



New Reactors Division

Step 4 Assessment of Chemistry for the UK Advanced Boiling Water Reactor

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EXECUTIVE SUMMARY

Hitachi-GE Nuclear Energy Ltd is the designer and GDA Requesting Party (RP) 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 chemistry.

The scope of the Step 4 assessment is to review the safety aspects of the UK ABWR in greater detail, by examining the evidence supporting the claims and arguments made in the safety documentation, building on the assessments already conducted during Steps 2 and 3. In addition, I have provided a judgement on the adequacy of the information contained within the Pre-Construction Safety Report (PCSR) and supporting documentation.

My assessment conclusions are:

- Hitachi-GE has identified a suitable set of claims on the operating chemistry, for all modes of operation, including commissioning. They have mostly provided sufficient supporting evidence to demonstrate that the claims can be achieved by the generic design. The engineered design of systems which provide chemistry-related safety functions are adequate to enable them to be delivered in a reliable and safe manner, consistent with the requirements of the safety case.
- The major chemistry parameters which would be expected to form part of the generic plant Operating Rules have been identified. While significant further work will need to be undertaken by a future licensee to fully define and substantiate the associated limits for many of these, the basic approach adopted for GDA is sound.
- Where chemistry-related assumptions are used in the supporting safety analysis, they have been identified. The adequacy and/or sensitivity to them has been demonstrated in a proportionate manner, and the related uncertainty has been treated in a reasonable manner.
- An appropriate balance has been achieved in demonstrating that overall plant relevant risks have been reduced SFAIRP, for those risks where chemistry can have an influence.
- The scope, structure and content of the generic safety case broadly meet my expectations for this stage of the project. From a chemistry perspective, an adequate generic PCSR has been produced, which provides a summary and links to the underlying evidence. However, significant further work will need to be undertaken by a future licensee to further develop the chemistry aspects of the PCSR. Collectively though, the generic safety case documentation provides an appropriate starting point for a future licensee to develop the UK ABWR safety case and design to facilitate the transition to operations.

My judgement is based upon the following factors:

- A detailed and in depth technical assessment, on a sampling basis, of the full scope of safety submissions at all levels within Hitachi-GE's hierarchy of generic safety case documentation.
- Independent information, reviews and analysis of key aspects of Hitachi-GE's generic safety case undertaken by Technical Support Contractors (TSCs) and my comparison and judgement thereof.
- Detailed technical interactions on many occasions with Hitachi-GE, alongside my assessment of the responses to the substantial number of Regulatory Queries (RQs) and Regulatory Observations (ROs) I raised during GDA, in addition to the response to Regulatory Issue (RI) RI-ABWR-0001, on radiological source terms.

Based on my assessment, I consider that certain matters remain, which are for a future licensee to consider and take forward in its site-specific safety submissions. These matters do not undermine the UK ABWR generic design and safety submissions, but require licensee input/decisions to be made at a specific site. These matters have been captured in 26 Assessment Findings. They broadly fall into three categories:

- Developing the full suite of chemistry-related Operating Rules, including the definition of appropriate limits and how these are linked to the safety case.
- Providing additional evidence and justifications for aspects of the generic UK ABWR design or safety case which are not sufficiently detailed at this stage; or where my assessment has identified further work is required to substantiate the decisions taken at this time.
- Justification for when important decisions are required by the future licensee for how they wish to operate the plant, the impacts this has on safety, and the requirement to demonstrate relevant risks continue to be reduced SFAIRP.

Similarly, I also raised 12 Minor Shortfalls. These are less significant matters than Assessment Findings, and represent aspects I have noted during my assessment of the UK ABWR generic safety case and design, which merit some further consideration by a future licensee.

In conclusion, based on my sample assessment undertaken during GDA, I am satisfied with the claims, arguments and evidence laid down within the PCSR and supporting documentation for chemistry. I consider that from a chemistry perspective, the Hitachi-GE UK ABWR design is suitable for construction in the UK subject to future permissions and permits being secured.

LIST OF ABBREVIATIONS

ACoP	Approved Code of Practice
ActP	Actinide
ALARP	As Low As Reasonably Practicable
AO	Air Off-take
AP	Activation Product
ATWS	Anticipated Transient Without Shutdown
BDB	Beyond Design Basis
BDL	Bottom Drain Line
BE	Best Estimate
BOC	Beginning of Cycle
BoSC	Basis of Safety Case
BSO	Basic Safety Objective
BWRVIA	BWR Vessel and Internals Application
CA	Cycle Average
CAMS	Containment Atmosphere Monitoring System
CCW	Component Cooling Water
CD	Condensate Demineraliser
CFDW	Condensate and Feedwater
CF	Condensate Filter
C&I	Control and Instrumentation
CIPS	Chloride Ingress Protection System
CILC	Crud Induced Localised Corrosion
CP	Corrosion Product
CPS	Condensate Purification System
CRD	Control Rod Drive
CRM	Containment Radiation Monitor
CRUD	Chalk River Unidentified Deposit
CS	Carbon Steel
CST	Condensate Storage Tank
CUW	Reactor Water Clean-up
DAC	Design Acceptance Confirmation
DB	Design Basis
DBA	Design Basis Analysis
DBF	Design Basis Fault
DF	Decontamination Factor
DH	Dissolved Hydrogen
DMW	Dissimilar Metal Weld
DO	Dissolved Oxygen
DRP	Design Reference Point
DSEAR	Dangerous Substances and Explosive Atmospheres Regulations 2002

LIST OF ABBREVIATIONS

DST	Deposit Source Term
DZO	Depleted Zinc Oxide
EA	Environment Agency
EC	Erosion-corrosion
ECCS	Emergency Core Cooling System
ECP	Electrochemical Corrosion Potential
EECW	Emergency Equipment Cooling Water System
EHWC	Early Hydrogen Water Chemistry
ELTNC	Enhanced Low Temperature NobleChem™
EOC	End of Cycle
EP	Electro-Polishing
EPRI	Electric Power Research Institute
ES	Extraction Steam
EUST	End User Source Term
F/D	Filter Demineraliser
FAC	Flow Accelerated Corrosion
FCI	Fuel-coolant Interaction
FCS	Flammability Control System
FCVS	Filtered Containment Venting System
FLSS	Flooder System of Specific Safety Facility
FMCRD	Fine Motion Control Rod Drives
FP	Fission Product
FPC	Fuel Pool Cooling and Clean-up system
FW	Feedwater
FWH	Feedwater Heater
GDA	Generic Design Assessment
HCW	High Chemical Impurities Waste system
HD	Heater Drain
HECW	HVAC Emergency Cooling Water System
HEPA	High Efficiency Particulate Air
HFT	Hot Functional Testing
HNCW	HVAC Normal Cooling Water System
HOIS	Hydrogen and Oxygen Injection System
HP	High Pressure
HPCP	High Pressure Condensate Pump
HPDT	High Pressure Drain Tank
HPFH	High Pressure Feedwater Heater
HPHD	High Pressure Heater Drain
HVAC	Heating, Ventilation and Air Conditioning
HWC	Hydrogen Water Chemistry

LIST OF ABBREVIATIONS

IAEA	The International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
IGSCC	Intergranular Stress Corrosion Cracking
J-ABWR	Japanese Advanced Boiling Water Reactor
JANTI	Japan Nuclear Technology Institute
JANSI	Japan Nuclear Safety Institute
JSME	Japanese Society for Mechanical Engineering
KK-6/7	Kashiwazaki-Kariwa Nuclear Power Station
LAS	Low Alloy Steel
LCO	Limits and Conditions of Operation
LCW	Low Chemical Impurities Waste System
LFL	Lower Flammable Limit
LOCA	Loss of Coolant Accident
LP	Low Pressure
LPDT	Low Pressure Drain Tank
LPHD	Low Pressure Heater Drain
LPRM	Local Power Range Monitor
LTNC	Low Temperature NobleChem™
LWMS	Liquid Waste Management System
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
MCCI	Molten Core Concrete Interaction
MDEP	Multi-national Design Evaluation Programme
MMS	Mitigation Monitoring System
MS	Main Steam
MSR	Moisture Separator Re-heater
MUWC	Make-up Water Condensate System
MUWP	Make-up Water Purified System
NBA	Nickel-base Alloy
NEDO	New Energy and Industrial Technology Development Organization
NMCA	Noble Metal Chemical Addition
NMIS	Noble Metal Injection System
NNL	National Nuclear Laboratory
NRA	(Japanese) Nuclear Regulation Authority
NRW	Natural Resources Wales
NUREG	US Nuclear Regulatory Commission Regulation
NWC	Normal Water Chemistry
OSCGS	Outer Secondary Containment Grab Sampling
OECD-NEA	Organisation for Economic Co-operation and Development Nuclear Energy Agency

LIST OF ABBREVIATIONS

OG	Off-gas
OG PO HM	Off-gas System Post-recombination Hydrogen Monitor
OI	Oxygen Injection
OLNC	On-Line Noble Chem™
ONR	Office for Nuclear Regulation
OPEX	Operating Experience
OR	Operating Rule
ORE	Operational Radiation Exposure
ORIGEN	Oak Ridge Isotope Generator
PASSAM	Passive and Active Systems on Severe Accident source term Mitigation
P&ID	Process and Instrumentation Diagram
PCI	Pellet-clad Interaction
PCSR	Pre-construction Safety Report
PCV	Primary Containment Vessel
PLR	Primary Loop Recirculation
ppb	Parts Per Billion
ppm	Parts Per Million
PRM	Process and Radiation Monitoring
PrST	Process Source Term
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
PST	Primary Source Term
PWHT	Post Weld Heat Treatment
PWR	Pressurised Water Reactor
R/B	Reactor Building
RCCV	Reinforced Concrete Containment Vessel
RCS	Reactor Coolant System
RCW	Reactor Building Cooling Water System
RGP	Relevant Good Practice
RHR	Residual Heat Removal
RI	Regulatory Issue
RINs	Reactor Internals
RIP	Reactor Internal Pump
RO	Regulatory Observation
RP	Requesting Party
RPV	Reactor Pressure Vessel
RPVHS	Reactor Pressure Vessel Head Spray
RSW	Reactor Building Service Water System
RQ	Regulatory Query
RVI	Reactor Vessel Instrument

LIST OF ABBREVIATIONS

SA	Severe Accident
SAM	Sampling and Monitoring
SAPs	Safety Assessment Principles
SBO	Station Black Out
SCC	Stress Corrosion Cracking
SDD	System Design Document
SFAIRP	So Far As Is Reasonably Practicable
SFP	Spent Fuel Pool
SHE	Standard Hydrogen Electrode
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SNL	Sandia National Laboratories
SoDA	Statement of Design Acceptability
S/P	Suppression Pool
SPCU	Suppression Pool Clean-up
SS	Stainless Steel
SSC	Structure, System or Component
SSE	Safety, Security and Environment
SSM	Strålsäkerhetsmyndigheten (The Swedish Radiation Safety Authority)
TAG	Technical Assessment Guide
T/B	Turbine Building
TCW	Turbine Building Cooling Water System
THAI	Thermal-hydraulics, Aerosols and Iodine
TMI	Three Mile Island
TOC	Total Organic Carbon
TSC	Technical Support Contractor
UK ABWR	United Kingdom Advanced Boiling Water Reactor
UKAEA	United Kingdom Atomic Energy Authority
US NRC	United States (of America) Nuclear Regulatory Commission
WENRA	Western European Nuclear Regulators' Association
WQS	Water Quality Specification
ZNIS	Zinc Injection System

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1 INTRODUCTION

1.1 Background

1. This report presents the assessment conducted as part of the Office for Nuclear Regulation (ONR) Generic Design Assessment (GDA) for the Hitachi-GE UK ABWR reactor design within the topic of Chemistry.
2. Information on the GDA process is provided in a series of documents published on the ONR website (www.onr.org.uk/new-reactors/index.htm). The outcome from the GDA process sought by Requesting Parties such as Hitachi-GE is a Design Acceptance Confirmation (DAC) for ONR and a Statement of Design Acceptability (SoDA) for the Environment Agency (EA) and Natural Resources Wales (NRW).
3. The GDA for UK ABWR followed a step-wise approach in a claims-argument-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 for these earlier Steps are published on ONR's website. The objective of Step 4 is to undertake an in-depth assessment of the safety, security and environmental submissions.
4. The full range of items that might form part of the assessment is provided in ONR's *GDA Guidance to Requesting Parties* (Ref. 1). These include:
 - Consideration of issues identified during the earlier Step 2 and 3 assessments.
 - Judging the design against the Safety Assessment Principles (SAPs) (Ref. 2) and whether the proposed design reduces risks So Far as is Reasonably Practicable (SFAIRP), or As Low As Reasonably Practicable (ALARP)*.
 - Reviewing details of the Hitachi-GE design controls, procurement and quality control arrangements to secure compliance with the design intent.
 - Establishing whether the system performance, safety classification, and reliability requirements are substantiated by the detailed engineering design.
 - Assessing arrangements for ensuring and assuring that safety claims and assumptions are realised in the final as-built design.
 - Resolution of identified nuclear safety and security issues, or identifying paths for resolution.
5. The purpose of this report is therefore to provide the assessment in the chemistry topic that underpins the judgement made in the potential issuing of a DAC, or otherwise. This assessment is focused on the submissions made by Hitachi-GE throughout GDA, in particular in response to those provided to address my Regulatory Queries (RQs), Observations (ROs) and Regulatory Issue (RI) I raised. The RI and ROs issued to Hitachi-GE as part of my assessment are published on ONR's website, together with the corresponding Hitachi-GE resolution plans.

