MCCI.

240. I note that the LDF relies on the S/P water inventory above the LDF inlet line to flood the LDW. If containment heat is being removed by venting excess pressure, then water in the S/P would eventually be depleted and the water in the LDW may have to be replenished by other means. The design includes a specific injection line to the LDW which can be serviced via the FLSS or FLSR. Injection into the PCV or RPV would also eventually migrate to the LDW floor. I am therefore satisfied that water cooling of corium in the LDW could be maintained.

4.6.3.4 Containment pressure suppression and control

241. Hitachi-GE has analysed the effectiveness of active sprays in limiting containment pressures below the failure criterion. The actuation of UDW spray via the FLSS or FLSR at about 1.5×Pd (in accordance with proposed accident management guidelines) allows the PCV to be maintained below this pressure. Hitachi-GE has also demonstrated that PCV spray is effective in limiting UDW and WW temperatures below 200°C for as long as it is available. Ultimately, spray operation would need to be terminated in order to keep the WW water level below the V/Bs and to maintain the ability to vent the containment from the WW. This means that containment spray is not a means for medium and long-term pressure control. Unless heat removal via the RHR

is restored in time, filtered venting would eventually be required to remove containment heat by releasing excess pressure.

242. The UK ABWR is provided with containment over-pressure protection which is designed to relieve pressure before the ultimate failure point is reached. This is achieved by manually venting the containment before the pressure reaches 2×Pd, or through the COPS passive bursting disks when the WW pressure reaches 2×Pd. In both cases, filtered venting from the WW via the FCVS is the preferred route. There is also the possibility to vent using the unfiltered hardened vent system, but this is not Hitachi-GE's preferred strategy.

243. For slow pressure transients, Hitachi-GE's analysis shows that the vent system, once opened, is effective in limiting the pressure in the WW to below the lower-bound failure criterion of 2×Pd. This is also confirmed by the MELCOR analysis. However, due to differences in pressure between the WW and DW, pressure in the DW could exceed the COPS setting of 2×Pd before the bursting disks fail. This raises the possibility that the pressure in the DW could reach the assumed ultimate failure pressure before venting occurs. This possibility has not been addressed in Hitachi-GE's analysis. I am satisfied that optimisation of set-points will be a matter for the future licensee and so raise the following Assessment Finding:

- AF-ABWR-SA-04: Ensuring the continuing integrity of the primary containment by protecting it from over-pressurisation is a vital objective for severe accident measures and management strategies. Hitachi-GE's severe accident analysis has shown that the assumed set-point for the containment overpressure protection system would not always ensure that pressure in the drywell remains below the containment ultimate failure pressure. For accident sequences where venting is claimed as an effective severe accident measure, the licensee shall optimise the containment over-pressure protection system opening set-point to ensure that containment pressures remain below the ultimate failure pressure so far as is reasonably practicable. This shall take into account containment conditions in severe accidents, including consideration of potential static and dynamic pressure differences between the drywell and wetwell.

244. Hitachi-GE's analysis of venting assumes that the systems will be designed to release an amount of steam corresponding to 1% decay heat power at a PCV pressure of 1×Pd. I am satisfied that the detailed design of the vent systems can be carried out after GDA, but expect the future licensee to demonstrate that the system design will be able to deliver the required performance. I therefore raise the following Assessment Finding:

- AF-ABWR-SA-05: In the absence of detailed design information during GDA, Hitachi-GE has made assumptions about achievable flow rates in its demonstrations of the effectiveness of primary containment vessel venting in severe accidents. The licensee shall demonstrate that the final design of the filtered containment vent system can meet the safety case claims placed on it by those severe accident sequences which credit venting.

245. In summary, I am satisfied that containment spray is likely to provide an effective means of pressure suppression for this sequence and that the COPS provides ultimate protection against slow pressurisation transients by passively venting the containment to atmosphere.

4.6.3.5 Recovery of core and containment cooling

246. Hitachi-GE uses the TQUV sequence to demonstrate the effectiveness of the RHR system in recovering core and containment cooling. The RHR system (which is part of the ECCS) could be recovered by:

- recovery of a Class 1 EDG;
- operation of the DAG to provide power to one division of the ECCS;
- operation of mobile power for one division of the ECCS, or
- use of the AHEF, in conjunction with a mobile power source.

247. In this case, Hitachi-GE assumes that the RHR system is restored after RPV failure (with corium in the LDW), but before PCV failure due to over-pressurisation. Core injection is provided by the LPFL mode using suction from the S/P. This provides cooling to corium in the LDW (via the RPV break) and the resulting steam is condensed in the S/P. The RHR provides S/P cooling and the UDW spray to control PCV pressure and temperature, without the need for venting.

248. Whilst this case was not considered in the confirmatory analysis, I am satisfied that the basis for Hitachi-GE's calculations seems reasonable and that this provides demonstration of how long term cooling after a severe accident would be achieved.

4.6.4 Station blackout scenario

4.6.4.1 Unmitigated case

249. The unmitigated TB sequence is characterised by a loss of active cooling functions due to SBO. Initially the RCIC remains functional and maintains RPV water level, while the safety relief function of the SRVs limit RPV pressure by discharging steam to the S/P. Failure of the DC batteries is assumed at 8 hours, although in reality this is likely to be conservative because post-Fukushima the RCIC battery duration has been extended to 24 hours. After termination of the RCIC, RPV water level drops and core degradation begins at about 12 hours. One feature of this case compared to the TQUV sequence is that decay heat at the time of core damage is much lower. The corium re-locates to the lower plenum and the RPV subsequently fails at high pressure at about 19 hours. Although LDW flooding is not assumed for the unmitigated case, a significant quantity of water is predicted to be present in this scenario due to overflow of saturated water from the S/P into the LDW, caused by long-term operation of the RCIC. Heat from DCH and steam from ex-vessel FCI, supplemented by hydrogen from further zirconium oxidation, is assumed by Hitachi-GE to result in immediate over-pressurisation and failure of the containment at the PCV head flange.

250. The uncertainties in development of the core degradation phase discussed for the TQUV case are also relevant for this sequence. The MELCOR confirmatory analysis (Ref. 20) predicts that during core degradation and relocation about three times the amount of hydrogen predicted by MAAP. This is explained by the relatively flat temperature profile in the core (due to lower decay heat) in conjunction with the MELCOR blockage model. This means that there is no steam starvation during the prolonged core degradation phase and a large part of the zirconium and steel is oxidised in-core. This hydrogen is transported to the S/P via the SRVs before RPV failure. The large influx of non-condensable gases leads to an increase of the PCV pressure from about 0.5×Pd to 1.5×Pd within less than 1 hour and well before RPV failure. However, it is important to consider this in the context of the total amount of hydrogen produced in-vessel and ex-vessel, which is broadly comparable between MAAP and MELCOR predictions for this scenario.

251. The RPV failure time of 19 hours predicted by MELCOR is comparable to that predicted by Hitachi-GE. After RPV failure, the corium is ejected at high pressure into the water pool in the LDW. Hitachi-GE's results confirm that rapid steam generation is particularly violent for HPME resulting from RPV failures at high pressure. FCI in this scenario is expected to be more onerous than for the low pressure failures due to increased fragmentation of the corium jet and the associated enhanced heat transfer with water that may be present in the LDW. The results show that containment failure due to over-pressure is unlikely to be prevented by the COPS. In Ref. 44, Hitachi-GE

has also considered the effect of direct debris interaction on the containment structures using CFD to consider the distribution of corium debris onto LDW components such as the access tunnels and hatches and the V/Bs. This analysis indicates that heat transfer from debris to the components would not be expected to result in high temperature failures of the PCV liner or components.

252. If the LDW were to be dry at the time of high pressure RPV failure then Hitachi-GE identifies that DCH would occur, but there would be no associated steam generation from the LDW pool. Pressure rise in the PCV will be driven by equalisation of pressures from the LDW through the connecting vents which suppress the rate of pressure rise in the UDW and WW. In this case Hitachi-GE's MAAP analysis shows that the resulting pressure rise from DCH does not challenge the containment integrity in the short-term and in the long-term could be controlled by manual venting or the COPS.

253. It is important to consider whether further containment failures modes, after the initial failure due to over-pressure, could become relevant. I am satisfied that Hitachi-GE's results demonstrate that the water in the LDW from S/P spill-over initially protects against MCCI. However I also observe that MCCI starts several hours after the water in the LDW has been evaporated. About two days after the start of the accident, the ablation due to MCCI is predicted to have progressed through to the vent pipes in the pedestal wall. I have already discussed uncertainties in the progression of unmitigated MCCI through the pedestal wall, but I am satisfied that in this case any uncertainties will be of lesser concern due to the longer elapsed time and lower decay heat rate.

4.6.4.2 RPV depressurisation

254. Hitachi-GE's safety case for this and similar sequences is based on the fundamental premise that the reactor can be depressurised using the SRVs, by way of the Class 1 ADS, the Class 2 RDCF or by manual operation using switching valves. Hitachi-GE's mitigated TB sequence considers reactor depressurisation when the water level in the core has dropped to 20% above BAF, reflecting Hitachi-GE's proposed accident management guidelines. Hitachi-GE claims that HPME is prevented if the RPV is depressurised to below 2 MPa prior to RPV failure (Ref. 24), but this value is not substantiated further. However, based on Hitachi-GE's analysis and that of my TSC, I am satisfied that, provided the SRVs can be operated, then they should be capable of rapidly controlling the reactor to a low pressure state, thus preventing the containment challenges associated with high pressure failures.

255. As discussed for the unmitigated case, Hitachi-GE has identified a high risk of a consequential containment failure if the reactor cannot be depressurised before RPV failure occurs. Although severe accident claims on the RDCF are cited in Hitachi-GE's safety case for the justification of the SRV design, these are not linked to relevant performance requirements. For example, the temperature resistance of the valves for severe accident conditions are not specifically included in the design documentation for SRVs. I consider this finding in the wider context of Hitachi-GE's safety case in Section 4.11.

256. During some severe accident scenarios, containment pressures above 1.5×Pd can be reached before the RPV fails. For example, Hitachi-GE's severe accident strategy for pressure control in the PCV assumes that containment spray (if available) is used to limit PCV pressure to about 1.5×Pd and filtered venting is initiated at the very latest if PCV pressure reaches 2×Pd (the COPS set-point). At PCV pressures above approximately 1.75×Pd, the pressure capacity of the RDCF accumulators is insufficient to keep the reactor depressurised to the PCV pressure. At this point, SRVs would start to cycle. As each SRV cycle uses a substantial amount of nitrogen, this could deplete the affected accumulators. The situation can be controlled either by operators connecting gas cylinder racks in the R/B and using the switching valves for the SRVs

(although frequent SRV cycling would still be a drain on nitrogen supplies) or by initiating PCV venting early and keeping PCV pressures below 1.5×Pd. Whilst these considerations are not reflected in Hitachi-GE's generic accident management guidance, I am satisfied that these potential scenarios could be addressed in the development of SAMGs by the future licensee.

257. Given the importance of RPV depressurisation, I would expect the SAMGs to encourage the operators to explore all possible means to depressurise the reactor through any available means. There should be emphasis on the need to manage nitrogen accumulator capacity and the importance of restoring nitrogen supplies from the high pressure nitrogen gas supply system (HPIN). If the reactor cannot be depressurised, then containment conditions would need to be managed to minimise the risk of containment failure at the time of RPV breach. This could include consideration of criteria for deciding when short-term containment failure after high pressure RPV failure should be assumed and what mitigation measures should be prioritised. For example, the water level in the LDW at the time of RPV failure could be a significant factor which affects the severity of the ex-vessel FCI. I would expect that the future licensee to consider these matters in the development of the SAMGs.

4.6.4.3 RPV re-flooding

258. Similar to the TQUV re-flooding case, Hitachi-GE makes the same conservative assumptions for the rate of low pressure injection, which would be from the FLSS or the FLSR. Hitachi-GE's MAAP analysis demonstrates that the damaged core is re-covered without progression to core support plate failure.

259. I have already made comments in relation to the potential for re-criticality during re-flooding. These comments also apply to the TB re-flooding case. The confirmatory analysis also considered the core degradation sequence for the TB sequence where core damage occurs after about 10 hours, which is much later than for the TQUV case analysed by Hitachi-GE. The MELCOR results show that the lower decay heat and the more homogeneous temperature profile prolong the time window between collapse of control rods and fuel assemblies. This suggests that Hitachi-GE's consideration of re-criticality risk using the TQUV re-flood scenario may not be bounding. I have already raised AF-ABWR-SA-03 in relation to this.

4.6.4.4 Containment pressure control

260. For this case, Hitachi-GE has claimed that rapid containment pressurisation due to ex-vessel FCI does not pose a challenge to PCV integrity provided that the RPV can be successfully depressurised before RPV failure. Hitachi-GE argues that this is because the condensation in the S/P in combination with the COPS, which is designed to open passively at 2×Pd, would be effective in limiting the PCV pressure rise.

261. Containment pressure response to FCI would be a short-term pressure spike lasting until the rate of steam condensation in the S/P exceeds the steam generation rate. Hitachi-GE has provided analysis (Ref. 55) for a representative case which shows a momentary pressure spike rising from an initial pressure of about 1.5×Pd to about 2.25×Pd. This indicates that the pressure peak just exceeds the assumed ultimate failure pressure of 2×Pd, although I note that this would still be below Hitachi-GE's best-estimate failure pressure. Thereafter, containment pressure quickly drops to below the ultimate failure pressure due to relief through the COPS. Hitachi-GE argues that the results for this representative case assume a conservatively large corium ejection mass flow modelled in MAAP and that the peak steam generation rate would actually be lower.

262. A key factor affecting the peak pressure is the initial containment pressure at the time of RPV failure. This is relatively high for the TB case due to the late time to RPV

failure. If a fast pressurisation due to FCI event were to happen at PCV pressures near the current COPS set-point, Hitachi-GE's analysis suggests that PCV integrity could be compromised. In this respect, I would expect accident management to be based on a good understanding of which combinations of PCV pressure, PCV water levels and RPV pressure would be susceptible to containment failure due to fast pressurisation. I would expect the future licensee to consider measures to manage the risk of FCI in the development of SAMGs.

263. In terms of longer-term pressurisation, Hitachi-GE's analysis for this sequence shows that conditions in the containment can be effectively managed using sprays and venting to prevent PCV failure. This is confirmed by MELCOR analysis of this scenario. I am satisfied that Hitachi-GE has demonstrated the effectiveness of these measures for long term pressure suppression and containment heat removal.

4.6.5 Hydrogen management

4.6.5.1 Assessment overview

264. In this section I assess Hitachi-GE's arguments and evidence that hydrogen control measures would be effective in preventing a flammable atmosphere within the primary containment or R/B. This builds on my assessment of Hitachi-GE's success criteria for hydrogen control in Section 4.5.

265. My assessment has primarily considered the adequacy of UK ABWR's design features and proposed management strategies for flammable gases generated in reactor and SFP severe accident scenarios. However, I have broadened the scope of my assessment to consider how the same design features and strategies are also used in design basis fault conditions. I have not considered the off-gas system where hydrogen could be present during normal operation; these aspects have been considered by other colleagues.

266. Hitachi-GE has presented two key documents (Ref. 42 & 43) covering flammable gas modelling and ALARP considerations; these form the basis of the safety case summarised in PCSR Chapter 26 (Ref. 24) and are the subject of my assessment.