1.2 Scope

6. The scope of my assessment is detailed in my assessment plan (Ref. 3). The plan contains full details of the objectives, scope and approach for my assessment. Consistent with this plan, my assessment is focussed on considering whether the generic safety case provides an adequate justification for the generic design of UK ABWR, in line with the objectives for GDA.

* The terms SFAIRP and ALARP are used interchangeably throughout the report. The outcome of meeting the legal duty (when and where this applies) of reducing relevant risks SFAIRP is that they will have been reduced to ALARP.

7. The scope of my assessment focused on two main themes, relating to the operating chemistry and chemistry effects during accidents. I have therefore considered all of the main plant systems in UK ABWR where there is a need to control the operating chemistry in order to ensure plant safety, or where chemical behaviour is important to consider as part of fault conditions. Both of these are considered on a system level in order to aid understanding.
8. I have considered all of the main submissions within the remit of my assessment scope, to various degrees of breadth and depth. I chose to focus my assessment on those aspects that I judged to have the greatest safety significance, or where the hazards appeared least well controlled. My assessment has also been influenced by the claims made on chemistry related systems, my previous experience of similar systems for reactors and other nuclear facilities and the identified gaps in the original submissions made by Hitachi-GE in a number of areas. A particular focus of my assessment has therefore been the RQs, ROs and RI I raised as a result of my on-going assessment, and the resolution thereof.
9. Further details of the scope of assessment can be found in Section 4.1 of my report. The scope of my assessment is appropriate for GDA given the influence both the operating chemistry and chemical behaviour has on the potential likelihood or consequences of faults or accidents. This is fully aligned with the guidance described in Section 2, including the ONR SAPs (Ref. 2) and Technical Assessment Guides (TAGs) (Ref. 4). This scope is also similar to that undertaken for previous GDA assessments, taking account of the specific design and safety case differences for UK ABWR.

1.3 Method

10. The methodology for the assessment follows HOW2 Guidance on Mechanics of Assessment within the Office for Nuclear Regulation (Ref. 5).

2 ASSESSMENT STRATEGY

2.1 Standards and criteria

11. The standards and criteria adopted within this assessment are principally the SAPs (Ref. 2), TAGs (Ref. 4), relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites.

2.1.1 Safety Assessment Principles

12. The key SAPs applied within the assessment are included within Annex 1.
13. As the SAPs (Ref. 2) constitute the regulatory principles against which dutyholders' safety cases are judged, they are therefore the basis for ONR's nuclear safety assessment. From a chemistry perspective, the most important SAPs are ECH.1 to 4, but I have also relied heavily on those relating to the safety case (SC.2 to 5), engineering key principles (EKP.1 to 5), integrity of metal components and structures (EMC.2, 3, 13, 21, 22 and 24) and faults analysis (the FA and AV series).

2.1.2 Technical Assessment Guides

14. The TAGs used as part of this assessment are set out in Annex 2. The most important TAG for my assessment is the information contained in NS-TAST-GD-088, *Chemistry of Operating Civil Nuclear Reactors* (Ref. 4).
15. In addition to the TAGs which align to the SAPs described above, a key consideration has been the expectations of NS-TAST-GD-051, *The Purpose, Scope and Content of Nuclear Safety Cases* (Ref. 4), which sets out some key expectations for safety cases against which I have compared Hitachi-GE's submissions. NS-TAST-GD-005, *ONR Guidance on the Demonstration of ALARP* (Ref. 4) has also been an important input to my assessment.

2.1.3 National and international standards and guidance

16. The international standards and guidance that have been used as part of this assessment are set out in Annex 3.
17. There are both International Atomic Energy Agency (IAEA) standards (Ref. 6) and Western European Nuclear Regulators Association (WENRA) Reference Levels (Ref. 7) of relevance. However, they are often not specific to chemistry and therefore the SAPs will be the foremost standard considered. It should be noted that the latest version of the SAPs (Ref. 2) has been benchmarked against both IAEA and WENRA guidance.

2.2 Use of Technical Support Contractors

18. It is usual in GDA for ONR to use Technical Support Contractors (TSCs) to provide access to independent advice and experience, analysis techniques and models, and to enable ONR's inspectors to focus on regulatory decision making.
19. Table 1 provides the areas where I utilised TSC support during my chemistry assessment, along with a description of the work I asked them to carry out and the relevant section of my report where I have drawn on the work in support of my conclusions. My general strategy for deciding when and whether to engage a TSC during my assessment was if one or more of the following criteria applied:
 - As commercial Boiling Water Reactor (BWR) technology is new to the UK, where I needed specific knowledge, skills and experience in BWR technology to support my regulatory decision making.

- If the topic under consideration was particularly contentious and access to a wider source of specialist knowledge and independent advice was required to facilitate effective regulatory decision making.
- To provide independent confirmatory analysis using computer models, codes and/or techniques which ONR does not have access to.

Topic Number	Description of Work	Assessment Report Section
1	Training course on BWR chemistry and related matters and membership of the Light Water Reactor Chemistry & Component (LCC) Integrity Programme	All sections
2	Working Group on BWR Chemistry – Part 1	4.3.1.1
3	Working Group on BWR Chemistry – Part 2	4.3.1.2 4.3.2
4	Initial review of UK ABWR materials degradation threats and materials selection.	4.3.1.1
5	Support on iodine chemistry.	4.7.4
6	Independent modelling of radioactivity generation and transport in UK ABWR and independent radiolysis and Electrochemical Corrosion Potential (ECP).	4.3.1.1 4.3.1.2 4.3.1.3
7	Review of the impact of the UK ABWR operating chemistry on fuel reliability and integrity.	4.3.4
8	Review of relevant good practice in BWR commissioning, start-up and shutdown and chemistry and independent review of relevant Hitachi-GE submissions.	4.3.1.2 4.3.1.3 4.3.1.4
9	Review of relevant good practice for BWR sampling systems and independent review of relevant Hitachi-GE submissions.	4.3.5.3 4.7.6
10	Independent review of selected Hitachi-GE submissions on materials degradation threats.	4.3.2

Table 1: List of chemistry TSCs

2.3 Integration with other assessment topics

20. 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. Given the scope and nature of my assessment, the following cross-cutting matters have been of particular relevance:

- Chemistry worked closely with structural integrity to assess the justifications for the materials selected for UK ABWR. Chemistry provides input to the integrity and corrosion aspects of the assessment; with the effects of the operating chemistry (environment) on the susceptibility to material degradation mechanisms being led by chemistry. However, the overall judgement on the adequacy of the safety case for material degradation aspects will also need to be informed by material and stress factors, which are led by the structural integrity inspector.
- The generation, accumulation, management and mitigation of radiolysis gases during normal operations is a cross-cutting topic. Chemistry led on aspects relating to the mitigation of radiolysis gas hazards within the plant processes,

working closely alongside internal hazards regarding the overall adequacy of the case for UK ABWR.

- Chemistry led on the assessment of the quantity and minimisation of radioactivity during normal operations. Such matters are broad, requiring coordination between several disciplines including radiological protection, radioactive waste management, structural integrity (to a lesser extent) and interactions with the EA's areas of assessment. This specifically included aspects such as definition of the source terms, minimisation of high cobalt alloys and the impact of the operating chemistry.
- Chemistry provided an input to the cladding corrosion and fuel deposits aspects of the fuel design assessment. The effects of the operating chemistry on these aspects were led by myself, as were the assessment of any chemistry related consequences (for example on radioactivity or deposition), but any non-chemistry related consequences are led by the fuel design inspector.
- Chemistry provides input into the fault studies and severe accidents areas, where chemistry effects are important in determining the consequences or effectiveness of mitigation measures and in the analysis of combustible gas and radiological accident source terms. This area is led by fault studies, with input from reactor chemistry. There were also interactions with Probabilistic Safety Analysis (PSA). Chemistry specifically led on aspects relating to pH control within the suppression pool.
- The chemistry control of gaseous and liquid radioactive waste treatment systems formed part of my assessment, which required close working with the Environment Agency and radioactive waste and decommissioning areas.

2.4 Overseas regulatory interface

21. 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 assessments, which can expedite assessment and helps promote consistency.
22. 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
 - Implication of MDEP products in order to facilitate the licensing of new reactors, including those being developed by Gen IV international Forum
23. In addition to these, I specifically note the following interactions with overseas regulators which have provided an input to my assessment:
 - Information exchange meeting with the Swedish Nuclear Safety Regulator (SSM) and Swedish utilities on BWR chemistry and design, November 2014 (Ref. 8).
 - Meeting with the Japanese Nuclear Regulation Authority (NRA) to exchange information in support of the chemistry assessment, January 2015 (Ref. 9).

2.5 Out of scope items

24. As stated in Section 1, the scope of my assessment is broad, including the operating chemistry within most of the UK ABWR plant systems. However, matters which are clearly site specific have not been included in my assessment, such as the water make-up system or fuel oil systems.

25. A key objective of my assessment has been to judge the adequacy of Hitachi-GE's approach to deriving and justifying the chemistry-based Operating Rules (ORs), and their associated limits and conditions contained within the generic safety case. During GDA I have assessed the adequacy of the bounding limits proposed and the completeness of the list of parameters. I have not assessed any details regarding how a future licensee may incorporate these into their hierarchy of Operating Rules and/or operating instructions. Similarly, I have not assessed any administrative limits (such as "Action Levels") proposed by Hitachi-GE during GDA, nor how a future licensee may choose to implement their chemistry control programme. Similarly, a detailed assessment of any proposed Technical Specifications, operating procedures and emergency arrangements, has not been performed.

3 REQUESTING PARTY'S SAFETY CASE

26. In this section I describe the overall approach and structure of the UK ABWR generic safety case documents provided by Hitachi-GE during GDA. The purpose of this section is to introduce some of the common themes which feature in Hitachi-GE's documentation. This section does not contain any formal assessment. Details of the technical content of the generic safety case documentation and my assessment of its adequacy; are reported in the subsequent sections of my report.

3.1 Claims, Arguments and Evidence Approach

27. The fundamental structure in the UK ABWR generic safety case is to adopt a Claims, Arguments and Evidence (CAE) approach. This approach is commonly used in structuring safety cases in the UK nuclear industry and elsewhere. The intent behind this approach is to provide a clear structure, and make complex safety justifications more understandable. In this context, Hitachi-GE defines these terms as:

- claim: High level proposition, assertion or statement;
- argument: The reason why the claim is justified. Supports the claim and points to where the supporting evidence can be found; and
- evidence: Facts and judgements that support the argument.

28. Hitachi-GE's approach to CAE is defined in their *Safety Case Development Manual* (SCDM) (Ref. 10). In summary, CAE is applied to all aspects of the generic safety case, but much more rigorously and formally in technical topics which are largely systems or engineering based (such as Control & Instrumentation (C&I)). For chemistry, the application of CAE is less rigorous. The approach is more narrative based. However, Hitachi-GE still provides CAE for the operating chemistry regimes defined for the plant; this directly influences the content of the supporting reports.

29. A number of high-level safety claims are made throughout the generic safety case, and are uniquely identified and traced throughout. These are assessed in the relevant sections of my assessment that follows, but those relating to the operating chemistry are summarised in Annex 4. Hitachi-GE's approach is to provide a "*top claim*", supplemented by a number of more detailed "*claims*", for each of the main systems considered in the PCSR. Note that Annex 4 is not a complete list, but does represent the main chemistry specific claims made by Hitachi-GE in Chapter 23 of the PCSR (Ref. 11).

3.2 Documentation Structure

30. The GDA documentation is structured in levels to clearly present the overall safety case in a consistent and coherent manner. The documentation is listed within the *Master Document Submission List* (MDSL) (Ref. 12). The overall documentation structure is divided into three levels as shown in Figure 1.

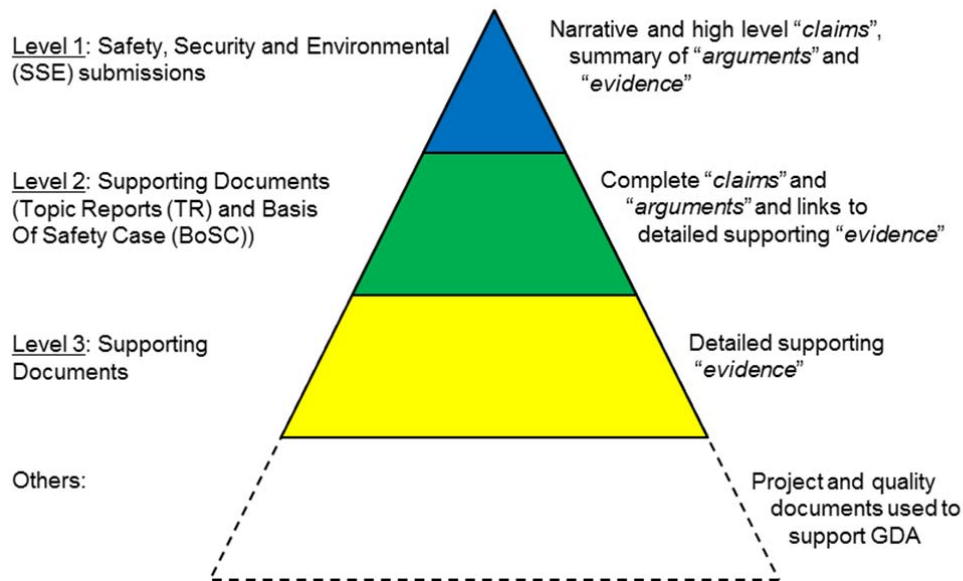


Figure 1: Hierarchy of UK ABWR generic safety case documentation

31. In summary:

- Level 1 information forms the top-tier documentation and consists of the Safety, Security and Environmental (SSE) submissions. These include the generic PCSR, the Generic Environmental Permit (GEP) documentation and the conceptual security arrangements. In the PCSR, safety claims are illustrated and developed in a way which aims to demonstrate, at a high-level, that the UK ABWR design meets UK safety requirements, and that relevant risks are reduced SFAIRP.
- Level 2 documentation is used to support and substantiate the claims in the SSE submissions and provide the link between the SSE submissions and detailed evidence found in Level 3 documentation. The documents comprise a large number of Topic Reports (TR), which consider specific topical assessment areas, and Basis of Safety Case (BoSC) reports, which cover specific Systems, Structures and Components (SSCs). TRs and BoSCs were "living documents" that were developed or revised throughout the course of GDA. Collectively, they aim to capture the key aspects claimed on the corresponding SSCs. Additionally, the BoSCs and TRs also provide information on the assumptions (including operating limits and conditions) that will be used to move the safety case to an operating regime.
- Level 3 documentation includes detailed design, evaluation and analysis documentation that are used to provide evidence to support and substantiate the arguments in Level 2 and demonstrate that the design meets the claims in Level 1.

32. The key documentation which form the basis of the chemistry aspects of the UK ABWR generic safety case are described further below. They are too numerous to list in detail here and are referred to in subsequent sections of my assessment. A summary of the Level 1 and 2 documents relating to the operating chemistry is also shown schematically in Figure 2, as an example to illustrate the typical approach and breadth of coverage.

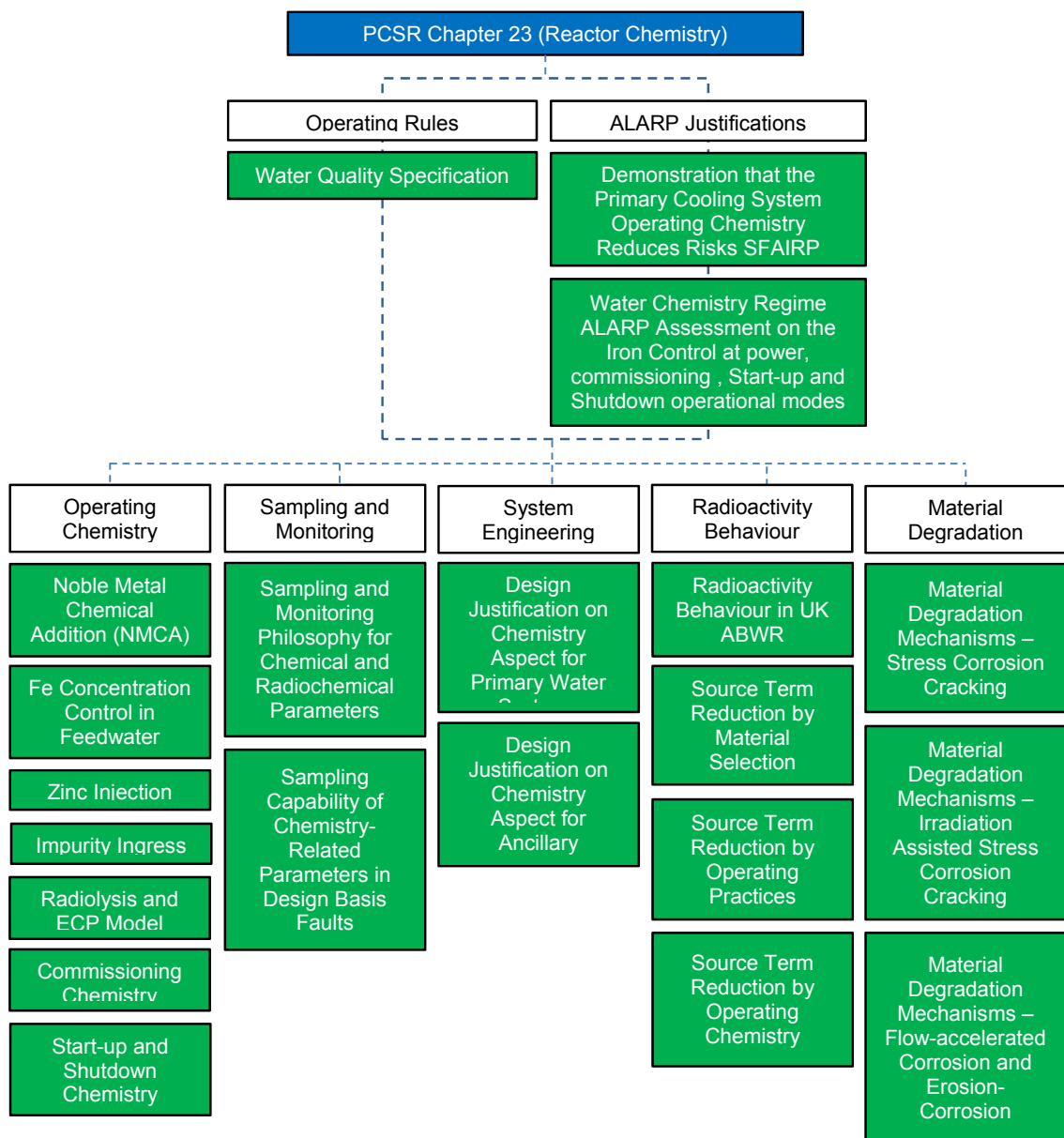


Figure 2: Key level 1 and 2 safety case documentation relating to the UK ABWR operating chemistry choice

3.2.1 Pre-Construction Safety Report

33. The top level document in the UK ABWR generic safety case is the PCSR (Ref. 11). The overall PCSR structure is based on relevant guidance (for example from IAEA or ONR), combined with Hitachi-GE's own views about how to apply certain safety case methodologies and approaches. It consists of 32 separate chapters, which are organised into five parts. These are:

- Part-I sets the scene for the generic PCSR, and describes some of the generic issues that impact on the safety of the plant, including generic site characteristics and internal and external hazards.
- Part-II describes the main technical systems of the UK ABWR and summarises the main functional requirements and technical specifications of SSCs required to deliver safety functions.
- Part-III defines the auxiliary and support systems that are necessary for performance of the SSCs. It also outlines processes for management of solid,

- liquid and gaseous radioactive materials, to protect and reduce the risks of radioactive exposure to operators, workers and the general public. “Reactor chemistry” is one of the chapters in this part of the PCSR.
- Part-IV presents a summary of the safety analysis and assessment work that has been used to confirm and inform the design of SSCs.
 - Part-V describes how safety will be achieved and maintained throughout the plant’s lifetime, through commissioning, operation and decommissioning.
34. Chapter 23 of the PCSR (Ref. 11) covers the “*reactor chemistry*” aspects of the generic safety case. Despite the title, this chapter is much broader than covering just the chemistry of the reactor and considers a wide range of plant systems where chemistry controls are necessary for safety. The chapter describes the UK ABWR chemistry regimes and provides the justification for why Hitachi-GE considers it to be aligned with relevant good practice and optimised when considering all relevant factors, for example: SSC degradation mechanisms, radiological dose implications, and waste management requirements. Chapter 23 also concludes why the chemistry aspects of the UK ABWR design mean relevant risks are reduced SFAIRP. This chapter uses a narrative led approach, supported throughout by referencing relevant Level 2 documents.
35. A number of other PCSR chapters are also relevant to the scope of my chemistry assessment. In addition to those engineering based chapters (Chapters 12 to 17), this specifically includes:
- Chapter 7 – internal hazards (Ref. 13);
 - Chapter 11 – reactor core (Ref. 14);
 - Chapter 20 – radiological protection (Ref. 15); and
 - Chapters 24 and 26 – design basis analysis, and beyond design basis and severe accident analysis (Ref. 16 and 17).

3.2.2 Design Documents

36. As noted above, formal application of the CAE approach is generally restricted to the engineering based topics. For those the main supporting references (Level 2) are BoSC documents. They therefore contain a more formal CAE approach, and are produced to demonstrate how the claims are linked to the design of the SSCs. They do not themselves contain detailed evidence, but signpost where this is found in other lower level documents. They typically consider matters such as SSC requirements, design description and engineering justification and include consideration of relevant codes and standards. In addition a range of design documents, such as System Description Documents (SDD) together with engineering drawings provide more information on the detailed engineering and design.
37. From a chemistry perspective, the BoSCs and supporting design documents are relevant to gain an understanding the plant engineering. However, it was recognised by Hitachi-GE during GDA that these did not necessarily consider chemistry-related matters, such as the justification for the chemical performance of the SSCs. To address this gap two TRs, *Topic Report on Design Justification in Chemistry Aspect for Primary Water Systems* (Ref. 18) and *Topic Report on Design Justification in Chemistry Aspect for Ancillary Systems* (Ref. 19), were produced. While not BoSCs, these documents contain a large proportion of the design information for my assessment.

3.2.3 Topic Reports

38. The other main type of Level 2 documents are TRs. The TRs do not use the formal CAE approach, but they do often reiterate key claims/arguments and provide the necessary links in the document to which specific claim/argument is being supported.

As a result, the scope and approach used in TRs is much wider than BoSCs. TRs often contain supporting evidence, as well as linking out to other more detailed Level 3 reports.

39. The vast majority of generic safety case information relevant to chemistry is found within TRs. These consider aspects such as ALARP justifications, design justifications, source term, water quality, specific chemistry regime aspects, and materials degradation and selection aspects. The system functionality is covered within other chapters of the PCSR and in turn by systems-based BoSCs. The chemistry aspects covered in the supporting TRs ensure the on-going maintenance of the design intent for the UK ABWR systems. Additionally, they specify the requirements that are to be delivered by those systems for effective chemistry monitoring and control.
40. A key document in the generic safety case underpinning the operating chemistry regime is the *Water Quality Specification* (Ref. 20). While not strictly a TR, it is analogous to one and is the most important Level 2 document relating to the operating chemistry in UK ABWR. This report contains the basis and tabulated values for the various controls that Hitachi-GE specifies in relation to the operating chemistry within a number of important systems. These represent the main chemistry-related ORs and this report is therefore a key element of my overall assessment. A summary of the most important aspects of the ORs presented therein is given in Annex 5 of my report. The approach taken by Hitachi-GE is to specify both a “*limit value*” and “*expected value*”, in addition to classifying each parameter as either a “*control*” or “*diagnostic*” parameter. This approach and the values themselves are assessed further as part of my assessment.

3.2.4 Other Documents

41. A number of other supporting documents were submitted and formed an input into my assessment. These included the SCDM (Ref. 10) and Hitachi-GE’s *GDA ALARP Methodology* (Ref. 21). Documentation at this level tends to be more related to processes and approaches that are applied in wider aspects of the generic safety case.

3.2.5 Responses to Regulatory Queries, Observations and Issues

42. As a consequence of my assessment of the generic safety case for UK ABWR, and in line with ONR guidance, I raised 186 RQs, 11 ROs and one RI. The ROs and RI are detailed in Annex 6 of my report. I have considered the responses to these as part of my assessment, which generally consisted of a plethora of TRs. The responses to all my RQs were included within other documentation at the end of GDA, but my assessment does make reference to individual RQs where necessary to exemplify my considerations.

4 ONR ASSESSMENT

43. This assessment has been carried out in accordance with HOW2 guide NS-PER-GD-014, *Purpose and Scope of Permissioning* (Ref. 22).

4.1 Scope of Assessment Undertaken

44. The scope covered by this assessment report is detailed in my assessment plan (Ref. 3). The principal objective is to conduct an in-depth assessment of the UK ABWR generic design and safety case presented by Hitachi-GE, for those aspects related to chemistry. The focus is on assessing, on a sampling basis, whether there are any fundamental shortfalls with the generic design which might preclude it from being safely constructed, operated, or decommissioned, and therefore threaten the issue of a DAC.

45. For the purpose of this assessment chemistry is taken to be: “*the chemistry of the design including the effects of coolant chemistry on reactivity, pressure boundary integrity, fuel and core component integrity, fuel storage in cooling pools, radioactive waste generation and radiological doses to workers*” (Ref. 3). Thus, it is principally concerned with five main areas, the relative influence of which can vary depending upon the system under consideration:

- control of coolant reactivity;
- protection of the structural materials (specifically related to integrity of the pressure boundaries);
- maintaining fuel integrity and performance;
- minimisation of out of core radiation fields, and;
- minimisation of releases during accident conditions.

46. Producing a justification that the influence of any of these effects on the relevant risks and hazards are minimised (and for relevant risks, reduced SFAIRP), can take many forms including: calculations, assessments, studies, reviews of experience or arguments from first-principles. My assessment therefore considers both the technical basis for the relevant chemistry effects, but also the engineering of the systems which deliver those safely as part of the design. This therefore covers the underlying basis for:

- any requirement or constraint placed on the operating chemistry of the plant which must be met in order to allow the plant to be operated safely;
- any chemistry related functional requirement which must be met to ensure that the plant is operated within its design basis; and
- any effect or consequence of chemistry during operations, during faults or during severe accidents, which must be understood and controlled in order to ensure the safety of workers and the public.