4.6.5.2 Assessment of design basis faults

267. Hitachi-GE claims that reactor faults within the design basis do not result in excessive oxidation of fuel cladding and therefore hydrogen generated from this effect is not significant. However, fault conditions could still result in the generation of relatively small amounts of hydrogen and oxygen in the PCV due to radiolysis of water. Under normal operation, radiolysis gas would be removed by the off-gas system, but this system is assumed to be unavailable in a post-fault condition. Hitachi-GE claims that radiolysis gases do not result in a risk of hydrogen combustion in the PCV because the UK ABWR operates with an inerted primary containment and has effective hydrogen control measures. I consider operation with a de-inerted containment in Section 4.6.5.5.

268. Even without benefit of the PARs, Hitachi-GE has calculated (Ref. 42) that a flammable concentration would only be reached in the PCV after about 104 hours. I note that Hitachi-GE assumes no post-fault recovery of PCV atmospheric control measures. Hitachi-GE has also analysed the effectiveness of the PARs in the PCV assuming provisional performance characteristics. The analysis shows that with the proposed installation of five PARs in the PCV the hydrogen concentration arising from radiolysis is limited to well below the flammable limits (Ref. 42). In Ref. 66, the Reactor Chemistry assessor has also considered relevant aspects of Hitachi-GE's analysis and raised two Assessment Findings on radiolysis gas production yield rates (AF-ABWR-RC-22) and the performance of the PAR units under fault conditions

(AF-ABWR-RC-23). Hitachi-GE has stated (Ref. 38) that the PAR units will be selected at the detailed design stage and that the locations and performance characteristics will be confirmed as being adequate at that point. I agree with the Reactor Chemistry assessor that these issues do not undermine Hitachi-GE's concept of using PARs in the PCV and the position is adequate for GDA. However, to ensure that the final design is reflected in the future licensee's safety case, I raise the following Assessment Finding:

- AF-ABWR-SA-06: Hitachi-GE's GDA analysis of the effectiveness of hydrogen management measures in the primary containment and reactor building has been based on provisional design information for passive autocatalytic recombiners. The analysis supports Hitachi-GE's hydrogen management strategy for design basis loss of coolant accidents and reactor severe accidents. The licensee shall update the hydrogen management safety case to reflect the design and performance characteristics of the recombiners selected in the final design, and reconfirm that the hydrogen management objectives are met.

4.6.5.3 Severe accidents involving the reactor at power

269. For severe accidents, Hitachi-GE identifies that the principal strategy is to prevent flammable gas generation by providing water injection for fuel cooling, therefore preventing or mitigating fuel damage. Diverse active systems are provided for this, including the ECCS, FLSS and FLSR amongst others. If injection for cooling is successful before significant fuel damage occurs, then hydrogen generation due to oxidation will be avoided, or at least limited. However, significant amounts of hydrogen would be generated in a severe accident as a result of steam oxidation of hot cladding material within the core and MCCI, and so the UK ABWR incorporates a number of flammable gas control measures to prevent combustion:

- During normal operation, the PCV is inerted with nitrogen, such that the oxygen concentration is below 4%vol. This means that even if hydrogen is generated in a severe accident, there is insufficient oxygen to support combustion.
- For some accidents, PCV venting is conducted to prevent containment overpressure. This would result in venting of flammable gases and re-inerting of the containment with steam. The ANI system is also available post-venting to supporting re-inerting of the PCV.

270. Hitachi-GE does not claim that the five PARs in the PCV would be effective in removing hydrogen for severe accidents since these are only provided to protect against radiolysis during design basis accidents. This is because relatively large quantities would be generated, beyond that which could be practically removed using the PARs. However, because the PCV is inerted, Hitachi-GE claims that there is no risk of a deflagration in the PCV. In accident scenarios the PCV would also be saturated with inert steam. Oxygen would be generated in the PCV from radiolysis and removed by PARs, but even if the available PARs were ineffective, this would take some time to reach levels that would support combustion. In the absence of recovery of containment cooling, most severe accidents are likely to require management of containment pressure by venting. This provides an effective means of reducing the mass of hydrogen present in the PCV. I accept Hitachi-GE's argument that hydrogen combustion in the PCV is not the primary concern for severe accidents.

271. In a severe accident, there is the potential for flammable gases to leak from the PCV into the R/B. In this case the strategy is to prevent hydrogen combustion in the R/B, thereby preventing damage to SSCs and the SFP which might otherwise jeopardise recovery actions. Hitachi-GE has identified the following measures for managing flammable gases in the R/B:

- maintenance of a leak-tight PCV boundary to limit leakage to the R/B;
- operation of the SGTS to ventilate the R/B (subject to availability of power);
- installation of 29 PARs on the reactor service floor of the R/B; and
- establishment of a natural ventilation route in the R/B by opening of a blowout panel in the R/B wall and a door at ground level.

272. The PCV is designed so that the leakage rate is less than 0.4% per day at 90 % of the maximum design pressure (Ref. 28). Hitachi-GE estimates a 'realistic' leakage rate of 1.5% per day under accident conditions, but has considered the effectiveness of the PARs and the SGTS assuming a rate of 10% per day (Ref. 42). Leakage could also occur at an enhanced rate if the containment fails, for example, due to exceeding over-pressure/temperature limits leading to failure of the PCV head flange, but there are measures to protect against such failures. I am satisfied that Hitachi-GE's assumptions are reasonable basis for considering hydrogen leakage into the R/B.

273. Hitachi-GE's analysis shows that one of the two SGTS trains would be effective in preventing flammable concentrations in the R/B for realistic PCV leakage rates. I am not fully convinced that Hitachi-GE's simplistic mixing model is sufficient to demonstrate effectiveness of the SGTS by itself, although it would be expected to facilitate mixing to support the PARs.

274. The effectiveness of PARs and the blowout panel have been analysed by Hitachi-GE using a GOTHIC model of the UK ABWR. This has been used to model the transport of hydrogen in the R/B and predict concentrations for comparison with flammable limits. I have not reviewed this model in detail. However, I do recognise that GOTHIC is an established code for this type of calculation (see Section 4.3). I consider that Hitachi-GE's nodalisation is sufficient to represent the reactor service floor. This location is the main area of interest for simulating the distribution of hydrogen released from the PCV head flange, or an open RPV to the reactor service floor, and also for simulating flows through the R/B blowout panel. Hydrogen transport in lower floors of the R/B are likely to be less well represented by Hitachi-GE's model, but leakage rates into these areas are expected to be less significant than for leaks into the reactor service floor.

275. Hitachi-GE has considered bounding PCV leakage assumptions based on a conservative leakage rate for the PCV of 10% per day and an assumption that leakage occurs from the PCV head at the PCV ultimate pressure (although COPS would be expected to activate before reaching this pressure). Hitachi-GE's calculations assume leakage at the PCV head before failure and therefore measures to prevent containment failure are a key part of the hydrogen management strategy. For this case Hitachi-GE assumes that the PCV atmosphere contains 35% hydrogen by volume, equivalent to oxidation of 100% of the cladding. No allowance is included for oxidation of steel in the RPV or MCCI, but overall I consider that this set of assumptions is likely to be bounding in terms of hydrogen concentrations in the R/B. Hitachi-GE's results show that concentrations in the R/B reach a maximum of 4% by volume on the reactor service floor (and lower elsewhere); this is below the concentration of 5% required for combustion. For a more realistic case, Hitachi-GE has chosen PCV conditions and gas composition from detailed MAAP analysis and considered leak locations from all of the hatch and air-lock locations. This realistic case indicates that all concentrations in the R/B would be below 1% by volume.

276. I acknowledge that Hitachi-GE's analysis will be subject to uncertainties, in particular the ability to capture local effects which would be beyond the resolution of the model. I also note that the Reactor Chemistry assessment has resulted in an Assessment Finding (AF-ABWR-RC-23) in relation to Hitachi-GE's modelling of the mixing of hot gases from the PAR exhaust and the effect these could have on SSCs. Whilst the detailed arrangement of PARs in the R/B will need to be addressed by a future licensee, I am satisfied that these issues do not invalidate Hitachi-GE's concept. For

the purposes of GDA, I consider that there has been an adequate demonstration that PARs could be an effective measure for controlling hydrogen from PCV leakage. However, the future licensee will need to ensure that the final PARs design is reflected in the hydrogen management safety case. This is covered AF-ABWR-SA-06.

277. Hitachi-GE states (Ref. 38) that the actual equipment specification will only be decided during licensing, which I accept. Whilst I am satisfied that the assumptions made by Hitachi-GE are sufficient to demonstrate the credibility of PARs as part of the hydrogen management strategy, the supporting analysis will need to be re-visited once the licensee has decided on specific equipment locations and technology.

278. Hitachi-GE has extended the GOTHIC model to include natural ventilation promoted by flow through the blowout panel and R/B door. Results show that this would also be an effective measure for limiting hydrogen concentrations in the event of PCV leakage. I note that the case with hydrogen release during shutdown and SFP severe accidents is a more limiting case for the blowout panel than PCV leakage; I have therefore focussed my assessment of the blowout panel on shutdown and SFP accidents.

4.6.5.4 Severe accidents involving a shutdown reactor or the SFP

279. There is the potential for flammable gas to be released directly into the R/B for severe accidents in certain shutdown modes when the PCV boundary is open (for example when the RPV head is removed to permit refuelling). Similarly, this could also occur for severe accidents involving fuel in the SFP. Again, Hitachi-GE identifies that for the SFP and the reactor at shutdown, adequate cooling can be achieved by ensuring that fuel remains covered by water. Even without water addition, Hitachi-GE calculates substantial times to uncovery of fuel and hydrogen generation – see Section 4.9.

280. Nonetheless, Hitachi-GE has further analysed the consequences of postulated damage to fuel and release of hydrogen directly into the R/B. In this case, Hitachi-GE has identified that PARs and SGTS would be ineffective due to the high rate of hydrogen release. For these situations, Hitachi-GE claims that natural ventilation of the R/B, promoted by opening of the blowout panel at high level in the R/B wall and a door at ground level, provides effective mitigation by limiting the hydrogen concentration.

281. Hitachi-GE has considered transient hydrogen and steam generation rates / concentrations based on MAAP analyses. Three severe accident cases have been considered:

- reactor at shutdown with PCV and RPV heads open;
- reactor at shutdown with PCV open, but RPV head closed; and
- the SFP.

282. In all three cases the analysis indicates that a flammable concentration could just be reached in the R/B (and in one case just exceeded) for a very short period of time coinciding with the peak hydrogen release rate. However, Hitachi-GE argues that the concentrations would be well below detonation limits, which I accept. For GDA, I am therefore satisfied that Hitachi-GE has demonstrated that the blowout panel, in conjunction with the equipment door, could be an effective hydrogen control measure.

283. As discussed in Section 4.5.2.2, there would be a prolonged period of steam generation before a fault would progress to a severe accident with generation of hydrogen. The intention is that the blowout panel would open automatically in response to the generation of steam from water boiling in the shutdown reactor or the SFP. If the fault progressed to a severe accident then the panel would remain open for hydrogen control. Fault Studies colleagues have considered the proposed arrangements for opening of the blowout panel for steam generation faults and an Assessment Finding AF-ABWR-FS-10 has been raised (Ref. 3). The scope of this

Assessment Finding includes a requirement for the future licensee to confirm opening set-points in relation to both steam generation and management of a severe accident.

284. In addition to the blowout panel, opening of the R/B equipment door is the last step of Hitachi-GE's strategy for managing hydrogen. This would only be required if all available measures for keeping fuel covered with water had failed. Details of how the equipment door could be opened have not been provided in GDA. It is not clear how the door would be opened, for example whether power sources would be required to operate any door mechanisms, or how long it would take. It is also unclear how the door will be opened (for example remotely) and whether workers would need to be protected. There also needs to be consideration of how operators in the MCR or B/B would know actions had been performed correctly and that the measure was effective. Whilst I am satisfied that there is a credible concept for GDA, I consider that this will need to be addressed by a future licensee in the detail design. I therefore raise the following Assessment Finding:

- AF-ABWR-SA-07: Hitachi-GE has identified in GDA a need to open the large equipment door in some severe accident conditions as part of the hydrogen management strategy. However the practicalities of how this will be done have not been determined due to limitations in GDA scope. The licensee shall determine the arrangements for opening of the reactor building large equipment door in accident conditions, taking appropriate steps to ensure that the risks to both the public (from a major event escalation caused by not opening the door) and workers performing crucial tasks are considered and reduced to ALARP.

4.6.5.5 Periods of de-inerted operation

285. Hitachi-GE has identified that there will be short periods immediately prior to outage shutdown, and then during restart, where the reactor will be operated at power with a de-inerted containment. Hitachi-GE states that this is to permit entry of personnel into the PCV to carry out inspection activities with the reactor sub-critical, but in a hot standby state. The de-inerted period accounts for less than 0.3% of an operating cycle (Ref. 43).

286. Hitachi-GE claims that there are benefits from performing inspections and that this offsets any increase in nuclear safety risk associated with de-inerted operation. My assessment has been restricted to consideration of the nuclear safety significance of operation with a de-inerted containment. I have not judged the possible benefit of these inspections or whether such benefits could be achieved by other means. Furthermore, I have not considered whether there would be any additional radiological or conventional health and safety risks to personnel entering the de-inerted containment with the reactor in a hot standby state.

287. For design basis accidents occurring during the de-inerted period, Hitachi-GE has demonstrated that the PARs in the PCV would be effective in limiting the hydrogen concentration below the flammable limit. Even without PARs, Hitachi-GE's analysis shows that it would take about 100 hours for flammable limits to be exceeded. Hitachi-GE considers that the PARs would start to remove excess hydrogen once the assumed starting threshold of 1.5% is reached (Ref. 42); the analysis is therefore essentially the same as the for inerted state. For the purposes of GDA, I am therefore satisfied that there is an adequate hydrogen management case for design basis faults with a de-inerted containment.

288. For severe accidents in the de-inerted state, a flammable concentration could occur at the time of fuel cladding damage. For this situation, Hitachi-GE states that the PARs in the PCV would be ineffective and there would be potential for combustion to occur. Hitachi-GE has presented a PSA sensitivity based on the assumption that for the de-inerted case combustion always occurs resulting in failure of the PCV. This shows that

there would be an associated 0.5% increase to the large release frequency compared to a base case assuming the containment to be always inerted. I consider that Hitachi-GE has made an adequate argument that the proposed short period of de-inerted operation would not have a significant effect on the overall PSA results.

289. Whilst I am satisfied that Hitachi-GE has presented an adequate justification of the accident risks for the proposed short period of de-inerted operation, there will still be a need for the future licensee to demonstrate that radiological and conventional health and safety risks associated with inspections in a de-inerted containment can be managed. I would expect this to be considered as part of normal business during licensing.

4.6.6 Assessment summary

290. In this section I have considered Hitachi-GE's analysis of severe accidents based on a sample of reactor accident sequences. For unmitigated sequences, I conclude that the timings for accident progression (core support plate failure, RPV failure and containment failure) are credible. For the purposes of GDA, I am satisfied that Hitachi-GE's analysis of unmitigated sequences provides an adequate basis for considering the effectiveness of severe accident mitigation measures.

291. In its analysis of mitigated sequences, Hitachi-GE has provided an adequate demonstration that severe accident measures would be effective in preventing and mitigating the effects of accidents. In particular, I am satisfied that Hitachi-GE has presented evidence to support its conclusions that:

- Re-flooding of the RPV using the Class 2 FLSS or Class 3 FLSR before failure of the core support plate is likely to be effective in terminating progression of core melt and restoring core cooling.
- In the absence of core cooling, reactor depressurisation prior to RPV failure would be effective in avoiding a high pressure melt ejection scenario.
- Flooding of the LDW using FLSS or FLSR, or the passive Class 3 LDF system, would provide adequate cooling of ex-vessel corium, sufficient to prevent containment failure due to MCCI. In the event of a pre-flooded LDW, an ex-vessel steam explosion would be unlikely to result in containment failure.
- Containment sprays using the FLSS or FLSR are sufficient to provide suppression of containment pressure in the short-term.
- If RPV failure at high pressure is avoided, PCV venting (whether initiated manually or passively by the COPS) is expected to provide effective protection against containment over-pressure.
- Measures for hydrogen control would be effective in preventing hydrogen combustion in the R/B.
- Recovery of the RHR system, supported by mobile equipment as required, is sufficient to return the plant to a safe, stable state.

4.7 Severe accident management

4.7.1 Background

292. A high level overview of the strategy for severe accident management is presented in Chapters 22 and 26 of the PCSR (Ref. 22, 24). Hitachi-GE states that the strategy will focus on preventing and/or mitigating severe accident progression, including failure of reactor fuel and preventing containment failure to ensure no large quantities of fission products are released to the environment. Hitachi-GE has determined that the submission of detailed procedures and guidance to support the UK ABWR accident management strategy is outside of the scope of GDA. However, a framework for the development of procedures and guidance by a future licensee has been presented.

This has been supported by examples of guidance developed for GDA based on practice for Japanese ABWR plants (Ref. 46, 91, 92, 93 & 94).

293. Hitachi-GE proposes a framework for emergency and accident management procedures / guidance based on:

- abnormal operating procedures (AOPs);
- emergency operating procedures (EOPs); and
- severe accident management guidelines (SAMGs).

294. The SAMGs provide guidelines for mitigating accident scenarios in which severe core damage has occurred, and where failure of the RPV and/or challenge to the containment integrity may follow. PCSR Chapter 22 provides further insights on how the procedures and guidelines might be developed by a future licensee, for example by building on the work of the BWR Owners Group. Chapter 22 expects that more detailed technical support guidance would be developed by a future licensee to support the SAMGs.

4.7.2 Severe accident management guidelines

295. SAP AM.1 expects that strategies and plans are in place to prepare for and manage severe accidents. Hitachi-GE's overall accident management strategy encompasses design features of the plant that provide resilience against severe accident conditions, pre-installed engineered safety functions and mobile equipment. It also includes accident management guidance for operators. These measures are needed to robustly implement Level 4 of defence-in-depth (SAP EKP.3) and are also essential for achieving practical elimination of large and early releases.

296. I accept Hitachi-GE's position that detailed procedures and guidance in this area are not required for GDA. However I do expect that there should be a clear strategy on which to base future development of the procedures and guidance. Furthermore, my expectation is that, at a fundamental level, severe accident progression in the UK ABWR should be understood and that there is a credible SAMG concept that could be taken forward by a future licensee.

297. I have restricted the scope of my assessment to the SAMGs and the entry point to these from the EOPs. Events which remain in the EOPs do not involve fuel damage and therefore do not fall within the scope of my severe accidents assessment. EOPs and AOPs will be matters of normal business after GDA.

298. I note that there is international guidance on the development of accident management guidelines including IAEA NS-G-2.15 on severe accident management programmes for nuclear power plants (Ref. 9) and Issue LM of the WENRA Reference Levels (Ref. 10). I take cognisance of this guidance, but consider that a detailed assessment against these expectations needs to be part of the licensing process, not GDA.

299. The focus in the EOPs (Ref. 91) is on maintaining fuel cooling by injection of water into the RPV to maintain an adequate level in the core. However, failure to maintain level above TAF, coincident with detection of abnormal radiation levels in the PCV, is taken as the prompt in the EOPs that core damage has occurred and signals the transition to the SAMGs. I am satisfied that these circumstances provide a reasonable basis to define the interface between the EOPs and SAMGs.

300. Hitachi-GE's SAMG for the reactor (Ref. 46) describes the key actions, which are to:

- restore cooling to the damaged core;
- provide containment heat removal and PCV pressure control, by the use of sprays and if necessary by venting;

- if the reactor is still in a pressurised state and RPV failure is anticipated, then depressurise the reactor to avoid HPME;
- provide injection to the RPV after failure and to LDW for corium cooling; and
- switch to normal cooling systems such as RHR as these become available.

301. The guidelines are at a relatively high level and, given their origin, do not reflect the detail of the UK ABWR severe accident measures. However, I am content that these guidelines capture the key aspects of severe accident progression for the UK ABWR and provide a credible basis for further development by a future licensee.

302. I am also satisfied that the guidelines provide an adequate basis for the key operator actions assumed in Hitachi-GE’s severe accident analysis (Ref. 35), for example in relation to the criteria for initiation of sprays, venting and reactor depressurisation.

303. I note that the Process Radiation Monitoring System (Ref. 85) and the severe accident C&I (Ref. 87) include provision for monitoring of radiation levels in the containment to support the EOP/SAMG transition. Measurement of RPV water level is also a key parameter for both the EOPs and SAMGs. The diversity of the water level instrumentation and performance under severe accident conditions has been subject of other ONR assessments, with the conclusion that Hitachi-GE’s design is ALARP.

304. Hitachi-GE recognises that SAMGs need to extend beyond the operating reactor and in Ref. 45 has identified this as one of the learning points from the 2015 IAEA OSART mission to Kashiwazaki-Kariwa Units 6 & 7. In relation to the SFP, Hitachi-GE has provided a procedure (Ref. 93) for the monitoring and control of water level in the pool in response to loss of cooling and loss of coolant faults. In this case the accident management strategy is straightforward - prevention of a severe accident can be achieved by provision of sufficient water to make-up losses due to pool boiling or from leakage. The procedure identifies that fixed SFP sprays provide cooling of fuel even if the pool level drops below TAF. Hitachi-GE has also provided a concept (Ref. 94) for managing accidents during shutdown with the reactor at atmospheric pressure and the RPV head removed. Again, the accident management strategy is to introduce alternative cooling and make-up to cover the fuel. I note that the UK ABWR design includes a range of independent and diverse methods for providing water make-up to the SFP and shutdown reactor.

305. Hitachi-GE has identified a number plant damage states which result either in containment failure before core melt or immediately after. Hitachi-GE’s PSA has also identified accident sequences where severe accident measures are assumed to fail, resulting in containment failure. The radiological consequences of these scenarios are included in Hitachi-GE’s Level 3 PSA and therefore contribute to the overall levels of risk that are predicted for the UK ABWR design. For these sequences, the focus of the design should be on prevention of the plant damage states by robust plant design and a demonstration that risks are reduced to ALARP by design features which prevent the damage state. However, I would still expect that the effectiveness of measures to reduce on-site and off-site doses needs to be considered for the purposes of accident management. I am satisfied that this would be best addressed by the future licensee during the development of site-specific SAMGs by the future licensee.

4.7.3 Internal and external hazards

306. Given the accidents at Fukushima Dai-ichi, I also give specific consideration to whether Hitachi-GE’s accident management concept adequately addresses internal and external hazards.

307. As part of the work to address RO-ABWR-0023 Action 2.2 (Ref. 48), Hitachi-GE has presented its severe accident safety case for extreme hazard impacts in Appendix H of Ref. 35. Hitachi-GE has identified earthquakes, internal fire and flooding, tornado and

turbine missiles, aircraft impact and external flooding as relevant hazards. For each hazard, Hitachi-GE has characterised potential negative impacts on the containment structures and other SSCs that would be used for severe accident management.

308. The UK ABWR includes the provision of a severe accident capability in the B/B (Ref. 86). The B/B systems are physically separated (to the extent this is possible), and independent, from the Class 1 systems in the R/B. The overall strategy is to use mobile equipment if B/B systems fail, or are unavailable, in a severe accident. The proposal for the UK ABWR is that mobile equipment will be stored in a garage, protected as far as reasonably practicable, from beyond design basis and severe accidents events (Ref. 22). The location of the mobile equipment garage will be determined for the specific site. I am satisfied with Hitachi-GE's case that these measures should provide robust protection against internal hazard events.

309. Earthquakes and external flooding events could have a particularly significant impact on the feasibility of accident management measures. For example, for an extreme seismic event engineered features such as the FLSS could be damaged and accident management could be negatively impacted by damage to structures and limited accessibility. Similarly, flooding of the site would restrict access to mobile resources and could potentially also fail systems located in the B/B. For seismic and flooding, I judge that Hitachi-GE's basic management concept will still be feasible in principle as catastrophic damage to the R/B, B/B and loss of mobile accident management resources can only occur in very extreme scenarios. I am satisfied that, for the purposes of GDA, Hitachi-GE's submission is sufficient to address the expectations in Stress Test Finding STF-3 (Ref. 15) on the impact of external hazards on required operator actions.

310. I note that Hitachi-GE has proposed that the B/B will be located above the elevation of the R/B ground floor, subject to site-specific considerations. Resilience to flooding in general has been considered by the External Hazards assessor through RO-ABWR-0067 (Ref. 95). As a result, an Assessment Finding (AF-ABWR-EH-04) has been raised (Ref. 96) in relation to water sealing and/or elevation of the key site buildings (including the R/B and B/B). This Assessment Finding will require a site-specific evaluation of external flooding beyond the design basis.

311. I am satisfied that Hitachi-GE's concept based on the B/B and use of mobile equipment is consistent the IAEA expectations post-Fukushima (Ref. 16) on implementation of the defence in depth. Hitachi-GE's approach provides independence, redundancy and diversity and provides protection against internal and external hazards.

312. In conclusion, I consider that severe accident features of the generic design demonstrate resilience against internal and external hazards. I accept that the development of accident management guidance that addresses site-specific specific challenges is out of the scope of GDA. I am satisfied that this will be a task for a future licensee as part of normal business.

4.7.4 Severe accident control & instrumentation

4.7.4.1 Background

313. In this section I consider the adequacy of the C&I to support the SAMGs. The UK ABWR design incorporates C&I for the monitoring of the plant during severe accidents and for the operation of facilities and systems required to deliver severe accident response. This C&I capability, described in PCSR Chapter 14 (Ref. 29), is delivered by a dedicated Severe Accident C&I system, supplemented by shared parts of the Class 2 HWBS. These provide the following functions:

- monitoring of plant conditions and mobile equipment in a severe accident to inform accident management; and
- monitoring and control of systems which provide a role during a severe accident, in particular the RDCF, FLSS and FCVS.

314. Hitachi-GE claims that these C&I systems are designed to be independent and diverse from Class 1 systems and, for the purposes of delivering severe accident functions, be able to withstand accident conditions. Reflecting Fukushima Dai-ichi learning, the systems are designed to be operable from two separate stations, located in the MCR and the B/B.

4.7.4.2 Severe accident parameters

315. During Step 4, Hitachi-GE has provided a strategy and approach for defining the list of parameters for inclusion in the Severe Accident C&I system (Ref. 98). Hitachi-GE has identified parameters based on the Accident Management Guidelines (Ref. 46) delivered as part of GDA. These guidelines are based on current Japanese practice, and I would expect these to be developed further for the UK ABWR post-GDA. However, I am satisfied that, for the purposes of GDA, these guidelines provide an adequate basis for identifying the key parameters relevant to the management of severe accidents for the UK ABWR.

316. Hitachi-GE has identified the key parameters relating to conditions in the RPV, PCV and SFP. These reflect the prompts for entry into, and progress through, the accident management guidelines. Parameters for monitoring of severe accident systems, both fixed and mobile, have also been identified. Hitachi-GE has chosen to designate parameters according to their importance in supporting the guidelines, recognising that some parameters provide a ‘primary’ basis for decision-making whereas other provide a secondary (or ‘backup’) role. This designation has been used as an input to Hitachi-GE’s categorisation and classification of Severe Accident C&I functions. I find the approach described for the selection and designation of parameters to be satisfactory for the purposes of GDA.

317. There may be a need for refinement of the list of parameters during detailed design, for example to reflect any accident management guidelines developed specifically for the UK ABWR.

4.7.4.3 Categorisation/classification

318. Hitachi-GE’s approach to categorisation and classification of severe accident measures is described in Chapter 26 of the PCSR (Ref. 24). Hitachi-GE states that SSCs for severe accident management should be categorised as Category B because severe accident management measures make a significant contribution to nuclear safety. Control and instrumentation functions shared with the HWBS (for example FLSS, FCVS and RDCF) are Class 2 as these systems deliver the principal means of delivering the Category B functions. Monitoring functions dedicated to the Severe Accident C&I are either Class 2 or Class 3, depending on whether the associated parameters have been designated as ‘primary’ or ‘backup’. C&I for monitoring of mobile equipment (for example fuel cooling using the FLSR) is Class 3, reflecting that this equipment is not the primary means of delivering the Category B function. Hitachi-GE’s approach meets my expectations.

4.7.4.4 Qualification

319. Hitachi-GE’s initial submissions indicated that the qualification of the Severe Accident C&I would be considered and justified after the specification of sensors had been determined and that this activity would therefore be carried out during site licensing. The submissions did not clearly explain how and where Severe Accident C&I (in

particular sensors and instruments) would be located within the plant or the likely operating conditions these might be exposed to in a severe accident. Within GDA I would not expect to see evidence that specific Severe Accident C&I equipment has been selected and qualified. However, I would expect assurance that, for a given monitoring role and environment, suitable technologies exist and that in principle equipment is capable of being qualified to the appropriate level at the licensing stage.

320. In response to RQ-ABWR-1336 (Ref. 103), Hitachi-GE has presented further information on the location of SA instrument sensors, likely environmental conditions in a severe accident, and proposed technologies for specific instrument sensors. My main focus has therefore been on equipment proposed to be physically located inside or near to the PCV as this is likely to have the most onerous severe accident qualification requirements. The RQ response explained which instrument components need to be located within the PCV and presents adequate arguments for GDA that the relevant technologies are capable of being qualified for the expected accident conditions.

321. The detection of hydrogen in the PCV is one of the prompts for confirmation that core damage has occurred in the SAMGs. I note that for this purpose Hitachi-GE is proposing to use a new technology based on a gas-sensitive composite material which changes electrical resistance depending on the hydrogen or oxygen concentrations. The qualification requirements for this technology, being introduced into Japanese plants post-Fukushima, is likely to be a specific area of interest during site licensing.

322. Based on the additional information provided by Hitachi-GE, I am satisfied that there is sufficient evidence for GDA that the proposed SA C&I is capable of being appropriately qualified during site licensing. My expectation is that qualification procedures would be developed during site licensing as part of normal business.

4.8 Lessons learnt from the Fukushima Dai-ichi accident

4.8.1 Background

323. The powerful earthquake and subsequent tsunami that affected the Fukushima Dai-ichi Nuclear Power Station on March 11, 2011 resulted in severe accidents involving three of the reactors at the site. Whilst the affected reactors at Fukushima Dai-ichi were of earlier BWR-3 and BWR-4 designs with Mark I containments, there are lessons from the accident which apply to all BWR types, including the ABWR, and also to nuclear power plants more generally.

324. Hitachi-GE sets out its response to the accident at the Fukushima Dai-ichi site in Chapter 28 of the PCSR (Ref. 34), with further consideration in Chapters 22 (Ref. 22) and 26 (Ref. 24). This is supplemented by Hitachi-GE's systematic review of the UK ABWR design against UK and international recommendations / learning (Ref. 45).

325. In PCSR Chapter 28, Hitachi-GE explains how the design of the UK-ABWR has evolved from the standard ABWR plant (based on Kashiwazaki-Kariwa units 6 and 7) to incorporate learning from the Fukushima Dai-ichi accident. To address Fukushima Dai-ichi learning, Hitachi-GE claims that the UK ABWR incorporates a number of enhancements compared to the standard Japanese plant. These are summarised below:

- For external hazards the plant is designed so that there are no 'cliff-edge' effects just beyond the design basis.
- Backup DC power supplies have been enhanced. C&I for Class 1 systems is supported by 8 hour DC supplies and the battery backup for the steam-driven RCIC has been extended to 24 hours.