47. My assessment report is structured around two main themes: the operating chemistry and chemistry effects during accidents. These are considered on a system level to aid understanding. It is also important to note that assessment of the potential for UK ABWR to load follow was beyond the scope of GDA.

4.2 Overview of the UK ABWR Design

48. Before providing my assessment of the chemistry aspects of the generic safety case for the UK ABWR design, some background information on BWR technology, the ABWR design and BWR operating chemistry is given in the following paragraphs. This is a general description only, but provides some important context to my assessment that follows.

BWR Technology

49. UK ABWR, like all BWRs, is an evolution of the original BWR technology developed as part of the commercialisation of nuclear power in the 1960s. There are currently 78 operational BWRs worldwide, with a further four under construction; of these seven are of the ABWR design (four operating, three under construction) (Ref. 23).
50. All BWRs use water as both coolant and neutron moderator. Heat released by the fission process is used to boil the water inside the Reactor Pressure Vessel (RPV) at a slightly elevated pressure (around 7.0 MPa). This water enters the core in a slightly sub-cooled state, is heated to boiling temperature, and partially evaporates. The steam-water mixture produced passes through steam separators located above the core where most of the water is removed. The water is returned to the bottom of the RPV and the steam is routed upward to the steam dryers. The dry main steam produced leaves the RPV through four main steam nozzles and flows directly to the turbine.
51. Once used to power the turbine, the steam is condensed, and this condensate is cleaned, de-aerated, and heated before being returned as feedwater to the core via feedwater nozzles on the RPV located above the core. The feedwater is routed downward past the core region through the downcomer annulus between the RPV wall and the core shroud; it flows along the bottom of the RPV toward the centre and then upward again in the direction of the core. In order to achieve the required power density forced circulation of the coolant inside the RPV is necessary. In the UK ABWR design several Reactor Internal Pumps (RIPs) are installed directly inside the annulus to provide internal forced circulation of the reactor water. The UK ABWR RPV and Reactor Internals (RINs) are shown in Figure 3.

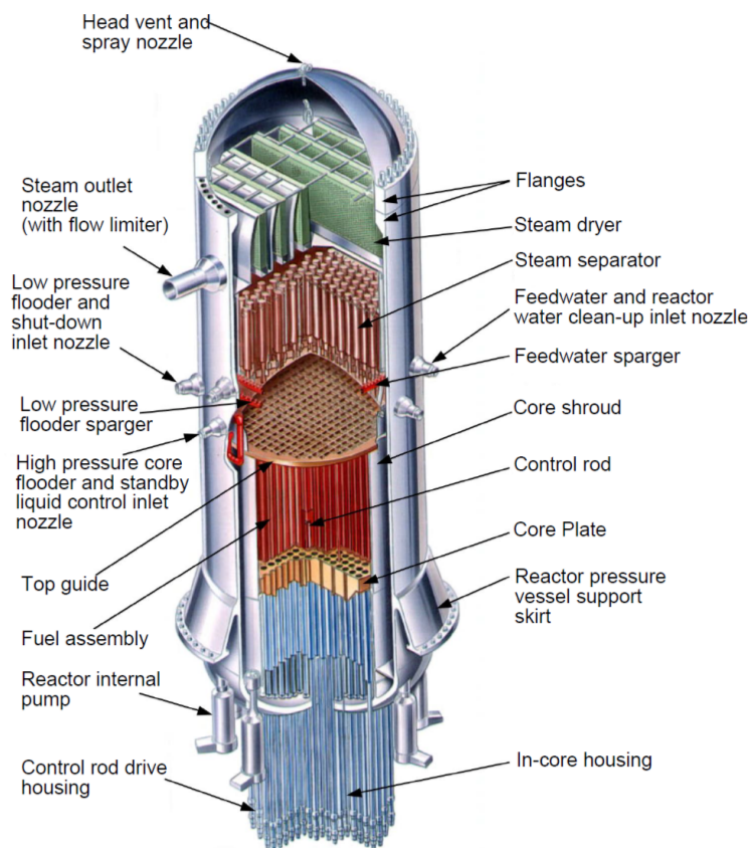
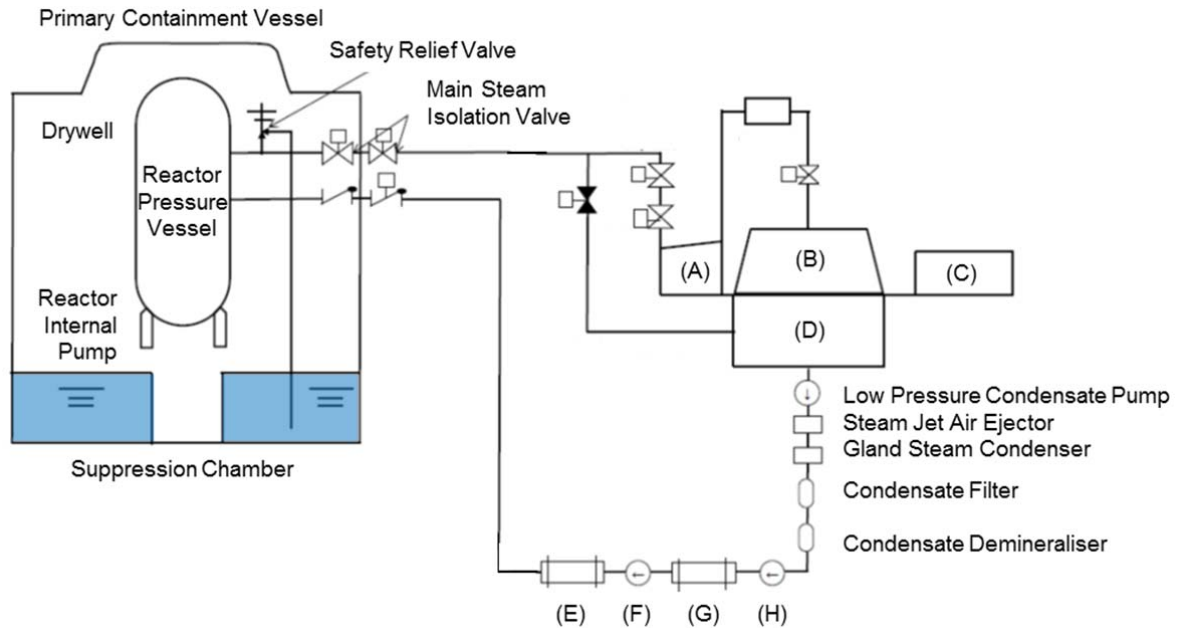


Figure 3: Schematic of the UK ABWR RPV and RINs

52. A schematic of the Reactor Coolant System (RCS) in UK ABWR is shown in Figure 4. The RCS consists of those systems which transfer the coolant to and from the reactor, via the turbine and condenser. The RCS is in fact not a defined single plant 'system', but consists of elements of: the Condensate and Feedwater (CFDW) system, RPV, RINs, nuclear fuel, Main Steam (MS) line, turbine and associated drains and condenser. This represents the main coolant loop during normal operations.



Key: (A) High Pressure Turbine; (B) Low Pressure Turbine; (C) Generator; (D) Condenser; (E) High Pressure Feedwater Heater; (F) Feedwater Pump; (G) Low Pressure Feedwater Heater; (H) High Pressure Condensate Pump.

Figure 4: Schematic of the UK ABWR RCS

53. The coolant within the RCS therefore contacts many surfaces within this circuit, including: the nuclear fuel, RPV, steam, feed, and other pipework. These surfaces comprise the main pressure boundary as well as both the primary and secondary containment boundaries for any radioactivity generated within the plant. Owing to the direct cycle nature of the plant, the pressure and temperature conditions also vary greatly around the RCS, as do material choices for the various SSCs. The operating chemistry for the RCS therefore needs to balance these often competing/differing requirements, whilst at the same time deliver the necessary claimed safety functions. The overall importance of the RCS means that a large part of my report focusses on assessing this system.
54. A number of other systems are connected to the RCS, which amongst other things, support: chemical control, sampling and delivery of the necessary safety functions. These systems include::
- the Reactor Water Clean-up (CUW) system;
 - the Condensate Purification System (CPS);
 - the Control Rod Drive (CRD) system;
 - the Residual Heat Removal (RHR) system;
 - the Sampling (SAM) system;
 - the Emergency Core Cooling System (ECCS);
 - the Stand-by Liquid Control (SLC) system; and
 - various chemical injection systems.

55. These systems, plus others of relevance, are considered as part of my assessment that follows. This considers both their ability to control the operating chemistry or the impact of the operating chemistry on their ability to fulfil any necessary safety functions, as appropriate.

ABWR Design

56. The ABWR design incorporates the evolutions in BWR technology since the early reactors of this type. The first ABWRs were Unit 6 and Unit 7 of Kashiwazaki-Kariwa Nuclear Power Station (KK-6 and KK-7) in Japan. KK-6 began operations in 1996 and KK-7 in 1997. Two further ABWRs have operated in Japan, at Shika and Hamaoka. Further ABWRs are under construction in Japan (Shimane and Ohma) and Taiwan (Lungmen). A licence has been granted to construct 2 ABWRs in the US (South Texas Project), but construction is yet to start.
57. Since the original BWR designs entered operation during the 1960's, design developments have occurred to improve plant safety, reliability and operability, and to simplify the design, in addition to taking advantage of improvements in technology. In UK ABWR major areas of such design development include:
- Adding RIPs mounted on the bottom of the RPV - 10 in total - which circulate the reactor coolant. In earlier BWR designs external pumps and loops were necessary to do this. The UK ABWR design has the advantage of eliminating the large recirculation pumps in containment and associated large-diameter and complex piping interfaces with the RPV found in earlier BWR designs. Only the RIP motor is located outside of the RPV in the ABWR design. This design therefore has benefits in reducing the worst potential leak below the core region to around an order of magnitude less than previous BWRs, in addition to reducing doses to workers due to inspection activities on this contaminated pipework. The first reactors to use such internal recirculation pumps were designed by ASEA-Atom and built in Sweden; these plants have operated very successfully for many years.
 - Improvements to the containment design. Like the original BWR designs, it is of the pressure suppression type, designed to remove steam in the event of a transient, incident, or accident by condensation into a pool of water enclosed in the wetwell. This aims to keep containment pressures low. The UK ABWR uses a cylindrical Reinforced Concrete Containment Vessel (RCCV) which is integrated within the reactor building and incorporates a steel liner to prevent leakage.
 - The ABWR design uses electric Fine Motion Control Rod Drive(s) (FMCRD(s)), first used in the German BWRs, instead of a hydraulic locking piston system to move the control rods in increments. The FMCRD allows for fine position adjustment using an electrical motor, while not losing the reliability or redundancy of traditional hydraulic systems which are designed to accomplish rapid shutdown from receipt of an initiating signal.
 - The ECCS of UK ABWR has been improved in many areas, to provide a high level of defence-in-depth against accidents. In particular, it is designed to ensure the core remains covered by coolant during a postulated Loss of Coolant Accident (LOCA). The overall system has been divided up into three divisions. Each division is capable, by itself, of terminating the accident prior to the core becoming uncovered even in the unlikely event of loss of offsite power and loss of feedwater. Previous BWRs had two divisions, and the core being uncovered (but no core damage) was predicted to occur for a short time, prior to the response of the ECCS.
58. Many other design developments have been incorporated into the design at a system or component level. These features are assessed as part of my assessment in subsequent sections of this report, where relevant.

BWR Operating Chemistry

59. As described previously, the main systems in a BWR comprise the RCS and connected auxiliary systems. The operating chemistry applied within this system has a significant impact on the nature, scale and scope of potential hazards and safety detriments that must be managed and minimised during operation. Therefore, before assessing the claims made by Hitachi-GE on the chemistry controls applied within the RCS, some background information on chemistry effects in the system during normal operations is described below, to give context to the assessment that follows. This description illustrates how some of the main hazards in this system arise, namely radioactivity, gaseous radiolysis products and conditions which could degrade fuel or structural materials.
- As described previously an important factor is that the coolant is allowed to boil. This means that the coolant must necessarily be of a high purity to stop aggressive species accumulating on the fuel or in the reactor water. Preventing this accumulation of impurities is primarily why a CUW system is needed.
 - Steam that is produced in the RPV is transferred to the turbine, condensed, purified, re-heated and then returned to the RPV as feedwater. This also means that any volatile impurities within the coolant can be carried within the steam to the turbine systems. Conditions are such that all of the steam path surfaces are expected to have a water film.
 - Due to the volume of the RPV and steam flow rate, the residence time of a given water molecule within the RPV is large. This means that as water flows through the core and is exposed to ionizing radiation (especially neutrons) a wide variety of radiolysis products are produced. For simplicity, the main species can be considered to be hydrogen and hydrogen peroxide. During boiling hydrogen partitions to the steam phase so the reactor water is oxidant rich. Hydrogen peroxide decomposes to water and oxygen, and the relative proportions of these vary with time and distance away from the core. Expressed as oxygen, pure reactor water in a typical BWR would have several hundred parts per billion (ppb) oxygen and only tens of ppb of hydrogen.
 - Essentially, all of the radiolysis products eventually are transferred with the steam to the turbine. They therefore need to be recombined in the Off-gas (OG) system, which draws the condenser vacuum (after the turbine), to prevent a flammable atmosphere from developing.
 - Also in the core, some ^{16}O (in the water coolant) is activated to ^{16}N , which has a half-life of only seven seconds, but is a very high energy gamma emitter. During operations ^{16}N is therefore the dominant radionuclide for dose. In pure water the most stable form of nitrogen in a BWR is nitrate, which remains soluble and stays in the reactor water.
 - Similarly other non-radioactive precursor species exposed to the core radiation field can become activated, particularly metallic corrosion products, leading to the production of radionuclides such as: ^{60}Co , ^{58}Co , ^{54}Mn and ^{51}Cr . These activated corrosion products deposit on system pipework and dominate dose rates during shutdown conditions.
60. Historically, the operating chemistry applied to BWRs was based upon the premise of maintaining the water coolant as pure as possible. This is the basis of so-called Normal Water Chemistry (NWC). All currently operating BWRs began their life operating under early versions of this regime. To a large extent this proved viable, however, the full impact of the resultant chemical environment on the integrity of the plant components and radioactivity in the longer-term was not fully appreciated. This led to safety issues later in life, in particular related to Stress Corrosion Cracking (SCC) of Stainless Steel (SS) components.
61. Many plants worldwide now add dissolved hydrogen to the coolant to mitigate the oxidising environment produced. This is known as Hydrogen Water Chemistry (HWC).