- The FLSS has been introduced as an alternative means for providing fuel cooling; this is a fixed system, independent and diverse from the Class 1 ECCS. This is supported by an independent means for reactor depressurisation using the RDCF. The FLSS is able to deliver all low pressure injection and flooding demands and is self-sufficient in fuel and water for 7 days.
- There is provision for the use of mobile equipment with dedicated connection points on the outside of the R/B. This includes the FLSR which replicates the low pressure injection and flooding functions of the FLSS. The AHEF is available to support re-instatement of containment heat removal in the event of LUHS. These systems are supported by mobile power trucks which can also supply R/B C&I and HVAC loads.
- Inclusion of the B/B which is remote from the R/B and is designed to withstand hazards. This houses severe accident systems including the FLSS and is powered by redundant air-cooled diesel generators, diverse from the Class 1 EDGs.
- Key severe accident systems such as the FLSS, RDCF and FCVS can be operated remotely from either the MCR or the B/B.
- As a further means for depressurising the reactor, there is provision for operation of SRVs by local manual operation using nitrogen cylinders.
- A dedicated severe accident C&I system, independent of the Class 1 system and qualified for severe accident conditions, is provided in the B/B. This can be used for the remote monitoring and control of the plant in a severe accident.
- Improvements have been made to PCV seals to enhance resilience of the primary containment to severe accident loads. The PCV head flange seal can also be protected against high temperatures by emergency flooding of the reactor well.
- The design includes enhanced measures for management of hydrogen in the primary and secondary containments.

4.8.2 Overall assessment

326. During GDA Step 3, I raised Regulatory Observation RO-ABWR-0039 (Ref. 49) seeking clarification from Hitachi-GE on how UK learning from the Fukushima accident had been addressed in the design of the UK ABWR. In particular, I requested that Hitachi-GE consider recommendations from the HM Chief Inspector of Nuclear Installations reports (Ref. 13 & 14) and the European Council stress tests findings (Ref. 15).

327. During Step 4 I extended RO-ABWR-0039 to include learning from the Fukushima Dai-ichi Accident Report by the IAEA Director General (Ref. 16). Hitachi-GE's RO response and the supporting references have formed an important part of my wider assessment of the severe accidents safety case.

328. In response to RO-ABWR-0039, Hitachi-GE has provided a comprehensive submission (Ref. 45) setting out how the individual UK and international recommendations and learning points have been addressed in the design of the UK ABWR. Hitachi-GE has also considered the improvements proposed for UK plants at Hinkley Point C, Sizewell B, Torness and Wylfa and determined, as appropriate, how these have been addressed in the design of the UK ABWR. Finally, Hitachi-GE has identified relevant findings from the 2015 IAEA Operational Safety Review Team (OSART) mission to the ABWRs at Kashiwazaki-Kariwa Units 6 & 7 in 2015 (Ref. 102).

329. I have considered Hitachi-GE's detailed responses to the RO as part of my Step 4 assessment activities, where necessary with input from ONR colleagues in other disciplines. Overall, I am satisfied that Hitachi-GE's response (Ref. 45), read in conjunction with supporting submissions, provides an adequate demonstration that UK and international learning has been addressed.

330. I would expect the licensee to consider any implications for the design of the plant, and the severe accident measures/response, in the light of further understanding of the accident progression and conditions of the Fukushima Dai-ichi units.

331. Hitachi-GE has identified (Ref. 45) that a number of the specific recommendations and learning points do not relate to the generic design and cannot be addressed by the Requesting Party as part of GDA. Hitachi-GE considers that such matters will be for the future licensee to address. To ensure that these matters are taken forward by the future licensee, I raise the following Assessment Finding:

- AF-ABWR-SA-08: Hitachi-GE has identified several lessons and learning points from the Fukushima Dai-ichi accident that are site-specific or matters for the licensee to consider, which cannot be fully addressed in GDA. The licensee shall review relevant lessons and learning points identified as being out of GDA scope in Hitachi-GE document AE-GD-0505 Rev.2 and demonstrate that these have been addressed in the design and proposed operation of the site-specific plant.

4.8.3 External hazards

332. A significant number of the key recommendations and learning points from the Fukushima Dai-ichi accident relate to external hazards withstand. These aspects were considered as part of ONR's Step 4 external hazards assessment and resulted in RO-ABWR-0067 (Ref. 95). This RO covers beyond design basis external hazards generally, but has a specific focus on external flooding. ONR's review of Hitachi-GE's summary response (Ref. 100) and supporting documents is included in the external hazards assessment report (Ref. 96).

4.8.4 Fuel Cooling

333. IAEA recommendations following the accident at Fukushima Dai-ichi (Ref. 16) highlight the need for robust and reliable cooling systems that can function for both design basis and beyond design basis conditions. I consider that the Class 1 design basis provisions, supplemented by independent FLSS (Class 2) and FLSR (Class 3) systems address this expectation.

334. Both the FLSS and FLSR have independent water sources and fuel supplies sufficient for at least seven days operation without external support. I am satisfied that this addresses the relevant aspects of IR-19 and IR-20 from the Chief Nuclear Inspector's Interim Report, FR-3 from Chief Nuclear Inspector's Final Report and STF-9 from the Stress Test Findings report.

335. I have also considered the expectations in STF-8 of the Stress Test Findings on the robustness of equipment connection points. Hitachi-GE has stated that the FLSR can be connected to the R/B at multiple, separate locations (Ref. 38). In response to RQ-ABWR-0666 (Ref. 97), Hitachi-GE has confirmed that the connection points will be designed to be robust against site-specific external hazards. I would expect this to be considered further for the site-specific design, however I am satisfied that Hitachi-GE's proposals are sufficient for GDA.

4.8.5 Containment

336. Stress Test Finding STF-6 is specifically related to containment of gas-cooled reactors, but I have considered the intent of this finding in the context of the UK ABWR. I have already considered Hitachi-GE's containment concept in Section 4.5. I am satisfied by Hitachi-GE's arguments and evidence (Ref. 44) that margins have been reviewed and that the point of failure has been established. I am also satisfied that the IAEA's post-

Fukushima expectation for a reliable confinement function for beyond design basis accidents has been met.

4.8.6 Hydrogen management

337. The need to consider of methods to control combustible gases was captured in the Chief Nuclear inspector's Interim Report. IR-21 recommends that: "The UK nuclear industry should review the ventilation and venting routes for nuclear facilities where significant concentrations of combustible gases may be flowing or accumulating to determine whether more should be done to protect them". I am satisfied that aspects relating to the accumulation of combustible gases in the PCV and R/B have been covered by the Hitachi-GE's work to address RO-ABWR-0023 Action 4. In particular, Hitachi-GE has provided measures to prevent hydrogen combustion to prevent accumulation of hydrogen in the R/B.

338. The accidents at Fukushima Dai-ichi also identified the potential for hydrogen to migrate through vent system pipe networks. I have investigated whether similar risks could arise for the UK ABWR. The preferred venting route for faults and severe accidents is via the FCVS. As this system is separate from other HVAC systems and has its own exhaust line up into the main stack (Ref. 39), I am satisfied it protects against hydrogen migration. In response to RQ-ABWR-1319 (Ref. 104), Hitachi-GE has acknowledged that the hardened vent system could be operated in severe accidents if the FCVS is unavailable. In that case, the vent path would be via the SGTS exhaust line (Ref. 104), which has its own pipework to the main stack. Backflow via the SGTS into the R/B is prevented by the SGTS isolation valves (which are normally closed if the SGTS is not in operation). I am satisfied that the risk of hydrogen migration from controlled venting has been adequately considered for GDA.

4.8.7 Severe accident control & instrumentation

339. Recommendation IR-22 of the Chief Nuclear Inspector's Interim report relates to the provision of on-site of emergency control, instrumentation and communications. I note that the B/B provides a diverse, remote means for the control and monitoring of the plant during a severe accident, using Severe Accident C&I powered by the BBGs. This provides a demonstration to me that Hitachi-GE has addressed the learning in this area. Complimentary to this, I am also satisfied that the provision of a dedicated severe accident C&I system, accessible from the MCR and B/B, meets the severe accident expectations of SAPs ESS.3 and ESR.1.

340. Recommendation IR-23 requires consideration of the robustness of the off-site communications for severe accidents. In response to IR-23, Hitachi-GE has set out a strategy for the future development of the Emergency Response Facility C&I system (Ref. 105). The intention is that this system will provide a means of collating and distributing real-time plant status information to various response facilities both on- and off-site in the event of an accident. Hitachi-GE's specification for on-site-communications has been reviewed as part of the Electrical Engineering assessment (Ref. 101). Overall, I am satisfied that, for the purposes of GDA, Hitachi-GE has provided an adequate response to IR-22 and IR-23 of the Chief Nuclear Inspector's Interim report.

341. Power for the SA C&I and HWBS is provided by the Class 2 BBGs. These are designed to operate for up to 7 days without external support. Hitachi-GE has demonstrated that there is robust separation from the Class 1 systems, for example, there is no electrical connection and the BBGs are air-cooled designs which are separated from the EDGs. The BBGs are supported by back-up batteries which provide power for a nominal period of one hour, sufficient to supply the Class 2 HWBS, Severe Accident C&I and other loads until the BBGs have started.

342. I note that IAEA learning from the accident Fukushima Dai-ichi (Ref. 16) identifies the need for instrumentation and control systems that are necessary during beyond design basis accidents to remain operable. Failure of the BBGs, and therefore failure of the FLSS to provide fuel cooling, may be a precursor for a severe accident. The B/B does not include long-term provision for backup power in the event that the BBGs are unavailable or have failed. Such an event would leave the Severe Accident C&I without power. In response to RQ-ABWR-1307 (Ref. 99), Hitachi-GE claims that failure of the BBGs is unlikely and that Class 1 C&I, supported by mobile power supplies, might be available to support accident management. Whilst I accept that these are relevant arguments, Hitachi-GE has not considered whether it would be ALARP to provide a means for connection of mobile power sources to the B/B to support the Severe Accident C&I functions. I accept that the practicality of providing backup power connections would need to be considered as part of the detailed design of the B/B at the site-specific stage. I therefore raise the following Assessment Finding:

- AF-ABWR-SA-09: For the reactor building, Hitachi-GE has included the provision to connect mobile power units to support Class 1 systems. However, the Severe Accident Control & Instrumentation system is powered by the backup building electrical power system. A failure of backup building power sources is a potential way for a fault condition to escalate to a severe accident scenario, resulting in the loss of severe accident control and instrumentation functions. As part of its work to develop a final design for the backup building, the licensee shall consider whether it is ALARP to provide a capability for mobile power supply sources to be connected to the Severe Accident Control & Instrumentation system, to ensure that control and monitoring of severe accident systems can be maintained in circumstances where the fixed backup building power sources have failed.

4.9 Practical elimination of large or early releases

4.9.1 Background

343. As discussed in Section 2 of this report, there is an international expectation that large or early releases be practically eliminated for new reactors. It is also a requirement of Hitachi-GE's own internal Nuclear Safety and Environmental Design Principles (Ref. 110), specifically Principle 8.11.1, that "significant radioactive releases are practically eliminated". Hitachi-GE has interpreted 'significant radioactive releases' in Principle 8.11.1 as being equivalent to large or early release.

344. Hitachi-GE has prepared a submission (Ref. 37) setting out how the UK ABWR meets these expectations. Hitachi-GE claims that large or early releases have been practically eliminated for the UK ABWR by:

- identifying the provisions which are designed to prevent or mitigate an accident;
- identifying conditions which could lead to large or early releases; and
- demonstrating that large or early releases are of 'extremely low likelihood' with a high degree of confidence.

345. Regarding design provisions, Hitachi-GE claims that:

- the design offers a high degree of defence in depth and flexibility; and that.
- multiple failures of design provisions are necessary for severe accidents to occur.

346. Hitachi-GE has used PSA to identify conditions which could give rise to large or early releases and used numerical risk values to support conclusions on the likelihood of releases.

4.9.2 Approach to practical elimination

347. Relevant guidance on practical elimination is provided in:

- ONR's Fault Analysis SAPs (para. 611, Ref. 2), which require that severe accident end states should be practically eliminated;
- IAEA SSR-2/1 Rev 1 (Ref. 9) and supporting information in IAEA-TECDOC-1791 (Ref. 112); and
- WENRA guidance on safety objectives for new reactors, specifically Objective O3 "Accidents with core melt" and Positions 4 & 5 of the report on Safety of New NPP Designs (Ref. 10).

348. Neither the SAPs, nor the international guidance prescribe specific approaches that should be used as part of a demonstration of practical elimination. However, WENRA guidance states that accident sequences with a large or early release can be considered to have been practically eliminated if:

- it is physically impossible for the accident sequence to occur; or
- the accident sequence can be considered with a high degree of confidence to be extremely unlikely to arise.

349. Hitachi-GE has not claimed that any specific sequence is 'physically impossible' and instead has identified all relevant severe accident phenomena and analysed these as part of the Level 2 PSAs. This is consistent with my expectations on identification of conditions which could lead to a large or early release.

350. WENRA (Ref. 10) defines early releases as situations that would require off-site emergency measures but with insufficient time to implement them. There is no WENRA definition of the time that should be assumed for implementation of emergency measures. In Ref. 37, Hitachi-GE refers to its PSA definition of an early release; an early release is one where containment failure occurs within four hours of RPV breach, or occurs before RPV breach, but within 10 hours of the initiating event. For the at-power Level 3 PSA release categories considered by Hitachi-GE, this effectively means that early releases are those which occur within 12 hours of the initiating event.

351. A key aspect of Hitachi-GE's case is that the design offers a high degree of defence in depth and is equipped with diverse and redundant systems which make multiple sequential failures extremely unlikely. Furthermore, in the case of severe accidents involving the SFP or the reactor at shutdown with the RPV head removed, Hitachi-GE claims that there are significant timescales between loss of cooling (and to a lesser extent start of a LOCA) and uncovery of fuel. In my opinion, consideration of these factors is relevant and forms an important part of the overall demonstration that large or early releases have been practically eliminated.

352. The available guidance is clear that an accident state should not be considered to have been practically eliminated simply on the basis of meeting probabilistic criteria. Hitachi-GE does consider results from the PSA, supported by sensitivity analyses, to inform its conclusions on practical elimination. In my opinion it is appropriate for Hitachi-GE to use results from the UK ABWR PSAs to support judgements as one part of its wider case. I comment in my assessment on how the PSA results have been used to support the position on practical elimination, but review of the PSA methodology and results has been performed as part of the PSA assessment (Ref. 4).

353. In the context of practical elimination of large or early releases, there is no common position in the international guidance on use of numerical targets to define what is 'extremely unlikely'. ONR does not set explicit targets for measures such as large release frequency. However in Ref. 107, ONR does equate such measures with

Target 9 in the SAPs. ONR considers that the BSOs are relevant measures for new reactor designs proposed for the UK. These are used by ONR as benchmarks that reflect modern standards and expectations, thus ONR refers to these objectives when judging whether analyses are demonstrating adequate results for new reactors (Ref. 12). However, the BSOs are 'objectives' and not requirements – the overriding legal requirement for new reactor designs is that the level of risk is demonstrated to be ALARP when the facility starts operation and over its lifetime. Hitachi-GE's comparison of risks against the Target 9 BSO is therefore a relevant consideration for my assessment, but needs to be considered alongside my assessment of Hitachi-GE's severe accidents ALARP case in Section 4.10.

4.9.3 Releases from the primary containment

354. For accidents with core melt, the expectation in WENRA Position 4 (Ref. 10) is that the reactor containment structure is the main barrier for protecting the environment from the radioactive materials. In the case of the UK ABWR, this structure is the PCV. Hitachi-GE claims that it has provided effective design provisions to prevent PCV failure for accidents with core melt. This is through a combination of active and passive measures, in particular:

- The Class 1 ECCS (including HPCF, RCIC and LPFL) provides a core cooling function in fault conditions to prevent fuel damage and therefore prevent progression to a severe accident.
- If a fault develops into a severe accident, then fixed (FLSS) and mobile (FLSR) low pressure injection systems are available to provide core cooling, cooling of corium in the LDW and water for containment sprays. Corium in the LDW can also be cooled by water from the passive LDF.
- Containment heat removal can be delivered by venting of excess steam pressure using manual venting or the passive COPS. The mobile AHEF system is available to support recovery of containment heat removal by the RHR.
- The PCV is inerted to minimise the potential for hydrogen combustion in the primary containment.

355. Appendix 4 of IAEA TECDOC-1791 (Ref. 112) also identifies specific accident conditions which could challenge the containment.

- Uncontrolled reactivity accidents - Hitachi-GE has identified that there is a theoretical possibility for re-criticality during RPV re-flooding, but considers that this would be unlikely. To mitigate this risk, Hitachi-GE has proposed that boron could be injected into the RPV during re-flooding using the SLC system.
- Direct containment heating – The RDCF can be used to depressurise the RPV in a severe accident to prevent high pressure melt ejection. The RDCF is diverse from the Class 1 ADS, with its own power and nitrogen supplies.
- Large steam explosion – Hitachi-GE claims that the containment would be expected to withstand the effects of an ex-vessel FCI steam explosion.
- Hydrogen detonation – Hitachi-GE claims that its concept for hydrogen management that would be effective in limiting concentrations to well below detonation levels.
- Molten core concrete interaction – The UK ABWR includes provisions for cooling of ex-vessel corium using diverse means for delivering cooling water. Hitachi-GE concludes that corium can be cooled by overlying water, sufficient to prevent containment failure due to MCCI.

356. Based on my assessments in Sections 4.5 and 4.6 of this report, I am satisfied that Hitachi-GE has made an adequate demonstration, for the purposes of GDA, that the proposed severe accident provisions would be effective in preventing releases due to containment failure.

357. Through the use of PSA, Hitachi-GE has identified accident sequences which could result in containment failure, for example due to failure of design provisions. For these sequences, Hitachi-GE has supported its arguments with reference to the Level 3 PSA for the reactor at power. The results show that the Target 9 risk for reactor accidents, summed for all large and large early release categories, is approximately 10^{-6}/year. This value is above ONR's BSO, but below the BSL. The risks are dominated by accidents initiated by internal hazard events. Hitachi-GE has presented arguments that further improvements to the PSA model are proposed post-GDA and that these would be expected to reduce the calculated risk for hazards. I am satisfied that, for the purposes of GDA, the PSA results support Hitachi-GE's report on practical elimination for reactor accidents.

4.9.4 Containment bypass as a resulting of venting

358. A consequence of Hitachi-GE's strategy for the UK ABWR is that radioactivity could be released during venting of the containment. Venting would be required under certain DB and BDB faults and in accident conditions to ensure that excess steam pressure can be released from the containment, thus preventing containment failure. The UK ABWR design includes filters to minimise radiological consequences of venting.

359. For a medium-term SBO (a DB fault) Hitachi-GE has shown that the off-site dose from venting is below the BSO for ONR's Numerical Target 4, even when assuming the release to be unfiltered. This is because Hitachi-GE's analysis predicts no consequential fuel damage for this DB fault. Hitachi-GE's analysis (Ref. 40) shows that venting would be required after about 13 hours. Long-term SBOs (treated as BDB faults) result in similar off-site dose levels for the same reason. On this basis I am satisfied that the radiological consequences of venting during DB and BDB faults are low (less than 0.01 mSv off-site) and venting does not result in a large or early release.

360. In the event of loss of containment heat removal in a severe accident, the strategy for the UK ABWR is to vent the primary containment using the FCVS, initiated either by operators or passively by the COPS. For Hitachi-GE's analysis shows that venting would not be required until at least 13 hours after the start of an accident (Ref. 35). Hitachi-GE claims that the radiological consequences of venting using the FCVS meet ONR's BSO for Numerical Target 9 (Ref. 53). Hitachi-GE's Target 9 calculation for venting has been reviewed as part of ONR's PSA assessment (Refs. 4 and 113) and found to be acceptable. On this basis, I am satisfied that severe accident venting does not challenge Hitachi-GE's claim that large or early releases have been practically eliminated.

4.9.5 Releases from the spent fuel pool and during shutdown

361. Hitachi-GE has considered SFP events arising from loss of fuel pool cooling and small LOCAs (Ref. 37). Hitachi-GE identifies that the large volume of water in the SFP is a significant factor in preventing a severe accident once an initiating event has occurred. For loss of cooling faults resulting in boiling of the pool, Hitachi-GE has calculated that the time to fuel damage would be about 330 hours, even without any water make-up. For a small LOCA (with an assumed leak rate of 30 m^3/hour) the time assuming no make-up would be lower, but still approximately two days. Hitachi-GE claims that even once the pool water level has dropped to the TAF, fuel damage could be prevented by providing modest amounts of make-up water to offset losses due to pool boiling and/or leakage. Hitachi-GE claims that SFP cooling/make-up could be provided by the FPC, RHR, MUWC, SPCU, FLSS, FLSR and FP systems. Given the timescales, I agree with Hitachi-GE's conclusion that there are robust provisions for preventing fuel uncovery, and therefore a large release, from the SFP.

362. For the majority of time during shutdown, the RPV is open to the reactor well and DSP which are flooded with a large volume of water shared with the SFP. Even if boiling or

leakage occurs, Hitachi-GE argues that this volume provides a large time to allow mitigation of any fault before fuel in the reactor is uncovered and a large release occurs. The time to fuel damage is at least 155 hours for boil-off following loss of decay heat removal and 19 hours in the case of a small LOCA (Ref. 37). Hitachi-GE claims that there are a number diverse and redundant systems that are available to provide the required cooling/make-up to the shutdown reactor, including the RHR, HPCF, LPFL, MUWC, FLSS and the FLSR. Hitachi-GE claims that these systems would ensure that fuel would remain covered, therefore preventing a large release. I am satisfied that there are robust measures for preventing fuel uncovery and large releases from the shutdown reactor.

363. For both the SFP and shutdown, Hitachi-GE has supplemented its arguments on design provisions with PSA evidence on the frequency of boil-off and small LOCA sequences. I note that these events, treated individually, do not challenge Target 9 and do not make a significant contribution to the cumulative risk. I also note Hitachi-GE's argument that there are known conservatisms in the PSA, in particular regarding the potential for recovery of the FPC system and the FLSS (as recovery is not considered in the PSA), and the assumptions on availability and success of water addition from mobile sources. In my opinion, the PSA evidence supports Hitachi-GE's conclusion that large or early releases from the SFP would be extremely unlikely.

364. A catastrophic failure of the SFP would result in a loss of water in excess of the make-up capacity of the fixed and mobile systems, resulting in rapid uncovery of fuel and an early release. Hitachi-GE has determined as part of the PSA that the frequency of such an event would be less than 10^{-10}/year. I consider that the robust design on the SFP and the elimination of large penetrations means that the risk of such a failure has been reduced ALARP. In this case I accept the argument that this event would result in risks for this event well below the SAPs Target 9 BSO.

4.9.6 Conclusions on practical elimination

365. Based on my assessment, I consider that Hitachi-GE has presented a thorough report on practical elimination. In particular, I conclude that Hitachi-GE has:

- considered the conditions, including releases due to containment venting, which could lead to a large or early release;
- demonstrated that there are design provisions which are intended to prevent large or early release;
- demonstrated through the use of PSA that large or early releases are in the ALARP region for Target 9.

366. Hitachi-GE's PSA considers internal events and internal hazards (fire and flood). Target 9 risks are dominated by events initiated by internal hazards (Ref. 53). During Step 4 of GDA Hitachi-GE has made refinements to the internal hazards PSA to remove conservatism, but Hitachi-GE believes that there is potential for further refinements for the specific site. I also note that Hitachi-GE has not quantified risks for internal hazard initiators for shutdown and the SFP. I would expect that future refinements to the PSA for the specific site would be used to support Hitachi-GE's conclusions on practical elimination. Furthermore, Hitachi-GE has not considered the PSA contribution from external hazards when considering practical elimination. This is consistent with my expectations for GDA because the necessary details on external hazards are site-specific. My expectation is that the arguments on practical elimination would be updated for the specific site and so I make the following Assessment Finding:

- AF-ABWR-SA-10: To meet UK and international expectations post-Fukushima, Hitachi-GE has provided a demonstration which argues that the generic UK ABWR design practically eliminates large or early releases. The extent to which hazards, and therefore the completeness of any practical elimination claim, can be considered in GDA is limited. In particular, external hazards will present an additional contribution to the site-specific risk profile. The licensee shall review and update as appropriate the deterministic and probabilistic arguments that support the claim that large or early releases have been practically limited on a site-specific basis, notably to consider the risks associated with the site-specific beyond design basis hazard profile.

4.10 Demonstration that risks are ALARP

4.10.1 Background

367. Demonstration that risks are ALARP is a fundamental requirement of UK law that a future licensee would have to comply with. Hitachi-GE's approach to ALARP in general is described in PCSR Chapter 28 (Ref. 34) and specifically in relation to severe accident measures in PCSR Chapter 26 (Ref. 24).

368. Hitachi-GE argues that the reference ABWR design already includes extensive severe accident measures as a result of the development of the design from earlier BWRs. Furthermore, following the Fukushima Dai-ichi accident, Hitachi-GE argues that the UK ABWR benefits from additional measures including:

- additional safety features provided by the B/B;
- provision of mobile equipment for flexible accident response; and
- development of accident management procedures and guidelines.

369. Hitachi-GE has reviewed 'relevant good practice' for other operating reactor designs, new reactor designs proposed for the UK and the ESBWR (a conceptual design which provides a further evolution of the BWR). Hitachi-GE has identified and considered alternative measures for prevention and mitigation of severe accidents as employed by these other reactor designs. Hitachi-GE argues that the UK ABWR has other comparable systems, or has measures that provide similar functions to other designs, and therefore the design is ALARP.

370. In its response to RO-ABWR-0076 (Ref. 106), Hitachi-GE also claims that the PSA has been used to identify and consider further risk reduction options for the design as a whole.

371. Hitachi-GE has also presented a case that the risk of large or early accidental releases has been practically eliminated (Ref. 37); this compliments the ALARP demonstration.

4.10.2 Assessment overview

372. As part of RO-ABWR-0023 Action 4, I asked Hitachi-GE to explain why the design of the UK ABWR represents relevant good practice and follows the ALARP principle in relation to severe accidents. Consideration of whether relevant good practice has been followed, and whether the risks are ALARP, has been a relevant consideration throughout my assessment. However, RO-ABWR-0023 was seeking to establish the ALARP arguments underpinning the fundamental severe accident design concepts for the UK ABWR.

373. Based on a review of important severe accident phenomena identified by my TSC (Ref. 17), I asked Hitachi-GE to explain why the UK ABWR design was ALARP in six key areas:

- methods / technologies for confining a molten core;
- passive methods of core or containment cooling;
- methods for further increasing grace / response times;
- methods of further capturing / reducing fission products inside containment;

- the design of the containment head flange and other systems to protect from containment leakage; and
- passive methods for flammable gas control.

374. ONR provides specific guidance on ALARP for proposed new civil nuclear reactors in Annex 2 of NS-TAST-GD-005 (Ref. 8). This is supported by guidance on how ONR uses risk to inform regulatory decision making, including for GDA (Ref. 107). Reflecting this guidance, the focus of my assessment has been on whether:

- there is a clear conclusion that there are no further reasonably practicable improvements that could be implemented;
- relevant good practice has been incorporated into the design;
- there is a rationale for the evolution of the proposed design from its forerunners; and
- risk assessment has been used to identify potential engineering and/or operational improvements in addition to confirming the numerical levels of safety achieved.

375. Consideration of what constitutes relevant good practice in the area of severe accidents is not straightforward. There are high level expectations for severe accidents in the SAPs and in the international guidance from the IAEA and WENRA, but there are no prescriptive requirements or expectations for any specific design features. I have been able to gain insights on measures adopted in other ABWR designs through the MDEP activities (Section 4.12) and I have used this to inform my judgements on relevant good practice. For other new reactor designs, ONR may have previously accepted particular severe accident measures as ‘good practice’, however we recognise that such good practice identified for one design may not necessarily be relevant to another.

376. Hitachi-GE provides useful insights to inform my judgement of the risk significance of severe accident phenomena and events. In particular, Hitachi-GE has presented a Level 3 PSA (Ref. 53) which allows direct comparison of risks with the Numerical Targets in ONR’s SAPs (Ref. 2). However, in my interactions with Hitachi-GE, I have recognised that the explicit consideration of cost-benefit analysis may have limited value when considering very low frequency, but high consequence severe accident events.

377. My assessment of Hitachi-GE’s demonstration of practical elimination is reported in Section 4.9. This compliments my assessment of Hitachi-GE’s ALARP arguments.

378. In the following sections, I consider Hitachi-GE’s arguments for why they believe the UK ABWR design is ALARP. In a number of the arguments, Hitachi-GE refers to relevant good practice from the ESBWR, a next generation BWR design offered by GE-Hitachi. I understand that there are no current projects to build this reactor design and therefore ONR would not necessarily consider features offered by the ESBWR as relevant good practice.

4.10.3 Confining a molten core

379. The strategy for the UK ABWR is that molten corium would be retained in-vessel by flooding the RPV before a breach occurred. If flooding was unsuccessful and the RPV failed, then corium would be contained within the LDW spreading area and cooled by over-lying water. Steam would be released from the containment via the filtered route (using the passive COPS if necessary) to prevent failure of the PCV. I am content that this strategy is credible and supported by adequate analysis showing that containment failure in such cases due to MCCI would be unlikely (see Section 4.6).

380. Hitachi-GE has reviewed alternative options for achieving the same objectives of managing ex-vessel corium (Ref. 41). Hitachi-GE argues that adoption of alternative technologies such as the Core Melt Stabilisation System (CMSS) employed on the EPR, or the passive Basemat-internal Melt Arrest Coolability (BiMAC) device proposed for the ESBWR, would have major implications for the design and would not be reasonably practical to install on the UK ABWR. I accept Hitachi-GE's argument that incorporation of these features would not be reasonably practical in the ABWR plant. Hitachi-GE does not explicitly discuss alternative strategies such as in-vessel retention (IVR) by exterior cooling of the RPV. Instead, Hitachi-GE has chosen to strengthen the design against MCCI using measures to contain and cool corium ex-vessel, thus achieving the objectives or preventing containment failure. In addition to the passive LDF, I note that the UK ABWR includes additional diverse provisions for flooding of the LDW using the FLSS and FLSR. Where considered reasonably practical, Hitachi-GE has targeted additional protection for vulnerable areas of the LDW, for example by using refractory material to protect the LDW sump. Overall, I am satisfied that Hitachi-GE has presented a credible strategy for GDA.