HWC also carries some detriments, and other additives can also be used to offset these, including Noble Metal Chemical Addition (NMCA), or the on-line variant, known as On-Line NobleChem™ (OLNC). The addition of noble metals to BWRs minimises the amount of hydrogen that needs to be added to counter the oxidising nature of the coolant. NMCA refers to the original method that was developed to enable noble metals to be added to BWRs, it is therefore also referred to as Classic NobleChem™ (CNC). This is conducted during reactor shutdown. OLNC, on the other hand, is conducted during at-power operations. It involves injecting a solution of noble metal (typically as $\text{Na}_2\text{Pt}(\text{OH})_6$) into the feedwater, where it is transported to the RPV and deposits as nano-sized particulates on surfaces. The other chemical species also purposefully to many BWRs is Depleted Zinc Oxide (DZO). Adding DZO minimises the deposition of activated corrosion products on plant pipework. In this context, DZO refers to zinc which is depleted in the ^{64}Zn isotope, which minimises the formation of ^{65}Zn . The evolution in operating chemistry choices for US BWRs is exemplified in Figure 5 (Ref. 24).

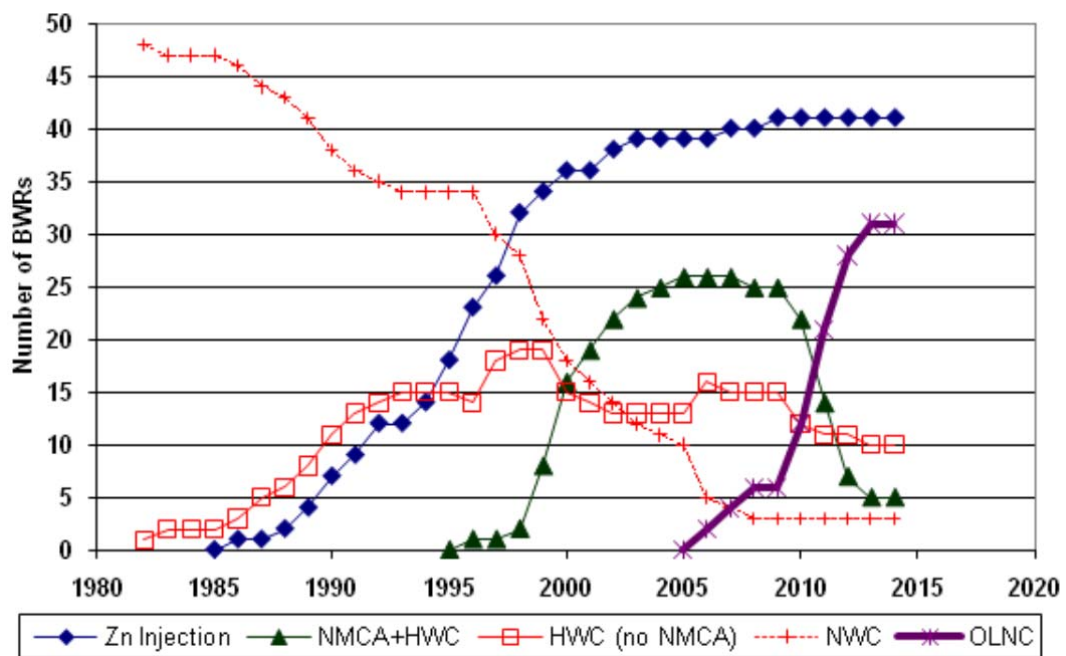


Figure 5: Evolution of BWR Chemistry Regimes in the US

62. While the majority of US plants have moved towards HWC and other purposeful chemical additions, the same clear trend is not true in European and Japanese BWRs many of which continue to operate under NWC conditions. In particular no ABWR has operated using HWC, aside from short duration tests.
63. Overall, the chemistry of operational BWRs is a complex interaction between many variables. Some of these are well understood, whereas others can only be empirically followed based on plant experience. Importantly, there is no single unified understanding of all aspects of BWR chemistry; in particular relating to how any changes to operating conditions may impact on the relevant risks which chemistry can provide mitigation for. There is no industry-wide consensus on the operating chemistry to be applied for BWRs; it is taken as being design specific. The decisions taken by Hitachi-GE for the UK ABWR design regarding the operating chemistry for the RCS are therefore a key part of my assessment that follows.

4.3 Reactor Coolant System

64. This section of my assessment report considers the RCS. As noted previously, the RCS covers the main coolant flow path for UK ABWR, including the: reactor, steam and condensate systems, as shown in Figure 4. In addition, it includes a number of supporting systems of particular relevance to chemistry: the RHR, CUW and SAM systems.
65. It is important to note here that the overall structure of my assessment is presented according to the principal hazards and risks which the operating chemistry (and the accident chemistry topic) has some influence over. Using the broad theme of the operating chemistry as an example, my assessment is presented against issues such as: materials degradation, radioactivity, fuel integrity *etc.* This means the assessment is not presented in a topic-wise fashion. For example, there is no discrete section covering all aspects of the start-up and shutdown chemistry topic, or commissioning chemistry *etc.* These, and many other topics, are split into the main themes covered by the report. The main body of the assessment therefore contains numerous links to other relevant sections to aid the reader's understanding.

4.3.1 Operating Chemistry

66. Chemistry control in the RCS can be particularly important given the potential risks posed by its inadequate execution on the integrity of fuel and structural materials, and the generation and transport of radioactivity, amongst others. The operating chemistry decision for the RCS is therefore one example of an area where Hitachi-GE can directly influence and control the magnitude of the radiological hazards and the resultant risks posed by the UK ABWR design. As previously explained, there is no industry-wide consensus on the RCS operating chemistry to be applied to BWRs; therefore the overall decision is taken as being design specific. This means the selection of an appropriate RCS operating chemistry regime for UK ABWR requires a detailed holistic assessment of all of the relevant risks and a demonstration that on balance, they are reduced SFAIRP.
67. Depending on which one of the RCS operating chemistry options, or combination of options, is selected, leads to different outcomes for the level of residual risk for the design. Furthermore, at the design stage, it is very important that the RCS operating chemistry decision is taken in tandem with materials selection choices, because each one influences the other. Also, the RCS operating chemistry decision may directly influence the engineered design and systems required for the plant, for example, the injection and/or monitoring systems.
68. This first section of my assessment of the RCS therefore considers:
- The adequacy of the process undertaken by Hitachi-GE to decide on the RCS operating chemistry choice(s) for the UK ABWR.
 - The justification that an overall ALARP balance has been reached.
 - The integration of this choice within the generic safety case documentation.
69. The influence this choice has on the resultant risks and hazards, namely: materials degradation, generation of radioactivity or its subsequent behaviour, and fuel integrity, is considered in more detail in Sections 4.3.2, 4.3.3 and 4.3.4, respectively, which follow.
70. In arriving at the RCS operating chemistry choice, it is also very important not only to consider the chemistry implemented during at-power operations, but also how implementing that chemistry is achieved when the plant is commissioned and the nature and extent of chemistry control required during start-up and shutdown operations. The scope of this section of my report therefore also considers Hitachi-GE's approach to justifying the chemistry choices for commissioning, and start-up and shutdown operations.

4.3.1.1 At-power Operating Chemistry Choice

71. During GDA, it became apparent Hitachi-GE's approach to justifying the RCS operating chemistry for UK ABWR may not meet regulatory expectations. Early in Step 3 I therefore raised RO-ABWR-0022 (Ref. 25). The scope of RO-ABWR-0022 (Ref. 25) only covered the justification required for the at-power operating chemistry proposed for UK ABWR. The focus was not on the operating chemistry choice itself, rather, the importance of showing a process of optimisation had been followed to arrive at a robust justification; and that this justification is part of the generic safety case for UK ABWR.
72. The main SAPs (Ref. 2) of relevance to this part of my assessment are therefore ECH.1, ECH.2 and ECH.3 and the guidance provided in NS-TAST-GD-088, *Chemistry of Operating Civil Nuclear Reactors* (Ref. 4). I have also used the information contained in NS-TAST-GD-005, *Guidance on the Demonstration of ALARP* (Ref. 4).
73. In response to RO-ABWR-0022 (Ref. 25) Hitachi-GE produced the report, *Demonstration that the Primary Cooling System Operating Chemistry Reduced Risks SFAIRP* (Ref. 26). Although not produced in direct response to this RO, *Topic Report on the Water Chemistry Regime ALARP Assessments on the Iron Control at Power, Commissioning, Start-up and Shutdown Operational Modes: Part 1 – Main Justification* (Ref. 27) and *Topic Report on the Water Chemistry Regime ALARP Assessments on the Iron Control at Power, Commissioning, Start-up and Shutdown Operational Modes: Part 2 – ALARP Workshop Method and Record* (Ref. 28) are also of direct relevance. These are also underpinned by a number of other TRs which contain much more detailed safety case evidence in support of Hitachi-GE's overall justification for the RCS operating chemistry choice for UK ABWR. These documents include:
- *Topic Report on Fe Concentration in Feedwater* (Ref. 29);
 - *Topic Report on Impurity Ingress* (Ref. 30);
 - *Topic Report on Zinc Injection* (Ref. 31), and;
 - *Topic Report on Noble Metal Chemical Addition (NMCA)* (Ref. 32).
74. As I would expect, Chapter 23 of the PCSR (Ref. 11) has a large section dedicated to the RCS operating chemistry and many of the formal explicit claims presented in it are concerned with the importance of controlling the RCS operating chemistry and the functions it provides in mitigating risks to: materials and structural degradation of SSCs, fuel integrity, radiological doses and the management of flammable gases. The claims are too numerous to repeat here, but again, are considered in detail later in the relevant parts of my assessment report
75. For each of the four TRs listed above, I am satisfied their overall scope and approach is reasonable. The main aim of the TRs is to evaluate the impact each of the chemistry parameters has on the behaviour of UK ABWR. Topics such as impact on: fuel integrity, plant integrity, radioactivity, radioactive waste generation and general plant operability issues are therefore covered by each TR. Each report also provides further information on relevant UK ABWR design features and/or operational practices which are in place to manage any detrimental effects associated with each of the chemistry parameters the reports deal with. Refs. 29 – 32 are clearly fundamental parts of Hitachi-GE's overall safety case for UK ABWR.
76. Considering the overall importance of RO-ABWR-0022 (Ref. 25), I also engaged several TSCs to provide independent advice on BWR operating chemistry choices and the factors they would consider and why, in reaching a decision on which operating chemistry to select for a BWR. The output is presented in Refs 33 – 35. The independent advice provided by my TSCs was a very valuable and necessary contribution to support my overall conclusion on Hitachi-GE's final choice of operating chemistry for UK ABWR.