381. Hitachi-GE has also considered the possibility of containment failure due to MCCI in the PSA. Level 3 PSA results (Ref. 53) show that the contribution to the overall Target 9 risk is small and for the relevant sequences in isolation is well below the BSO. Whilst there is a degree of uncertainty attached to the analysis, I am satisfied that this provides support to Hitachi-GE's argument.

382. In conclusion, I accept Hitachi-GE's ALARP argument that, for the UK ABWR, the design implications for incorporating alternative methods for confining a molten core would be disproportionate to any safety benefits. I am satisfied that Hitachi-GE's concept for ex-vessel corium cooling in conjunction with filtered venting achieves the objective of preventing gross containment failure, and thus meets my expectations.

4.10.4 Core and containment cooling

383. Hitachi-GE has considered options for the provision of alternative means of providing core and containment cooling (Ref. 108).

384. The strategy for the UK ABWR is that core cooling in the RPV in a severe accident is delivered by diverse active systems. During the early stages of an accident, cooling is likely to be delivered by the RCIC, part of the Class 1 ECCS. This system operates automatically using decay-heat steam, without any requirement for external power. As demonstrated at Fukushima Dai-ichi, the RCIC is likely to be reliable and effective, however ultimately this will be constrained by heat-up of the S/P and depletion of C&I battery supplies. Longer-term core cooling would need to be delivered by active systems, which could be fixed or mobile. If in-vessel cooling is ineffective then Hitachi-GE claims that cooling of ex-vessel corium would be by the passive LDF, although ultimately this would require active make-up to the containment.

385. Until active systems such as RHR and AHEF are available, containment heat removal would be by venting excess steam pressure to atmosphere through filters. Ultimately this can be achieved passively before containment failure occurs by rupture of the COPS bursting disks.

386. Hitachi-GE has reviewed in detail the passive core and containment cooling systems adopted by the AP1000® and ESBWR designs (Ref. 108). Hitachi-GE argues that the active and passive features of the UK ABWR deliver the same safety functions, although unlike the other designs, this requires filtered venting of steam from the containment to provide long term heat removal until active cooling can be restored. Hitachi-GE has not explicitly considered whether it would be reasonably practical to incorporate the passive features of the AP1000® and ESBWR into the UK ABWR, however my understanding is that this would likely entail substantial design changes.

Hitachi-GE has demonstrated in its Level 3 PSA (Ref. 53) that severe accident venting through a filtered route meets the relevant Numerical Targets (see Section 4.9) and therefore I conclude that it would not be proportionate to expect such changes.

387. In PCSR Chapter 28 Hitachi-GE provides some explanation of how the arrangements for cooling have evolved. For example, for the ABWR the RCIC has been integrated into the Class 1 ECCS, providing greater reliability. The system is improved further in the UK design by extending the battery duration to up to 24 hours, reflecting learning from the Fukushima Dai-ichi accident. I note that some of the earliest designs of BWRs incorporated an isolation condenser, instead of the RCIC, to provide a passive means of removing heat from the core. An isolation condenser works by condensing decay-heat steam from the reactor via a closed loop to a water pool heat sink outside of the containment. A similar feature is proposed for the ESBWR. Hitachi-GE does not explain why this passive feature was discontinued in later BWRs, although the need for operator action to replenish the heat sink and the requirement for an additional large water pool near the top of the R/B are likely to be relevant factors. I am satisfied that, in the short-term at least, the RCIC reliably achieves the same objective.

388. In summary, I accept the principles of Hitachi-GE's argument that, for core and containment cooling, the UK ABWR delivers the same safety functions as proposed for fully passive reactor designs. A key factor in my judgement is that the UK ABWR provides effective filtering of vented releases from the containment through the FCVS.

4.10.5 Grace times

389. Hitachi-GE has considered what more could be done to extend grace times to key phases of a severe accident, for example, the times to core melt, RPV failure and containment failure (Ref. 109). A key part of the argument is that the UK ABWR incorporates diverse active systems (ECCS, FLSS and FLSR) for providing core cooling and that these are available for prolonged operation. Similarly, containment heat removal can be achieved by the RHR (supported by the mobile AHEF) and through filtered venting. Based on my assessment, I am satisfied that Hitachi-GE has demonstrated that these measures would be effective in preventing core melt, RPV failure and ultimately containment failure (depending on when the systems were actuated).

390. Hitachi-GE has considered alternative methods of achieving core and containment cooling functions, including the passive systems of the AP1000® and ESBWR. Many of the arguments adopted are similar to those discussed in Section 4.10.4 above, so are not repeated them here.

391. I recognise Hitachi-GE's arguments in PCSR Chapter 28 that the UK ABWR includes a number of enhancements to the original ABWR concept (for example the B/B and mobile equipment) and as such would have a beneficial effect on grace times. Following the accident at Fukushima Dai-ichi, the DC battery supply to the RCIC (part of the ECCS) has been extended to provide power for up to 24 hours. Fuel and water supplies for the FLSS and FLSR allow operation for up to 7 days without external support. From my participation in MDEP activities I consider that these enhancements reflect relevant good practice from other ABWR-type designs.

4.10.6 Capturing / reducing fission products

392. Hitachi-GE identifies (Ref. 74) that the UK ABWR includes a number of measures for the capture/reduction of fission products, including:

- containment sprays;
- LDW flooding (for scrubbing of gases from ex-vessel corium);
- S/P (with pH control) for scrubbing of WW gases prior to venting;

- FCVS; and
- SGTS for the R/B.

393. Hitachi-GE's has compared these measures with those found on other modern designs and concludes that the UK ABWR provides an equivalent capability, with scrubbing by the S/P being a notable feature which provides significant fission product retention. I am satisfied that Hitachi-GE has made credible arguments to support the design of the UK ABWR. The evidence to support the effectiveness of these measures has been considered in the Reactor Chemistry assessment (Ref. 66).

394. A key evolution from earlier designs is that venting through the FCVS will be filtered – I consider that this now reflects relevant good practice for BWRs generally, based on recent trends in Western Europe and Japan. For the UK ABWR, the provision of vent filters means that reactor severe accident releases as a result of venting do not contribute to Target 9 in the Level 3 PSA (Ref. 53).

395. Hitachi-GE has identified that provision for control of pH in the S/P during severe accidents is also relevant good practice. The arrangements for this have been subject to detailed assessment by the Reactor Chemistry assessor (Ref. 66). The potential risk reduction has been analysed in the Level 3 PSA by making bounding assumptions on the effectiveness of pH control on release source terms. Further analysis is required for the site-specific safety case (Ref. 66).

396. Overall, I am satisfied that Hitachi-GE has made a strong case that the measures for capturing and reducing fission products are ALARP.

4.10.7 Hydrogen management

397. Hitachi-GE has carried out an ALARP review of options for the management of hydrogen in a severe accident (Ref. 43). This has resulted in a change to the FCS, which now uses PARs in the PCV instead of active recombiners. I am satisfied that Hitachi-GE has considered the relevant options and that the use of passive methods involving PARs, in combination with a nitrogen inerted PCV, reflects relevant good practice.

398. In Ref. 43, Hitachi-GE also considers options for hydrogen management in the R/B and has proposed a strategy consisting of PARs and the SGTS for potential primary containment leakage. Opening of a blowout panel and large equipment door in the R/B is also proposed for managing hydrogen during severe accidents involving the shutdown reactor (with PCV head removed) and the SFP. I have considered the effectiveness of these measures in Section 4.6.

399. I accept that the measures for the R/B reflect relevant good practice, based on changes made to Japanese ABWRs post-Fukushima. As part of the ALARP justification to support the use of the blowout panel (Ref. 43), Hitachi-GE balances the benefits of preventing a hydrogen deflagration in the R/B against the risk of radioactivity release through the blowout panel. Hitachi-GE's arguments form part of a wider consideration of the arrangements for managing steam in design basis steam generation events, and for managing hydrogen in a severe accident. As discussed in Section 4.4, the use of the blowout panel has been considered in conjunction with fault studies colleagues. In the context of severe accidents, I accept Hitachi-GE's arguments that use of a blowout panel to manage hydrogen in the R/B is the ALARP solution. A key factor in my judgement is Hitachi-GE's claim that the uncovery of fuel in a shutdown reactor, or the SFP, has been practically eliminated. I consider Hitachi-GE's arguments which support this claim in Section 4.9.

4.10.8 PCV head

400. Hitachi-GE has presented an ALARP discussion on the design of the PCV head flange and other systems to protect from containment leakage (Ref. 75). This considers options for the selection of seal materials and protection of the PCV head flange seal against over-temperature by flooding of the reactor well. These aspects are covered in detail in my assessment of containment performance in Section 4.5. In summary, I consider that Hitachi-GE has identified worthwhile post-Fukushima improvements which should enhance the resilience of the PCV in a severe accident.

4.10.9 Further risk reduction

401. In its severe accident ALARP arguments, Hitachi-GE makes reference to the risks results from the Level 3 PSA (Ref. 53) and makes comparisons with Numerical Targets. I note that the overall Target 9 risk (a relevant measure for severe accidents) is in the 'tolerable if ALARP' region, between the BSO and the BSL.

402. Through assessment of the PSA, ONR has used RO-ABWR-0076 (Ref. 106) to gain further understanding of Hitachi-GE's position on ALARP for the overall design. One of ONR's key expectations in RO-ABWR-0076 was for Hitachi-GE to review the UK ABWR PSA results and consider whether it would be reasonably practicable to implement further safety measures.

403. The PSA inspector has reviewed the RO response and is broadly supportive of Hitachi-GE's approach (Ref. 4). Hitachi-GE has identified a number of areas where design improvements could be made and/or where the PSA will need to be further developed for the specific site. I would expect this work to include consideration of all severe accident measures, including accident management guidelines.

4.11 Safety case documentation

404. In this section I consider the overall adequacy of the suite of documents that comprise Hitachi-GE's severe accident safety case. The relevant documents comprise Chapter 26 of the PCSR and supporting references, together with other chapters of the PCSR.

405. My expectations are derived from TAG NS-TAST-GD-051 (Ref. 8), recognising that the principal expectations will inevitably be driven by the design basis safety case. I also take into account that expectations for accident management arrangements should be appropriate for GDA, recognising that much of the detailed work will need to be done by the future licensee.

406. On the general validity of the safety case, the documentation reflects the design intent for the UK ABWR. The severe accident submissions have been generated specifically for the UK ABWR over the course of GDA and the analysis is directly applicable to the UK ABWR. I welcome that the additional, post-Fukushima, severe accident features included in the UK ABWR feature prominently in the safety case.

407. Hitachi-GE's early PCSR and supporting documents were limited in scope. As a result, I raised RO-ABWR-0023 (Ref. 48) in Step 3 of GDA to set ONR's expectations for the development of a comprehensive severe accident safety case. Through the responses to the wide-ranging actions, Hitachi-GE has delivered against these expectations. In particular, it has:

- provided comprehensive analysis to demonstrate the effectiveness of severe accident measures and support the PSA;
- developed a severe accident safety case for shutdown and the SFP;
- considered the effect of hazards on the proposed accident management strategies;

- provided justification for computer codes used in the analysis;
- developed and justified a clear strategy for managing hydrogen;
- identified how the containment could be challenged in a severe accident;
- explained, at a level appropriate to GDA, proposed accident management strategies; and
- considered whether severe accident design features support an ALARP position.

408. In response to RO-ABWR-0023, the suite of severe accidents documentation has evolved significantly during GDA, in terms of both scope and detail. Where necessary, this has mirrored improvements made to the PSA, for example to consider additional plant damage states, shutdown modes and the SFP.

409. As part of GDA, Hitachi-GE has provided copies of example accident management guidelines, reflecting Japanese practice. It is not my expectation that detailed, UK ABWR-specific, procedures are provided in GDA. However, the procedures that have been provided give sufficient explanation of how severe accident progression would be managed. I consider that the severe accident safety case provided in GDA provides a strong foundation for the development of detail accident management guidelines by a future licensee.

410. Ref. 35 is a key document which provides an entry point into much of the lower level safety case documentation. I observe that there is significant overlap between this document and the severe accident aspects of Chapter 26. This is perhaps an area that would benefit from rationalisation after GDA, but it will be a matter for the future licensee to decide how best to move forward with the next version of the PCSR.

411. During GDA I put specific challenges to Hitachi-GE in relation to the management of hydrogen in the R/B. This was an area where Hitachi-GE's early submissions lacked clarity, but is a good example of how the safety case has been improved during GDA. In this case, Hitachi-GE's response was to provide documentation which:

- clearly set out its safety case objectives;
- systematically considered options for managing hydrogen;
- demonstrated that the proposed measures would be effective; and
- presented clear arguments as to why the chosen solution was ALARP.

412. In response, Hitachi-GE has also developed PCSR Chapter 26 so that it provides improved visibility of the links between the engineering specifications for severe accident measures and the analysis which provides substantiation of the measures. Whilst this is an area that could be improved further by a future licensee, I am satisfied that this is suitable for GDA and that any limitations have no impact on the adequacy of the basic design.

413. As might be expected at this stage, the engineering requirements for severe accident design provisions are less well developed than for the design basis. In particular I observe that:

- SSCs such as the RDCF, FLSS and FCVS are part of Hitachi-GE's safety case for design basis, beyond design basis and severe accidents, but the focus of the engineering documentation is principally on design basis requirements.
- Beyond design basis hazards withstand claims for severe accident mechanical systems are generally not considered in the engineering documentation, even though these may be required to operate in a severe accident initiated by a beyond design basis hazard.
- Severe accident withstand claims for some mechanical systems or components are not reflected in the engineering submissions.

- Claims for systems identified in the severe accident safety case as providing defence in depth, are generally not reflected in the engineering documentation.
- Severe accident claims for the primary containment function are not clearly identified, although the important elements of a safety case have been provided.

414. I consider that the position is sufficient for GDA to allow me to make a judgement on the severe accident safety case. However, to ensure that this is addressed by the future licensee I raise the following Assessment Finding:

- AF-ABWR-SA-11: Hitachi-GE's GDA safety case documentation provides limited and variable levels of detail on the claims and performance requirements placed on structures, systems and components (SSCs) in severe accident conditions, unless the SSC's role is specifically for severe accidents. The licensee shall identify so far as is reasonably practicable the expected requirements on SSCs in severe accidents to inform detail design work and equipment qualification work, as appropriate.

415. A fundamental requirement for a UK safety case is the demonstration of ALARP. Hitachi-GE has given specific and detailed consideration to whether it would be reasonably practical to introduce alternative severe accident design features. A series of studies have been provided by Hitachi-GE which consider:

- identification of relevant good practice;
- a qualitative comparison of the effectiveness of the UK ABWR design measures relative to the identified examples of relevant good practice;
- reference to PSA results and comparison with ONR's Numerical Targets.

416. I am satisfied that Hitachi-GE's ALARP documents present clear and reasoned arguments that are sufficient to allow me to make a judgement on the cases being made. Linked to this, Hitachi-GE has presented clear and reasoned arguments with the aim of supporting the claim that large and early releases have been practically eliminated for the UK ABWR.

417. The PCSR identifies the systems that would be required to deliver safety functions in a severe accident. These will be the subject of specific severe accident limiting conditions of operation (LCOs) to ensure that, when required to operate, the system availability and performance is consistent with the assumptions of the severe accident analysis. However, in general, LCOs for severe accident measures have not been identified in GDA. I accept that this will be best done by a future licensee and can be progressed through normal business after GDA.

418. Overall, I am satisfied that Hitachi-GE's severe accident safety case submissions are of suitable scope, detail and quality for GDA. For the purposes of my GDA assessment, the identified shortfalls and limitations that do exist are not sufficient to prevent me reaching conclusions on the adequacy of the severe accidents aspects of design of the UK ABWR.

4.12 Overseas regulatory interface

419. 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 outcomes, which can expedite assessment and helps promote consistency.

420. 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;
- implement the products it develops in order to facilitate the licensing of new reactors, including those being developed by Gen IV International Forum.

421. MDEP has established a Working Group for the ABWR, with representatives of the regulators from the UK, US, Japan and Sweden. (Note that whilst Sweden does not have plans to develop ABWR technology, it does operate existing BWRs which have design characteristics similar to those of the ABWR.) A Severe Accident Technical Expert Sub-Group (TESG) was established in 2015 and this has met at six-monthly intervals since then under the auspices of the Working Group. The TESG has discussed common areas of interest in a range of severe accident topics, including phenomena, modelling, engineered measures and accident management. The TESG has also held joint discussions with the separate Instrumentation & Control Sub-Group in the area of severe accident C&I. ONR's involvement in these activities has provided useful insight to inform my assessment.

422. A Common Position addressing issues related to the Fukushima accident (Ref. 114) has been prepared by the Working Group. Because not all of the participants have yet completed regulatory reviews of ABWR applications, the paper identifies common preliminary approaches to address potential safety improvements for ABWR plants, as well as common general expectations for new Nuclear Power Plants, as related to lessons learnt from the Fukushima Dai-ichi NPP accident. The common preliminary approaches are organised into seven sections, namely evolutionary improvements in safety, hazards, reliability of safety functions, accidents with core melt, spent fuel pools, emergency preparedness in design, and safety analysis. These approaches have been reflected in my assessment of the UK ABWR.

4.13 Assessment Findings

423. 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 1.

424. 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.

425. I have raised 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;
- to resolve this matter the plant needs to be at some stage of construction / commissioning.

426. Assessment Findings are residual matters that must be addressed by the Licensee and the progress of this will be monitored by the regulator.

5 CONCLUSIONS

427. This report presents the findings of my Step 4 severe accidents assessment of the Hitachi-GE UK ABWR.

428. During GDA, Hitachi-GE has provided comprehensive severe accident analysis. This has been developed specifically for the UK ABWR design in response to challenges from ONR. I have examined the safety case provided in Revision C of Hitachi-GE's PCSR and supporting references against the applicable expectations of the SAPs, TAGs and relevant international guidance published by the IAEA and WENRA. I am satisfied that:

- Hitachi-GE has provided suitable severe accident analysis of the UK ABWR, complementing the wider safety case. Relevant severe accident sequences have been identified for the all operating modes, including the reactor at power, the reactor during shutdown and the SFP.
- Hitachi-GE has analysed severe accidents using computer codes which are suitable for modelling the accident phenomena relevant to the UK ABWR.
- Hitachi-GE has identified the importance of the containment in its severe accident management strategy and has identified the relevant challenges to the containment.
- Severe accident analysis has been used to demonstrate the effectiveness of severe accident measures in preventing and/or mitigating accidents.
- Hitachi-GE has provided a clear explanation of how severe accidents would be managed and, for the purposes of GDA, has adequately described strategies and concepts that can be taken forward by the future licensee.
- Hitachi-GE has explained how learning from the accidents at Fukushima Dai-ichi has been used to positively influence the design of the UK ABWR.
- In accordance with UK and international expectations, I consider that
- Hitachi-GE has presented a thorough report on practical elimination of large or early fission product release for the design.
- Hitachi-GE has demonstrated that the severe accident design features support ALARP claims on the adequacy of the UK ABWR design.
- A severe accident safety case has been presented which is adequate for GDA.

429. Several Assessment Findings (Annex 1) were identified; these are for future licensees 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.

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

6 REFERENCES

1. Step 4 Assessment Plan for Severe Accidents. ONR-GDA-AP-15-019 Revision 0, TRIM Ref. 2015/350079

2. Safety Assessment Principles for Nuclear Facilities. 2014 Edition Revision 0. November 2014. http://www.onr.org.uk/saps/saps2014.pdf

3. Step 4 Assessment of Fault Studies for the UK ABWR. ONR-NR-AR-17-016 Revision 0, TRIM Ref. 2017/98169.

4. Step 4 Assessment of PSA for the UK ABWR. ONR-NR-AR-17-014 Revision 0, TRIM Ref. 2017/98147.

5. Step 2 Assessment of the Probabilistic Safety Analysis (PSA) and Severe Accident Analysis (SAA) of Hitachi-GE's UK Advanced Boiling Water Reactor (UK ABWR), ONR-GDA.-AR-14-003 Revision 0, August 2014, TRIM Ref. 2014/180670.

6. GDA Step 3 Assessment of the Severe Accidents of Hitachi GE's UK Advanced Boiling Water Reactor (UK ABWR), ONR-GDA-AR-15-004 Revision 0, November 2015, TRIM Ref 2015/230441.

7. Guidance on Mechanics of Assessment within the Office for Nuclear Regulation (ONR) – TRIM Ref. 2013/204124

8. ONR TAGs:
   Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable)
   NS-TAST-GD-005 Revision 8. ONR. July 2017
   Severe Accident Analysis
   NS-TAST-GD-007 Revision 3. ONR. September 2017
   Probabilistic Safety Analysis
   NS-TAST-GD-030
   Validation of Computer Codes and Calculation Methods
   NS-TAST-GD-042 Revision 3. ONR. July 2016
   The Purpose, Scope, and Content of Safety Cases
   NS-TAST-GD-051 Revision 4. ONR. July 2016
   http://www.onr.org.uk/operational/tech_asst_guides/index.htm

9. IAEA
   IAEA Safety Standards Series – Safety of Nuclear Power Plants: Design, Specific Safety Requirements (SSR) 2/1 Revision 1, 2016.
   Safety Guide No. No. NS-G-2.15 - Severe Accident Management Programmes for Nuclear Power Plants, 2009
   www.iaea.org

10. Western European Nuclear Regulators' Association.
    Reactor Harmonization Working Group - WENRA Safety Reference Levels for Existing Reactors. WENRA. September 2014.
    WENRA Statement on Safety Objectives for New Nuclear Power Plants, November 2010
    Reactor Harmonization Working Group - Safety of new NPP designs, March 2013
    www.wenra.org

11. Vienna Declaration on Nuclear Safety On principles for the implementation of the objective of the Convention on Nuclear Safety to prevent accidents and mitigate radiological consequences. CNS/DC/2015/2/Rev.1, February 9, 2015
    https://www.iaea.org/sites/default/files/cns_viennadeclaration090215.pdf

12. The United Kingdom's Seventh National Report on Compliance with the Obligations of the Convention on Nuclear Safety, Department for Business, Energy & Industrial Strategy, August 2016. http://www-ns.iaea.org/downloads/ni/safety_convention/7th-review-meeting/uk_03052017_nr-7th-rm.pdf

13. HM Japanese earthquake and tsunami: Implications for the UK nuclear industry - Final Report, HM Chief Inspector of Nuclear Installations, September 2011 http://www.onr.org.uk/fukushima/final-report.pdf

14. Japanese earthquake and tsunami: Implications for the UK nuclear industry - Interim Report, HM Chief Inspector of Nuclear Installations, May 2011 http://www.onr.org.uk/fukushima/interim-report.pdf

15. European Council "Stress Tests" for UK nuclear power plants - National Final Report. ONR, December 2011 http://www.onr.org.uk/fukushima/stress-tests-301211.pdf

16. The Fukushima Daiichi Accident - Report by the Director General. International Atomic Energy Agency, 2015 http://www-pub.iaea.org/MTCD/Publications/PDF/Pub1710-ReportByTheDG-Web.pdf

17. Requirements on Severe Accident Analyses for ABWR Plant Design, ONR195 WP8.1 June 2014, TRIM Ref. 2014/276321.

18. Review of Hitachi-GE's Topic Report Regarding Severe Accident Analyses, ONR195 WP8.2 July 2014, TRIM Ref. 2014/453621.

19. Best Practices for Independent Confirmatory Severe Accident Analyses for ABWR Plant Design, ONR195 WP8.3 November 2014, TRIM Ref. 2014/453622.

20. Independent Confirmatory Severe Accident Analysis for the UK ABWR. Amec Foster Wheeler report 203171-AA-0014 Issue 2, June 2017, TRIM Ref. 2017/236561.

21. RCCV Ultimate Capacity Assessment of the ABWR Level 2 PSA, Amec Foster Wheeler, ONR/301/R/003 Issue 02, 31st May 2017, TRIM Ref. 2017/211140.

22. Generic PCSR Chapter 22: Emergency Preparedness, HE-GD-0044 Rev.C, TRIM Ref. 2017/335086.

23. Generic PCSR Chapter 1: Introduction, XE-GD-0214 Rev.C, TRIM Ref. 2017/335101.

24. Generic PCSR Chapter 26: Beyond Design Basis and Severe Accident Analysis, AE-GD-0148 Rev.C, TRIM Ref. 2017/335083.

25. Generic PCSR Chapter 5: General Design Aspects, XE-GD-0645 Rev.C, TRIM Ref. 2017/335111.

26. Generic PCSR Chapter 9: General Description of the Unit (Facility), SE-GD-0136 Rev.C, TRIM Ref. 2017/335019.

27. Generic PCSR Chapter 10: Civil Works and Structures, LE-GD-0035 Rev.C, TRIM Ref. 2017/335066.

28. Generic PCSR Chapter 13: Engineered Safety Features, XE-GD-0647 Rev.C, TRIM Ref. 2017/335042.

29. Generic PCSR Chapter 14: Control and Instrumentation, 3E-GD-A00063 Rev.C, TRIM Ref. 2017/335110.

30. Generic PCSR Chapter 16: Auxiliary Systems, XE-GD-0649 Rev.C, TRIM Ref. 2017/335093.

31. Generic PCSR Chapter 23: Reactor Chemistry, WPE-GD-0058 Rev.C, August 2017, TRIM Ref. 2017/335055.

32. Generic PCSR Chapter 24: Design Basis Analysis, UE-GD-0208 Rev.C, August 2017, TRIM Ref. 2017/335109.

33. Generic PCSR Chapter 25: Probabilistic Safety Assessment, AE-GD-0171 Rev.C, August 2017, TRIM Ref. 2017/335076.

34. Generic PCSR Chapter 28: ALARP Evaluation, SE-GD-0140 Rev.C, August 2017, TRIM Ref. 2017/335068.

35. Topic Report on Severe Accident Phenomena and Severe Accident Analysis, AE-GD-0102 Rev. H, TRIM Ref. 2017/163822.

36. Severe Accident Safety Case for Shutdown Reactor and SFP, AE-GD-0633 Rev.1, June 2016, TRIM Ref. 2016/263197.

37. Demonstration of Practical Elimination of Early or Large Fission Product Release for UK ABWR, AE-GD-0992 Rev.0, TRIM Ref. 2017/254924.

38. Basic Requirement Specification of Severe Accident Management Measures, AE-GD-0367 Rev.3, TRIM Ref. 2017/249265.

39. Basis of Safety Cases on Severe Accident Mechanical Systems, SE-GD-0219 Rev.2, TRIM Ref. 2017/256854.

40. Containment Venting Strategy in UK ABWR, AE-GD-0524 Rev.2, TRIM Ref. 2017/263519.

41. An ALARP Evaluation on Methods/Technologies for the Mitigation of Molten Core Concrete Interactions for the UK ABWR, AE-GD-0437 Rev.2, TRIM Ref. 2016/341983.

42. Flammable Gas Control and Supporting Analysis in UK ABWR, AE-GD-0457 Rev.4, TRIM Ref.2017/168046.

43. ALARP Discussion on Flammable Gas Control, AE-GD-0438 Rev.2, TRIM Ref. 2017/173494.

44. Containment Performance Analysis Report in UK ABWR, AE-GD-0561 Rev.3, TRIM Ref. 2017/256271.

45. Applicability of the HM Chief Inspector's Recommendations and ONR's Stress Test Findings to UK ABWR Design, AE-GD-0505 Rev. 2, TRIM Ref. 2017/196971.

46. Accident Management Guideline (After Core Damaged) for UK ABWR, AE-GD-0363 Rev.0, TRIM Ref. 2015/123908.

47. Consideration of Fuel Coolant Interactions for UK ABWR, AE-GD-0382 Rev.0, TRIM Ref. 2015/195981.

48. RO-ABWR-0023 - Severe Accident Safety Case http://www.onr.org.uk/new-reactors/uk-abwr/reports/ro-abwr-0023.pdf

49. RO-ABWR-0039 - UK learning from the Fukushima Dai-ichi events
http://www.onr.org.uk/new-reactors/uk-abwr/reports/ro-abwr-0039.pdf

50. RO-ABWR-0046 - UK ABWR Containment Performance Analyses for Severe Accidents
http://www.onr.org.uk/new-reactors/uk-abwr/reports/ro-abwr-0046.pdf

51. Topic Report on Internal Event at Power Level 2 PSA, AE-GD-0258 Rev.4, TRIM Ref. 2017/147410.

52. Topic Report on Beyond Design Basis Analysis, AE-GD-0473, Rev.5, TRIM Ref. 2017/299917.

53. Level 3 PSA for UK ABWR: Accident Consequences, Compiled Plant Risk Profile and Assessment Against Numerical Risk Targets 7, 8 and 9 for GDA, HE-GD-0306 Rev.2, TRIM Ref. 2017/291576.

54. Topic Report on Physics Models and Benchmarking of MAAP Code, AE-GD-0144 Rev.B, TRIM Ref. 2016/30671.

55. Phenomenological Uncertainty Studies on Severe Accident Analysis, AE-GD-0372 Rev. 2, TRIM Ref. 2016/347553.

56. MAAP Input Deck and Parameter File for Severe Accident Analysis in UK ABWR, AE-GD-0769 Rev.0, TRIM Ref. 2016/364182.

57. Sensitivity Analysis and Uncertainty Analysis Report for Internal Event at Power PSA, AE-GD-0345 Rev.2, TRIM Ref. 2017/55002.

58. MELCOR Computer Code Manuals, Vol. 1: Primer and Users' Guide, Version 2.1.6840 2015, SAND2015-6691 R, August 2015.

59. A MELCOR model of Fukushima Daiichi Unit 1 accident, T. Sevón, Annals of Nuclear Energy 85 (2015) pp. 1–11.

60. A MELCOR model of Fukushima Daiichi Unit 3 accident, T. Sevón, Nuclear Engineering and Design 284 (2015) pp. 80–90.

61. Step 4 Fuel and Core Design Assessment of the UK ABWR. ONR-NR-AR-17-019 Revision 0, TRIM Ref. 2017/492101.

62. Report on Physics Model and Benchmark of GOTHIC Code for Supporting Analysis of Flammable Gas Control, AE-GD-0871 Rev.0, TRIM Ref. 2016/504925.