77. Hitachi-GE's approach to providing the necessary demonstration to respond to RO-ABWR-0022 (Ref. 25) and provide "ALARP demonstrations" for chemistry control during all modes of operation, was to conduct a series of workshops with representatives from a number of relevant technical disciplines including: chemistry, structural integrity, radiation protection, radioactive waste management and decommissioning and fuel and core (Ref. 26) and (Ref. 28). The scope of the disciplines invited clearly shows Hitachi-GE understands the importance of balancing a number of competing factors in deciding which RCS operating regime to implement. Furthermore, from Ref. 28, it is clear that Hitachi-GE engaged various independent recognised BWR industry experts in reactor chemistry related matters to support their decision making process. The main parts of the process were:
- All representatives were asked to score (relative to a baseline option (NWC)) several RCS operating chemistry choices against a range of different attributes and sub-attributes; to evaluate their relative benefits and detriments. For at-power operations, Hitachi-GE considered seven distinct operating chemistry regimes, which based on existing BWR practices, are demonstrably bounding. The options ranged from considering NWC only, all the way through to HWC, OLNC and DZO and partial options in-between, including (amongst others), NWC plus DZO, and/or whether NMCA should be implemented.
 - Hitachi-GE explicitly included the materials choices for the CFDW system when deriving the options to consider, because it could have a direct impact on the role the operating chemistry plays in mitigating relevant risks. It may necessitate the need for the at-power operating chemistry regime for the RCS to also include feedwater oxygen injection.
 - Hitachi-GE also considered several principal options for feedwater iron control in Refs. 27 and 28. These ranged from operating with extremely low levels, to concentrations similar to many currently operating BWR plants, along with various combinations of options in-between.
78. Reflecting on the scope of the options considered it is clear Hitachi-GE has captured all credible available options, in the context of BWR RCS operating chemistry choices.
79. Refs. 27 and 28 present the assumptions and key pieces of evidence used by Hitachi-GE during options scoring. This includes a review of worldwide operating practices and experience for BWR RCS chemistry control. I am satisfied the information is comprehensive, appropriate and relevant. I consider it represents suitable and sufficient evidence as a basis to judge the relative benefits and detriments for the purposes of an options scoring exercise. Furthermore, Ref. 26 and Ref. 28 each provide detailed reasoning to underpin the scoring for all options.
80. The principal relevant risks Hitachi-GE considered in deciding on the optimum chemistry choice(s) for the RCS during each phase of operation included:
- SCC of austenitic materials;
 - Radiation doses to workers as result of increased ^{16}N carry-over to the main steam line and the generation, transport and deposition of activated corrosion products such as ^{58}Co and ^{60}Co .
 - Fuel integrity and the possibility of fuel failures.
81. I am satisfied these are the most important aspects to focus on, but I also note from Refs. 27 and 28 that Hitachi-GE has given consideration to the full scope of a range of relevant topics I would expect to see for this particular ALARP demonstration.
82. Based on the information summarised in Ref. 26, Hitachi-GE's overall decision for the at-power RCS operating chemistry for UK ABWR is to implement:

- HWC to mitigate the risk of SCC and avoid the need to possibly replace UK ABWR SSCs which have an increased residual risk of developing SCC cracks if NWC is applied.
 - OLNC to reduce the feedwater hydrogen concentrations required to achieve SCC mitigation of the target SSCs and therefore the amount of ¹⁶N carry-over to the main steam line.
 - DZO injection to reduce the generation and transport of radioactivity and worker doses; specifically to suppress the deposition of ⁶⁰Co on CUW system main piping.
 - Trend plant operations to <1 ppb feedwater iron, or move to “optimum iron” control if required to achieve “low” RCS and ex-core dose rates.
 - Oxygen injection into the main feedwater line to mitigate Flow Accelerated Corrosion (FAC) in CFDW system main piping.
 - Minimise impurities, as is common with NWC.
83. Hitachi-GE argue this decision reflects the latest advances in BWR chemistry control and therefore represents relevant good practice. In a chemistry context, they conclude this option, on balance, represents the overall minimum level of residual risk that can be achieved for SCC, worker doses and fuel integrity for UK ABWR. Ref. 26 also includes a forward action plan; in which Hitachi-GE recognise the importance of addressing other “residual detriments” associated with the RCS at-power operating chemistry decision, including the need to optimise the materials selected for the design. I have assessed materials selection decisions for UK ABWR in detail in Section 4.3.2 of my report.
84. Based on Hitachi-GE’s decision for the at-power RCS operating chemistry, I note the specific technology selected to apply platinum is OLNC. While I am broadly satisfied with the justification provided by Hitachi-GE to select OLNC, a more minor observation I have relates to the terminology used throughout the generic safety case. The main submission supporting the selection of OLNC, Ref. 32, is titled, *Topic Report on Noble Metal Chemical Addition*. There are subtle, yet important differences, between the available technologies to apply noble metals to BWRs. Strictly, NMCA (also known as CNC) refers to the original noble metal application process applied for BWRs. This process is performed during hot shutdown, prior to a refuelling outage, and can also use rhodium as the noble metal. OLNC, on the other hand, always uses platinum and is performed when the reactor is at-power. While I am satisfied this does not undermine the generic safety case, it is important to note that the proposed option “accepted” by ONR during GDA is specifically, OLNC. I expect a future licensee to undertake a review of, and update the UK ABWR safety case accordingly, to ensure the correct terminology is used consistently throughout, when referring to noble metal chemistry. **I consider this to be a Minor Shortfall:**
- MS-ABWR-RC-01:** In the UK ABWR generic safety case, the Topic Report containing the necessary evidence to underpin the decision to select On-line NobleChem™ as part of the at-power operating chemistry is titled *Noble Metal Chemical Addition*. There are subtle, yet very important differences between these two technologies. The licensee should review and update the UK ABWR safety case to ensure the correct terminology is adopted throughout, when referring to noble metal chemistry.
85. Despite this minor discrepancy, in terms of Hitachi-GE’s overall decision for the “ALARP” RCS at-power operating chemistry, as a suite of documentation to underpin it, I consider Refs. 27 and 28 represent a particularly strong and comprehensive demonstration of ALARP when judged against the guidance contained in Ref. 4. In some places Hitachi-GE has used relatively sophisticated methods to analyse and evaluate the robustness of their conclusions. This provides the necessary confidence that a robust process has been followed, and therefore that a robust decision has been reached. Hitachi-GE has clearly worked very hard on this aspect of the generic safety

case during GDA and on the basis of the information I have provided above, I am satisfied that the expectations of SAPs ECH.1 and ECH.2 have been met.

86. There is an important caveat to draw out here at this stage in my assessment report though. While I consider Hitachi-GE has demonstrated the RCS at-power operating chemistry selected is broadly justifiable, my assessment of other fundamental aspects of the UK ABWR generic design and safety case which come in tandem with this decision, most notably the need to optimise materials selection decisions, amongst other matters, are assessed in much more detail in the proceeding parts of Section 4.3 of my report. My more detailed assessment of the overall impact of this decision has resulted in several Assessment Findings (Annex 8) being raised.
87. Implementing HWC, OLNC and DZO injection represents a change to the UK ABWR RCS operating chemistry, compared to the J-ABWR reference plant, which uses NWC. I broadly agree with Hitachi-GE that implementing this modified operating chemistry choice is a justifiable and reasonable change to make for a “new”, modern BWR design, like UK ABWR because:
- The design life for J-ABWR is 40 years; whereas the design life for UK ABWR is 60 years.
 - UK ABWR RINs are largely welded components. Some BWRs are designed to have predominately bolted RINs. The basis for Hitachi-GE’s generic safety case is therefore to avoid SSC replacement. An important contribution to this aim is to reduce the residual risk of SCC in RINs by implementing HWC, as opposed to NWC.
 - When optimised with suitable materials choices, on balance, there is no overall safety detriment to implementing HWC, OLNC and DZO injection compared to NWC. The technical aspects supporting this overarching conclusion are assessed in much more detail from Section 4.3.1, up until the end of Section 4.3, of my report.
 - Hitachi-GE’s choice of operating chemistry demonstrates they are beginning to optimise the ABWR design with a UK health and safety regulatory context in mind, in particular the requirement to minimise relevant risks SFAIRP.
88. Overall, in principal, I agree with Hitachi-GE’s assertions that they have selected an at-power operating chemistry regime for the UK ABWR RCS which represents relevant good practice. My own conclusion is also broadly aligned with the balance of independent advice provided to me by my TSCs (Ref. 33 – 35). Hitachi-GE’s scope and approach for deciding upon the at-power operating chemistry for the RCS is consistent with my TSC’s wealth of knowledge and experience in BWR chemistry matters.

4.3.1.2 Start-up and Shutdown Chemistry Choice

89. Once the operating chemistry choice for at-power conditions is chosen, a similar consideration needs to be given to how the operating chemistry during start-up and shutdown periods should be controlled. These periods, although short in length, can be particularly significant in terms of some of the relevant risks the operating chemistry mitigates (i.e. SSC integrity, radioactivity and fuel reliability and performance). Chemistry control options for the RCS during reactor start-up and shutdown is considered by Hitachi-GE in Refs. 27 and 28. In terms of scope and approach, these reports are broadly similar to the TR produced for at-power operations (Ref. 26). The “ALARP option” identified by Hitachi-GE for chemistry control during start-up is:
- Hydrogen injection begins between 5-10% reactor thermal power;
 - DZO and main feedwater oxygen injection automatically follow feedwater flow rates; and

- Implement OLNC as soon as practicable, but constrained by limits and conditions (or ORs) on fuel reliability and performance.
90. For shutdown operations Hitachi-GE's "ALARP option" for chemistry control is:
- Hydrogen, DZO and main feedwater oxygen injection automatically follows feedwater flow rates and trips at specified low flow conditions.
 - Conduct a so-called "soft shutdown", by carefully controlling the reactor cooling rate.
 - Use Low Temperature Shutdown Cooling mode to bring the RHR system into service. (This is described in further detail in Section 4.3.3.5 as part of my assessment of operating practices to reduce radioactivity).
91. For both start-up and shutdown conditions these particular chemistry controls are also supplemented by the key requirement to minimise impurities.
92. Chapter 23 of the PCSR (Ref. 11) does not make many explicit claims regarding start-up and shutdown operations. The main claims are presented at such a level that they can effectively be considered as equally as applicable, irrespective of the operating mode. There is only one explicit claim which specifically relates to start-up and shutdown operations:
- "[RC SC1.3]: *SCC propagation will be mitigated during start-up by minimising the reactor water oxygen.*"
93. Taking claim [RC SC1.3] as read, the scope of Ref. 11 appears to be relatively narrow, with respect to start-up and shutdown operations. The main body of the text in Ref. 11 does, however, provide more detail regarding chemistry control and associated measures required during shutdown operations. However, unlike start-up operations, there are no formal, explicit claims presented for shutdown. Overall though the main claims, as presented, could be equally applicable regardless of the operating mode, and the main body of Chapter 23 of the PCSR (Ref. 11) does provide further detail on chemistry control during start-up and shutdown operations. On this basis I am satisfied that as the top level of a generic safety case, Ref. 11 gives adequate coverage to the importance of chemistry control during reactor start-up and shutdown.
94. Hitachi-GE's hierarchy of generic safety case documentation also contains a specific TR on this subject, *Topic Report on Start-up and Shutdown Chemistry* (Ref. 36). The TR contains more detailed safety case evidence necessary to support Hitachi-GE's claims and arguments in Refs. 27 and 28. Ref. 36 also presents a list of claims and sub-claims; however, they are not consistent with the structure and wording of the claims presented in Chapter 23 of the PCSR (Ref. 11). This does not undermine the case presented, but it is an inconsistency which I consider a licensee should aim to rectify. **I consider this to be a Minor Shortfall:**
- MS-ABWR-RC-02:** The safety case claims made on commissioning, start-up and shutdown chemistry in Chapter 23 of the generic UK ABWR Pre-construction Safety Report are inconsistent with the claims presented in the related Topic Reports, in the lower tiers of the hierarchy of UK ABWR generic safety case documentation. The licensee should revise and update the UK ABWR safety case to ensure a consistent set of claims are presented.
95. Ref. 36 summarises a number of chemistry control practices that are implemented for BWRs during reactor start-up and shutdown operations to mitigate several risks and hazards. A number of the practices are presented and assessed later in Section 4.3.3.5 of my report. Ref. 36 provides selected evidence to underpin each practice and seeks to justify which ones are most applicable to implement for UK ABWR. In summary, Ref. 36 concludes several of the measures should be implemented for UK

ABWR. In addition to what would be considered “normal” for any reactor (such as minimising impurities and removing radioactivity), Ref. 36 identifies a number of measures for UK ABWR owing to the adoption of HWC, OLNC and DZO injection.

96. In support of my assessment of Ref. 36 I asked my TSC to collate and review relevant good practice for BWR chemistry start-up and shutdown operations and to undertake an independent review of the practices Hitachi-GE has described above, against the identified relevant good practice. The outcome of their work is reported in Ref. 37. Their overall conclusion based on their experience in BWR chemistry matters and the review of international relevant good practice, is that the start-up and shutdown chemistry practices Hitachi-GE are suggesting for UK ABWR are broadly fit-for-purpose.
97. Considering the choice of HWC, OLNC and DZO injection for the at-power operating chemistry, I am content Hitachi-GE has approached the chemistry choices during start-up and shutdown periods in a consistent and similarly robust manner. It is clear from the detailed supporting evidence provided in Ref. 36 that a comprehensive range of measures were considered.
98. As explained in Section 4.3.1.1 for the RCS at-power operating chemistry choice, once again, the same important caveat applies here for this decision. While I consider Hitachi-GE has demonstrated the RCS start-up and shutdown chemistry choice selected is broadly justifiable, my detailed assessment of the overall impact of this decision on the UK ABWR generic design and safety case; is presented in the proceeding parts of Section 4.3 of my report. This has resulted in several Assessment Findings (Annex 8) being raised, which are related to this topic.