63. Nuclear Energy Agency Committee on the Safety of Nuclear Installations. International Standard Problem ISP-47 on Containment Thermal Hydraulics, final report, NEA/CSNI/R(2007)10, September 2007,
https://www.oecd-nea.org/

64. NEA-1838 JASMINE V.3,
http://www.oecd-nea.org/tools/abstract/detail/nea-1838/

65. A Study of Chemistry Effects in UK ABWR Fault Studies, HE-GD-0175 Rev.2, TRIM Ref. 2017/295457.

66. Step 4 Assessment of Reactor Chemistry for the UK ABWR. ONR-NR-AR-17-020 Revision 0, TRIM Ref. 2017/98232.

67. In-vessel and Ex-vessel Corium Behaviour (Response to RQ-ABWR-0865), AE-GD-0703 Rev.0, TRIM Ref. 2016/251246.

68. Step 4 Assessment of Civil Engineering for the UK ABWR. ONR-NR-AR-17-013 Revision 0, TRIM Ref. 2017/98126.

69. Step 4 Assessment of Structural Integrity for the UK ABWR. ONR-NR-AR-17-037 Revision 0, TRIM Ref. 2017/98277.

70. Nuclear Energy Agency Committee on the Safety of Nuclear Installations. OECD Research Programme on Fuel-Coolant Interaction Steam Explosion Resolution for Nuclear Applications - SERENA, NEA/CSNI/R(2007)11, September 2007, https://www.oecd-nea.org/

71. RO-ABWR-0043 - Demonstration of the adequacy of pH control in the Suppression Pool during accident conditions http://www.onr.org.uk/new-reactors/uk-abwr/reports/ro-abwr-0043.pdf

72. RO-ABWR-0066 - Demonstration of suitable and sufficient consideration of chemistry effects in fault analysis http://www.onr.org.uk/new-reactors/uk-abwr/reports/ro-abwr-0066.pdf

73. Topic Report on ALARP Assessment for Steam Generation Resulting from a Loss of Decay Heat Removal from the SFP and Open RPV, AE-GD-0989 Rev.0, TRIM Ref. 2017/206561.

74. ALARP Discussion on Methods of Further Capturing or Reducing Fission Products Inside Containment, AE-GD-0447 Rev.0, TRIM Ref. 2015/406830.

75. ALARP Discussion on Design of the Containment Head Flange and Other Systems to Protect from Containment Leakage, AE-GD-0446 Rev.0, TRIM Ref. 2015/325555.

76. RO-ABWR-0059 - Provision of Water Cooling for the RCCV Drywell Head of the UK ABWR http://www.onr.org.uk/new-reactors/uk-abwr/reports/ro-abwr-0059.pdf

77. Internal Structures of Reinforced Concrete Containment Vessel Structural Design Report, DD-GD-0005, Rev. 5, TRIM Ref. 2017/131508.

78. RPV Pedestal Design Report, DD-GD-0016 Rev.1, TRIM Ref. 2017/173486.

79. Effects of Steam Explosions from ex-Vessel Fuel-Coolant Interactions (Response to RQ-ABWR-1236), AE-GD-0914 Rev.0, TRIM Ref. 2017/79180.

80. Final Safety Evaluation, Report Related to the Certification of the Advanced Boiling Water Reactor Design - Main Report. NUREG-1503 Volume 1.

81. Light Water Reactor Hydrogen Manual, NUREG/CR-2726, SAND82-1137, June 1983. https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr2726/

82. Carbon Monoxide - Hydrogen Combustion Characteristics in Severe Accident Containment Conditions, NEA CSNI/R(2000)10, March 2000. http://www.oecd.org/officialdocuments/publicdisplaydocumentpdf/?cote=NEA/CSNI/R(2000)10&docLanguage=En

83. SA Claims on the Support Function of the Pedestal Wall (Response to RQ-ABWR-0557), AE-GD-0471 Rev.0, TRIM 2015/273808.

84. Design and Location of Vacuum Breakers (Response to RQ-ABWR-1385), AE-GD-0982 Rev.0, TRIM Ref. 2017/197047.

85. Topic Report on Process Radiation Monitoring System (Containment Radiation Monitor), 3E-GD-K109 Rev.1, TRIM 2017/134135.

86. Topic Report on Overview of Backup Building Safety Case, XE-GD-0750 Rev.0, TRIM 2017/268203.

87. Topic Report on Severe Accident C&I System, 3E-GD-A0290 Rev.2, TRIM Ref. 2017/235593.

88. RPV Head Spray Line During Severe Accidents (Response to RQ-ABWR-1160), AE-GD-0893 Rev.0, TRIM Ref. 2017/43750.

89. Failure Modes of SRV Relief Function in Severe Accident Conditions (Response to RQ-ABWR-1299), AE-GD-0954 Rev.0, TRIM Ref. 2017/120030.

90. Basis of Safety Case Claims for the Lower Drywell Flooder and other Severe Accident Measures (Response to RQ-ABWR-1468), AE-GD-0995 Rev.0, TRIM Ref. 2017/249069.

91. Emergency Procedure (Symptom Based) for UK ABWR, AE-GD-0364 Rev.0, TRIM Ref. 2015/123916.

92. Guideline During Prolonged Station Black Out for UK ABWR, AE-GD-0365 Rev.0, TRIM Ref. 2015/123935.

93. Guideline for Monitor and Control SFP for UK ABWR, AE-GD-0366 Rev.0, TRIM Ref. 2015/123945.

94. Concept of Countermeasures for Abnormal events during Plant Outage, AE-GD-0826 Rev.1, TRIM Ref. 2017/167572.

95. RO-ABWR-0067 - UK ABWR Generic Site Envelope – External Flooding and Beyond Design Basis events
http://www.onr.org.uk/new-reactors/uk-abwr/reports/ro-abwr-0067.pdf

96. Step 4 External Hazards Assessment for the UK ABWR. ONR-NR-AR-17-018 Revision 0, TRIM Ref. 2017/98329.

97. Post-Accident Management Arrangements (Response to RQ-ABWR-0666), AE-GD-0575 Rev.0, TRIM 2015/454301.

98. Monitoring Parameters for Severe Accident Management in UK ABWR, AE-GD-0774 Rev.1, TRIM Ref. 2017/167530.

99. Further Clarification on Backup Building Power Supplies for Severe Accident Management (Response to RQ-ABWR-1307), AE-GD-0955 Rev.0, TRIM Ref. 2017/120034.

100. A Short Summary to Close out RO-ABWR-0067. AE-GD-0969, Revision 1, TRIM Ref. 2017/206520

101. Step 4 Electrical Engineering Assessment for the UK ABWR. ONR-NR-AR-17-018 Revision 0, TRIM Ref. 2017/98198.

102. Report of the Operational Safety Review Team (OSART) Mission to the Units 6 and 7 of Kashiwazaki-Kariwa Nuclear Power Plant Japan 29 June – 13 July 2015. NSNI/OSART/183/2015, International Atomic Energy Agency, 2015.

103. Further Clarification on Severe Accident C&I (Response to RQ-ABWR-1336), 3E-GD-A0472 Rev.0, TRIM Ref. 2017/143036.

104. Severe Accident Venting Strategy and Hydrogen Management for Vent Lines, AE-GD-0939 Rev.0, TRIM Ref. 2017/139536.

105. Topic Report on Emergency Response Facility C&I System, 3E-GD-A0436 Rev.0, TRIM Ref. 2017/141331.

106. RO-ABWR-0076 - PSA ALARP Demonstration and Optioneering http://www.onr.org.uk/new-reactors/uk-abwr/reports/ro-abwr-0076.pdf

107. Risk informed regulatory decision making, ONR, June 2017 http://www.onr.org.uk/documents/2017/risk-informed-regulatory-decision-making.pdf

108. ALARP Discussion on Passive Methods of Core or Containment Cooling, AE-GD-0398 Rev. 0, TRIM Ref. 2015/199687.

109. ALARP Discussion on Methods for Further Increasing Grace/Response Times, AE-GD-0553 Rev.0, TRIM Ref. 2015/406807.

110. UK ABWR Nuclear Safety and Environmental Design Principles (NSDEPs), XD-GD-0046, Rev.1, TRIM Ref. 2017/269935.

111. Behaviour and Effectiveness of Reflective Metal Insulation in Accident Scenarios (Response to RQ-ABWR-1223), AE-GD-0900 Rev.1, TRIM Ref. 2017/86813.

112. International Atomic Energy Agency (IAEA). Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants, IAEA-TECDOC-1791, May 2016. http://www-pub.iaea.org/MTCD/publications/PDF/TE-1791_web.pdf

113. File note: Radiological Consequence Calculations for ABWR GDA, Revision 11, ONR, June 2017. TRIM 2017/238892.

114. MDEP Design-Specific Common Position CP-ABWRWG-01 - Common Position Addressing Issues Related to the Fukushima Daiichi Nuclear Power Plant Accident, 11 October 2016.

Annex 1

Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-ABWR-SA-01	Failure of the pedestal wall has been identified by Hitachi-GE as a potential challenge to the containment in a severe accident. In GDA, Hitachi-GE has not presented detailed design calculations to justify the failure criterion for the pedestal wall when subject to molten core-concrete interaction. The licensee shall substantiate the failure criterion for the pedestal wall in severe accidents, including specific consideration of challenges to the pedestal wall structure from molten core material which may break though into the pedestal wall vent pipes.	4.5.7
AF-ABWR-SA-02	Hitachi-GE has assumed that the vacuum breakers would be robust against severe accident conditions, thus preventing suppression pool bypass. However, specific safety case claims or performance requirements for the vacuum breakers in severe accident conditions have not been identified in the GDA safety case documentation. The licensee shall identify the requirements placed on the vacuum breakers by the severe accident safety case and demonstrate that these can be met by the final design.	4.5.9
AF-ABWR-SA-03	Hitachi-GE has identified the theoretical possibility of re-criticality in a severe accident during re-flooding of the reactor pressure vessel, resulting in potential challenges to the primary containment. Hitachi-GE has presented limited analysis of the conditions which could give rise to re-criticality. To inform site-specific accident management guidelines, the licensee shall perform sufficient additional analysis to identify the range of conditions that could lead to a possible re-criticality. For the conditions which could potentially result in re-criticality, the licensee shall consider the requirements for any design provisions which could reduce the risk of re-criticality so far as is reasonably practicable.	4.6.3.2
AF-ABWR-SA-04	Ensuring the continuing integrity of the primary containment by protecting it from over-pressurisation is a vital objective for severe accident measures and management strategies. Hitachi-GE's severe accident analysis has shown that the assumed set-point for the containment overpressure protection system would not always ensure that pressure in the drywell remains below the containment ultimate failure pressure. For accident sequences where venting is claimed as an effective severe accident measure, the licensee shall optimise the containment over-pressure protection system opening set-point to ensure that containment pressures remain below the ultimate failure pressure so far as is reasonably practicable. This shall take into account containment conditions in severe accidents, including consideration of potential static and dynamic pressure differences between the drywell and wetwell.	4.6.3.4
AF-ABWR-SA-05	In the absence of detailed design information during GDA, Hitachi-GE has made assumptions about achievable flow rates in its demonstrations of the effectiveness of primary containment vessel venting in severe accidents. The licensee shall demonstrate that the final design of the filtered containment vent system can meet the safety case claims placed on it by those severe accident sequences which credit venting.	4.6.3.4

AF-ABWR-SA-06	Hitachi-GE's GDA analysis of the effectiveness of hydrogen management measures in the primary containment and reactor building has been based on provisional design information for passive autocatalytic recombiners. The analysis supports Hitachi-GE's hydrogen management strategy for design basis loss of coolant accidents and reactor severe accidents. The licensee shall update the hydrogen management safety case to reflect the design and performance characteristics of the recombiners selected in the final design, and reconfirm that the hydrogen management objectives are met.	4.6.5.2
AF-ABWR-SA-07	Hitachi-GE has identified in GDA a need to open the large equipment door in some severe accident conditions as part of the hydrogen management strategy. However the practicalities of how this will be done have not been determined due to limitations in GDA scope. The licensee shall determine the arrangements for opening of the reactor building large equipment door in accident conditions, taking appropriate steps to ensure that the risks to both the public (from a major event escalation caused by not opening the door) and workers performing crucial tasks are considered and reduced to ALARP.	4.6.5.4
AF-ABWR-SA-08	Hitachi-GE has identified several lessons and learning points from the Fukushima Dai-ichi accident that are site-specific or matters for the licensee to consider, which cannot be fully addressed in GDA. The licensee shall review relevant lessons and learning points identified as being out of GDA scope in Hitachi-GE document AE-GD-0505 Rev.2 and demonstrate that these have been addressed in the design and proposed operation of the site-specific plant.	4.8.2
AF-ABWR-SA-09	For the reactor building, Hitachi-GE has included the provision to connect mobile power units to support Class 1 systems. However, the Severe Accident Control & Instrumentation system is powered by the backup building electrical power system. A failure of backup building power sources is a potential way for a fault condition to escalate to a severe accident scenario, resulting in the loss of severe accident control and instrumentation functions. As part of its work to develop a final design for the backup building, the licensee shall consider whether it is ALARP to provide a capability for mobile power supply sources to be connected to the Severe Accident Control & Instrumentation system, to ensure that control and monitoring of severe accident systems can be maintained in circumstances where the fixed backup building power sources have failed.	4.8.7
AF-ABWR-SA-10	To meet UK and international expectations post-Fukushima, Hitachi-GE has provided a demonstration which argues that the generic UK ABWR design practically eliminates large or early releases. The extent to which hazards, and therefore the completeness of any practical elimination claim, can be considered in GDA is limited. In particular, external hazards will present an additional contribution to the site-specific risk profile. The licensee shall review and update as appropriate the deterministic and probabilistic arguments that support the claim that large or early releases have been practically limited on a site-specific basis, notably to consider the risks associated with the site-specific beyond design basis hazard profile.	4.9.6
AF-ABWR-SA-11	Hitachi-GE's GDA safety case documentation provides limited and variable levels of detail on the claims and performance requirements placed on structures, systems and components (SSCs) in severe accident conditions, unless the SSC's role is specifically for severe accidents. The licensee shall identify so far as is reasonably practicable the expected requirements on SSCs in severe accidents to inform detail design work and equipment qualification work, as appropriate.	4.11

Assessment Finding Number	Assessment Findings raised in other Assessment Reports which are referred to in this report	Report Section Reference
AF-ABWR-RC-22	The UK ABWR generic safety case makes important assumptions about the rates and quantities of flammable gases generated during design basis faults. These assumptions differ from other established international practices and have not been adequately justified during GDA. The assumptions directly influence the design of safety measures which mitigate the impact of flammable gases. The licensee shall review and update the UK ABWR safety case, to provide an adequate justification to show the assumptions made about the production rates of hydrogen and oxygen from water radiolysis, during a design basis Loss of Coolant Accident, are applicable to UK ABWR.	4.6.5.2
AF-ABWR-RC-23	The UK ABWR generic safety case adopts a simplified approach to model the Passive Autocatalytic Recombiners used in the design. During GDA, this approach was adequate to demonstrate the design concept, but it has not been demonstrated to be adequate to model many of the other potentially important processes which can occur during their use. The licensee shall review and update the UK ABWR safety case to demonstrate that the modelling of Passive Autocatalytic Recombiners used in the flammable gas analysis, adequately accounts for the physical and chemical processes taking place, which are not limited to recombination, combustion, accumulation and thermal effects.	4.6.5.2 & 4.6.5.3
AF-ABWR-EH-04	As a result of the assumptions made in GDA, a future licensee shall consider and implement adequate water sealing and/or elevation of the Heat Exchanger Building and transformer, Reactor Building, Control Building, Electrical Diesel Generator Buildings, and Backup Building. The determination of requirements shall be based on site-specific evaluation of external flooding beyond the design basis in accordance with the principles of ALARP.	4.7.3
AF-ABWR-FS-10	The UK ABWR secondary containment is provided with a blowout panel to protect the civil structure from high pressure steam releases. However, over the course of GDA the number of claims on this panel has expanded from the original design intent. The licensee shall review and optimise the opening set-point of the secondary containment blowout panel, cognisant of the safety requirements for high pressure piping ruptures, spent fuel pool (SFP) and reactor design basis loss of active cooling events resulting in steam generation, and the management of radioactivity and hydrogen in severe accidents.	4.6.5.4