4.3.1.3 Commissioning Chemistry Choice

99. The chemistry adopted during commissioning of a nuclear reactor is generally accepted to be one of several key factors which has important consequences for the longer-term behaviour of the plant; notably on the build-up and release of radioactivity and the plant’s susceptibility to materials degradation threats. Many nuclear power plants therefore pay close attention to the chemistry control measures implemented during this phase of reactor operations, in order to achieve low plant dose rates during the operating life of the facility. Conversely, the precise conditions and timings for such periods are greatly influenced by commercial constraints and licensee choices. The scope of assessment during GDA is therefore limited, and mainly focuses on demonstrating that the UK ABWR design has not foreclosed options that may have significant benefits for longer term plant safety.
100. As well as Refs 27 and 28, Hitachi-GE’s hierarchy of safety case documentation also contains a specific TR on commissioning, *Topic Report on Commissioning Chemistry* (Ref. 38). The TR contains more detailed evidence which supports the claims and arguments in Refs. 27 and 28 regarding the adequacy of UK ABWR chemistry control measures proposed during commissioning.
101. Hitachi-GE’s initial approach for GDA was to assert that based on past practices for commissioning BWRs, the “ALARP option” for UK ABWR commissioning would be to use deaerated, demineralised water with minimised impurities. Hitachi-GE approached the topic with some certainty and the generic safety case at that time essentially constrained the future operator to take this very narrow and specific course of action. This largely dismissed the importance of tailoring chemistry control measures during commissioning; in particular relating to the minimisation of deposited radioactivity, and hence worker Operational Radiation Exposure (ORE). Previous UK experience and recent international work clearly demonstrates that real benefits for ORE can be achieved during commissioning (Ref. 39).

102. Considering these benefits and given the selection of HWC, OLNC and DZO injection for the at-power operating chemistry regime for UK ABWR, it is not clear whether past practices for commissioning BWRs are directly applicable to UK ABWR. This is because no BWR has ever commenced operations from the very first fuel cycle using this particular at-power operating chemistry. During GDA I therefore raised RO-ABWR-0072 (Ref. 25). The focus was to ensure technically feasible and/or reasonably practicable chemistry control options which a future licensee may choose to implement to commission UK ABWR, had not been foreclosed by the generic design and/or safety case.
103. Similar to start-up and shutdown operations, there is only one explicit claim made in Chapter 23 of the PCSR (Ref. 11) related to chemistry control during commissioning:
- “[RC SC5.4] *Commissioning activities will be optimised to ensure that the longer-term radiation exposure will be minimised to ALARP*”.
104. Again though, the other explicit claims could be interpreted as being equally applicable during commissioning. Presenting claim [RC SC5.4] in Chapter 23 of the PCSR (Ref. 11) is a very positive and important step in providing the necessary visibility of the importance of implementing adequate commissioning chemistry. In addition, claim [RC SC5.4] is very clearly targeted towards the importance of aiming commissioning activities at reducing longer-term ORE. This clearly shows, even at the top-level of the safety case, Hitachi-GE has given a more appropriate consideration to the aims and objectives of commissioning chemistry. My detailed assessment of the evidence supporting claim [RC SC5.4] is presented in Section 4.3.3.4 of my report.
105. *Topic Report on Commissioning Chemistry* (Ref. 38) also presents a set of claims on commissioning chemistry which do not appear to be replicated in Chapter 23 of the PCSR (Ref. 11). This does not undermine the case presented, but it is an inconsistency which I consider a licensee should rectify. **This has been captured as part of Minor Shortfall MS-ABWR-RC-02.**
106. The approach in Ref. 38 is to separate the definition of commissioning into three discrete phases:
- Phase one: Pre-operational commissioning at <100°C without fuel loaded.
 - Phase two: Pre-operational commissioning at ≥100°C without fuel loaded.
 - Phase three: Commissioning at ≥100°C with fuel loaded.
107. The definition is based on the different risk and hazards, their potential significance, and the availability of key plant systems. For each phase, Hitachi-GE gives separate consideration to potential options to mitigate the degradation (mainly SCC) of structural materials, fuel failure risks and doses to workers. Owing to the nature of reactor commissioning activities, in practice there are many individual stages of commissioning, but I consider Hitachi-GE’s definition of the three different phases is appropriate at this stage, as it reflects the main systems and risks involved.
108. Ref. 38 also provides a detailed breakdown of the major system tests typically performed to commission J-ABWR and a separate breakdown of the specific tests performed before fuel is loaded, which are directly related to chemistry control activities during commissioning. The report also provides information on the availability and capability of relevant UK ABWR systems during commissioning and the typical chemistry measurements that need to be taken during commissioning activities. As well as presenting the overall conclusion of the “ALARP option” for each of the three phases, four options are left open for a future licensee to give further detailed consideration to. These relate to using hydrazine during system flushing and storage and the “early” implementation of hydrogen, OLNC and DZO injection. In effect the decision regarding the “early” adoption of the at-power UK ABWR operating chemistry

is left for the future licensee to make. In general, at this generic stage of the project, I consider this is reasonable.

109. In the context of making a reasoned and informed choice, it is clear Hitachi-GE has given a significant amount of thought to the chemistry control options available during commissioning. Ref. 38 provides evidence and assertions to support the basis of the conclusions reached. It is neither possible, nor reasonable, to present all of it here. In terms of decision making, I am satisfied Hitachi-GE has supplied sufficient information at this stage to justify what chemistry control during commissioning is aiming to achieve for UK ABWR. Based on my TSC's experience of BWR plants, they are also satisfied that the aims, objectives and target SSCs identified by Hitachi-GE appear reasonable for this stage of the project (Ref. 37). I also consider that Rev. 2 of Ref. 38 deals with uncertainty much more appropriately. The overall conclusion is that there are options still available and important decisions to make for a future licensee regarding chemistry related UK ABWR commissioning activities. These include earlier introduction of DZO, platinum and/or hydrogen. In this regard, I am content the generic safety case does not impose any unnecessary constraints on a future licensee.
110. As explained in Section 4.3.1.2 for the RCS start-up and shutdown operating chemistry choice, once again, the same important caveat applies here for this decision. While I consider Hitachi-GE has demonstrated the position on the RCS commissioning chemistry choice is broadly justifiable for GDA, my detailed assessment of the overall impact of this decision on the UK ABWR generic design and safety case; is presented in the proceeding parts of Section 4.3 of my report. This has resulted in several Assessment Findings (Annex 8) being raised, which are related to this topic.

4.3.1.4 Operating Chemistry Summary

111. The choice of operating chemistry regime to be applied to the RCS for UK ABWR is a vital decision in the context of making the safety case for chemistry. It directly influences the claims, and therefore justifications that need to be provided. As such, this was the most significant decision taken by Hitachi-GE during GDA in the chemistry topic. My assessment was therefore correspondingly targeted towards the basis and justification for the UK ABWR operating chemistry choice made by Hitachi-GE, particularly in terms of the demonstration that relevant risks will be reduced SFAIRP.
112. For all normal operating modes, the technical basis and rationale for the chemistry control decisions taken are prominent features in the relevant parts of the generic safety case. The decision making processes applied by Hitachi-GE are rigorous and consider the most important and relevant factors I expect to see.
113. During my assessment of Hitachi-GE's approach to underpinning the normal operating mode chemistry decision(s) for UK ABWR, I have raised two Minor Shortfalls, **MS-ABWR-RC-01 and -RC-02**. The subsequent parts of Section 4.3 of my assessment report build on my consideration of Hitachi-GE's approach to justifying the RCS operating chemistry decision(s), to consider each relevant risk in turn, as well as the capability of the generic design to safely implement the operating chemistry choices made.

4.3.2 Materials Degradation

114. Over the many decades BWRs have been operating, a large number of materials degradation threats have emerged. These have occurred across several different material classes (including certain nickel-base alloys (NBAs) and certain types of austenitic SS) and operating chemistries. They have been attributed to both off-chemistry scenarios and/or sub-optimal materials conditions associated with manufacturing. Degradation has occurred within RCS components, piping and in generic systems that apply to a number of different BWR plant designs (Ref. 40). In

addition to the issues that have arisen with materials degradation, other considerations such as reducing ORE have driven the concurrent development of BWR materials and operating chemistry choices.

115. Operating under oxidising NWC conditions, combined with residual high tensile material stresses led to a series of Inter-granular Stress Corrosion Cracking (IGSCC) events being reported for BWRs during the 1960s and 1970s. Different plants and operators have responded in different ways to tackle these problems. Some of the measures include adopting extensive materials and/or component replacement programmes and/or modifying the operating environment (to HWC) to mitigate the effects of degradation. Considering this synergy, ONR expects the choice of operating chemistry to be considered in tandem with the materials selection process. In addition, given the high number of materials degradation events experienced in early BWRs, ONR expect the generic safety case for a “new” BWR plant such as the UK ABWR, to provide a robust and proportionate justification for the materials selected for the plant, to be able to demonstrate the risks associated with materials degradation are well understood, have been appropriately addressed by the design and reduced SFAIRP.
116. Owing to the importance of the materials topic to chemistry and the long history of materials degradation events in BWRs, I placed a considerable emphasis on assessing materials degradation threats in UK ABWR during my GDA chemistry assessment. This emphasis was also driven by the chemistry claims made by Hitachi-GE in the generic safety case; a large majority of which are associated with the impact of chemistry on maintaining the integrity and performance of a range of UK ABWR SSCs. During GDA I therefore raised a joint RO with ONR’s structural integrity specialist inspector, RO-ABWR-0035 (Ref. 25). The overall intent of RO-ABWR-0035 (Ref. 25) was to influence Hitachi-GE to widen the scope of their materials selection decisions provided during GDA. Ref. 25 was a long-standing RO throughout GDA and was subsequently revised several times following ONR’s assessment of Hitachi-GE’s submissions in response to it.
117. The main SAPs (Ref. 2) of relevance to my assessment of chemistry and materials degradation threats are EKP.1, ECH.1, EAD.1 and EAD.2. In addition to the SAPs, NS-TAST-GD-088, *Chemistry of Operating Civil Nuclear Reactors* (Ref. 4) provides specific guidance on ONR’s expectations for how the effects of chemistry on materials selection and degradation ought to be addressed in safety cases for Light Water Reactors (LWRs). Guidance in NS-TAST-GD-005, *Guidance on the Demonstration of ALARP* (Ref. 4), is also pertinent to parts of my assessment.
118. To undertake my assessment of this topic I also made use of TSCs. I let three separate work packages to TSCs, the outputs of which formed an important input to my assessment. The work included:
 - Undertaking an initial review of UK ABWR material degradation threats and materials selection (Ref. 40).
 - Supporting my assessment of Hitachi-GE’s approach to undertaking materials degradation threats risk ranking for UK ABWR SSCs (Ref. 41 and Ref. 42).
 - Providing independent advice as part of a “working group” on BWR chemistry, where the majority of topics discussed concerned matters associated with materials selection and degradation (Ref. 43).
119. My assessment of materials degradation that follows has therefore been broadly split into three discrete sub-topics:
 - mitigation of materials degradation by the operating chemistry;
 - Hitachi-GE’s overall approach to justifying materials choices; and
 - a sample assessment of the residual materials degradation risks for UK ABWR SSCs.

4.3.2.1 Mitigation of Materials Degradation by the Operating Chemistry

120. Chapter 23 of the PCSR (Ref. 11) recognises the important link between chemistry and materials selection and degradation. There are numerous claims made on chemistry with regard to the requirement to maintain the integrity of a number of UK ABWR SSCs. The relevant “top claims” made by Hitachi-GE are:
- “[RC SC1] *The UK ABWR reactor chemistry regime will contribute to the maintenance of the integrity of the RPV and SSCs in contact with the reactor coolant by controlling within the limits and conditions*”.
 - “[RC SC2] *The UK ABWR reactor chemistry regime will contribute to the maintenance of the integrity of the associated SSCs which contact with the primary coolant outside of the RPV by controlling within the limits and conditions*”.
121. Specific claims then made under [RC SC1] are:
- “[RC SC1.1] *SCC propagation will be mitigated by appropriate Electrochemical Corrosion Potential Control by HWC and OLNC.*”
 - “[RC SC1.2] *Corrosion (such as SCC, FAC) and corrosion rates will be mitigated by control of impurities by operation of the clean-up systems.*”
 - “[RC SC1.3] *SCC propagation will be mitigated during start-up by minimising the reactor water oxygen.*”
122. For [RC SC2], the same claims for [RC SC1.1] and [RC SC1.2] are made, plus:
- “[RC SC2.1] *Corrosion such as FAC and pitting corrosion will be reduced by oxygen control.*”
123. Overall, I am content these are reasonable claims to make. I have considered the adequacy of the safety case evidence presented by Hitachi-GE in support of these claims in much more detail in my assessment that follows. Like the other relevant risks covered by my assessment report, the majority of Hitachi-GE’s detailed arguments and evidence to demonstrate the impact of the operating chemistry on the potential degradation of materials, is contained in the suite of chemistry-related TRs which cover: NMCA (and HWC), zinc, iron, impurity ingress, start-up and shutdown, and commissioning chemistry (Refs 29 – 32, 36 and 38). Chapter 23 of the PCSR (Ref. 11) and the *Water Quality Specification* (Ref. 20) also present relevant limits and conditions (or ORs) which the UK ABWR operating chemistry needs to be controlled within. The focus of my assessment presented here is also to consider the adequacy of these ORs, with respect to minimising risks to materials degradation.

Impact of HWC and OLNC on Materials Degradation

124. As previously explained, the principal reason why BWRs inject hydrogen and platinum into the feedwater is to mitigate the risk of SCC of austenitic materials. In Chapter 23 of the PCSR (Ref. 11), Hitachi-GE therefore makes specific claims on the UK ABWR operating chemistry which reflect this objective (claim [RC SC1.1], as described above).
125. The UK ABWR operating chemistry uses HWC in combination with the injection of platinum to provide a reducing environment and therefore suppress the Electrochemical Corrosion Potential (ECP) into a region where SCC is mitigated. ECP is a measure of the thermodynamic tendency of a material to undergo an electrochemical (i.e. corrosion) reaction. In BWRs, ECP is also a measure of the electrochemical driving force for SCC. The higher the ECP, the greater the thermodynamic tendency for SCC initiation and growth in materials susceptible to SCC. Importantly, the claim made by Hitachi-GE on SCC mitigation ([RC SC1.1]) is

based on ECP. Strictly this could be achieved by HWC without OLNC, however as detailed further in Section 4.3.3.4 of my report, the combination of HWC and OLNC has further benefits in terms of radioactivity behaviour.

126. Ref. 32 explains that SCC crack growth rates are sufficiently mitigated if the ECP of the target SSC is reduced to <-230 mV (vs. Standard Hydrogen Electrode (SHE)). Extensive studies have shown lowering the ECP of austenitic materials used in BWRs to <-230 mV (vs. SHE), by injecting hydrogen into the feedwater (i.e. HWC), coupled with reducing the coolant conductivity, mitigates SCC. Consequently, operating to achieve an ECP of <-230 mV (vs. SHE) is the industry standard accepted value, below which all operational BWRs which implement HWC aim to achieve “protection” for regions of the plant where SCC mitigation is required.
127. It is clear, therefore, <-230 mV (vs. SHE) is an acceptable value to select for UK ABWR. Ref. 32 presents information from a number of wide ranging and authoritative sources, which taken together constitute suitable and sufficient evidence to justify why an ECP target value of <-230 mV (vs. SHE) is appropriate. This is consistent with evidence elsewhere, for example Refs 44 and 45. I am therefore content with the technical basis for this value. Typical data for SSs showing the large reductions in SCC crack growth rates when ECP is suppressed to low potentials is shown in Figure 6 below.

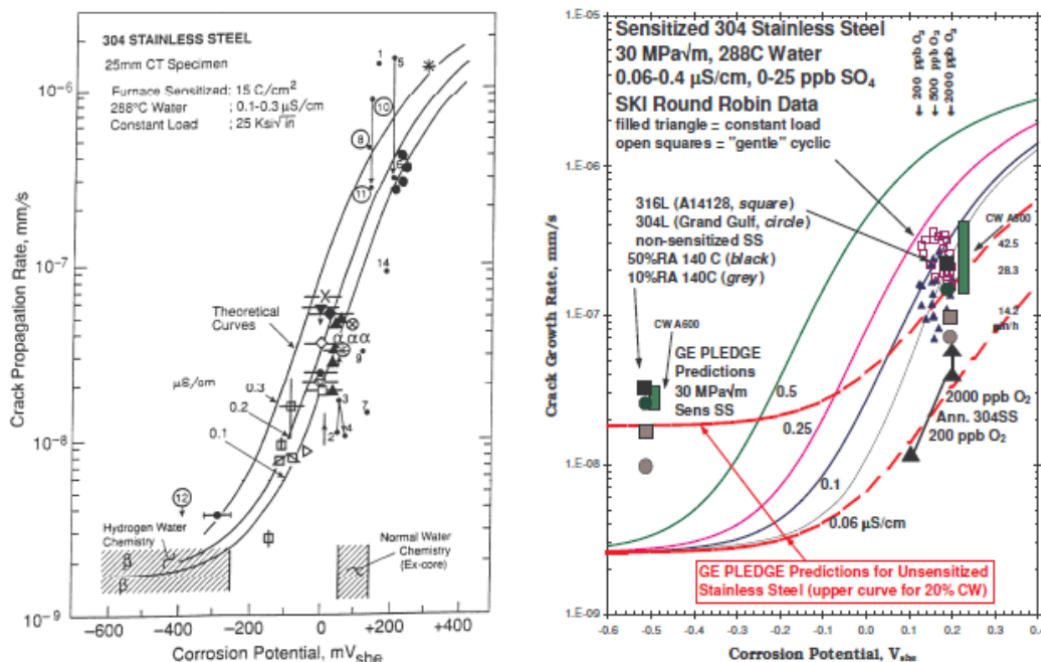


Figure 6: SCC crack growth rates at differing ECP values for SS (Ref. 46)

128. It is important to clarify that Hitachi-GE's claim on the operating chemistry is about mitigating SCC propagation, not initiation. I am satisfied with the scope of this claim, as justifying a benefit in stopping SCC initiating by chemistry would be difficult. Some data exists which shows operating with ECP <-230 mV (vs. SHE) mitigates SCC crack initiation, however, as reflected in the generic safety case evidence provided by Hitachi-GE in Ref. 32, the vast majority of studies focus on the effect of chemistry in mitigating SCC crack growth rates.
129. It is also important to understand that the operating chemistry applied cannot provide “protection” against SCC to all parts of the RCS. As described earlier, by definition, as boiling inside the core of a BWR, hydrogen is removed from the reactor water and platinum cannot reach parts of the RCS which contain steam. At these locations the

chemistry means an ECP < -230 mV (vs. SHE) cannot be achieved. This means that for any BWR there are parts of the core that are oxidising and parts that are reducing, irrespective of the operating chemistry. This behaviour is further complicated by a number of other factors which ultimately mean the responsiveness of BWRs to HWC is very much plant specific. Some of the factors that affect this responsiveness include:

- Neutron and gamma dose rates in the downcomer region. This region is the part of the BWR main circuit between the shroud wall and the RPV wall.
 - The core design and the power distribution in the core.
 - Plant geometry factors such as the shroud wall thickness and width of the shroud wall-RPV gap.
 - Coolant flows around different parts of the RCS and the degree of mixing between injected hydrogen and reactor coolant, this leads to.
130. Many of these factors are important due to the fact that much of the recombination of hydrogen and oxidants in a BWR occurs around the downcomer region, driven by the radiation present and also the thermal hydraulic conditions. This variability from plant-to-plant and even from cycle-to-cycle, means determining which parts of the RCS are “protected” from SCC, and how much hydrogen is needed to achieve “protection”, are complex. It is for many of these reasons that operating BWRs make extensive use of radiolysis modelling and ECP calculations. These calculations are used to determine the levels of feedwater hydrogen required for the plant from cycle-to-cycle, and at different points in the same fuel cycle, to be able to achieve SCC mitigation in the required locations.
131. Considering these technical aspects of BWR behaviour, the claims ([RC SC1], [RC SC1.1] and [RC SC2]) Hitachi-GE formally identify and present in the generic safety case are aligned with what I expect to see. However, despite this conclusion, I also consider the claims are very high-level and not specific enough to the UK ABWR design. Importantly, they do not reflect which specific parts of the plant HWC and OLCN are providing SCC mitigation for in UK ABWR. During GDA, I therefore asked Hitachi-GE to clarify which specific UK ABWR SSCs the above claims ([RC SC1], [RC SC1.1] and [RC SC2]) apply to. Hitachi-GE’s response was that target SSCs for SCC mitigation are:
- SS sections of the RHR and CUW systems; and
 - The bottom head region of the RPV.
132. These locations are identified in Ref. 32; however, the same information is not reflected in Chapter 23 of the PCSR (Ref. 11). I would expect the PCSR to at least inform the reader which specific regions/SSCs the plant operating chemistry is claimed to provide SCC mitigation for. This is important, but it is largely a PCSR presentational matter given it is clear from other parts of the lower tier safety case documentation, where SCC mitigation by the operating chemistry is aimed at.
133. In Ref. 32, Hitachi-GE lists four criteria they have applied to the UK ABWR generic design to decide which specific SSCs it is important the operating chemistry provides SCC mitigation for. If an SSC meets three or more of the criteria, it is selected. These criteria appear reasonable; however, having to meet three of the four is arbitrary. Nevertheless, based on the design and materials choices for the UK ABWR, the outcome is SSCs I would expect the UK ABWR operating chemistry may need to play a key part in mitigating the risk of SCC for.
134. The detailed evidence to support the selection of these locations is provided in, *Topic Report on Radiolysis and ECP Model* (Ref. 47). This TR is Hitachi-GE’s principal source of evidence to justify why HWC will be effective for UK ABWR. It contains a description of Hitachi-GE’s radiolysis model and methods and tools they use to undertake radiolysis and ECP calculations to assess the effectiveness of HWC for UK

ABWR. The outputs of the model are used as the basis for Hitachi-GE's justification for the normal operating range (i.e. limits and conditions (or ORs)) for feedwater hydrogen for UK ABWR. Ref. 47 is therefore a very important document in the overall "reactor chemistry safety case".

135. Considering the overall importance of Ref. 47, the novelty associated with UK ABWR implementing HWC, and the plant specific nature of the effectiveness of HWC in BWRs, during GDA I engaged a TSC to:
- Undertake an independent review of Hitachi-GE's radiolysis and ECP modelling, codes and methods to determine whether it is fit-for-purpose and represents relevant good practice in the field.
 - Perform independent radiolysis modelling to calculate ECP in order determine the likely effectiveness of the UK ABWR design to HWC.
 - Compare and contrast the outcome of their independent modelling calculations and Hitachi-GE's calculations, to highlight the similarities and differences, and provide advice on the significance of any major differences.
136. The outcome of my TSC's work is presented in Ref. 48. I have drawn heavily on Ref. 48 to determine the adequacy of Hitachi-GE's approach to radiolysis and ECP modelling. In Ref. 47 Hitachi-GE provide radiolysis calculations for two UK ABWR core designs – an initial core and an equilibrium core. The initial core represents the first operating cycle; the equilibrium core corresponds to a cycle with re-used fuel. For each of these core designs calculations are then performed to estimate the hydrogen demand for UK ABWR at the beginning of the fuel cycle (BOC) and the end of the fuel cycle (EOC). In Ref. 48 my TSC has evaluated Hitachi-GE's approach to the key inputs to this model and concludes that:
- In terms of chemistry, the basic radiolysis equations solved in the model are nominally identical to those used in other radiolysis models, but Hitachi-GE's model does not use the latest recommended water radiolysis chemical rate constants.
 - The basis of the dose rates used in the model are unclear.
 - No analysis is provided with regard to flow conditions in some important areas of the plant.
137. During GDA, a number of RQs were raised regarding the input dose rate information used by Hitachi-GE (RQ-ABWR-0406, -0407, -0475, -1099 and -1258 (Ref. 49)). My TSC's experience in comparing the dose rate outputs from Hitachi-GE's code with calculations performed using more modern codes and methods the BWR industry is moving towards, is that Hitachi-GE's code does reasonably well in predicting the dose rates in some parts of the plant, but not in others (Ref. 48).
138. Furthermore, there are a number of over simplifications and uncertainties with the dose rate data provided by Hitachi-GE. For example, the dose rate profiles used by Hitachi-GE did not match with experience from US BWRs which routinely perform the same analysis. This matter was raised with Hitachi-GE in RQ-ABWR-1258 (Ref. 49). Their response explained that neutron dose rate distributions were not available for both the BOC and EOC calculations. Based on this response, there therefore remains some considerable uncertainty regarding the correct input dose rates for UK ABWR. For the purposes of conducting their independent work, my TSC had to evaluate the impact of these uncertainties to be able to draw meaningful and comparable conclusions regarding the adequacy of Hitachi-GE's radiolysis and ECP modelling. **This has been captured as part of Assessment Finding, AF-ABWR-RC-01, below.**
139. As part of my assessment I also asked Hitachi-GE what consideration, if any, had been given to the hydrogen efficiency of the UK ABWR in the core designs presented during GDA. This is because some BWRs design their cores to maximise the

recombination rates in the downcomer, thereby optimising the hydrogen demand of the plant. Ref. 32 essentially states this isn't required for UK ABWR, mainly because of the impact using OLNK in combination with HWC. Ref. 32 provides various technical arguments for why the UK ABWR hydrogen demand has already been optimised based on a number of factors. I am satisfied with these arguments, and this position for GDA. However, as part of normal business, I would expect the future licensee to consider such factors when developing and optimising future UK ABWR core designs.

140. As noted earlier, efficient mixing of feedwater and reactor water is important. The only measured plant data looking at the effectiveness of HWC for the ABWR design, namely mini-test results from the KK-7 reference plant (Ref. 32), suggests that this mixing is sufficient. These are only limited data and the responsiveness of the ABWR plant to HWC may depend more heavily on flow rate; for operations below 100% power, these data may therefore not be representative. BWRs can move into low power conditions following fuel failures or if they load follow, so such situations are not unusual (Ref. 48). For UK ABWR, it is also important to understand the flow in the region of the RIPs. Generally, high flow rates lead to high ECPs. In Ref. 47 there is no description of how the UK ABWR RIPs are accounted for in Hitachi-GE's radiolysis model. This is particularly important given Hitachi-GE's specific claim on the operating chemistry is that it provides SCC mitigation in the bottom head region of the RPV, which is where the RIPs are located.
141. Overall, for the purposes of a generic safety case, I consider Hitachi-GE have provided sufficient evidence on the above aspects. However, given the current level of uncertainty and the overall importance of radiolysis and ECP modelling, both for making the generic safety case and for the future licensee to be able to demonstrate the plant can be within the limits and conditions defined in the safety case, I would expect the licensee to be implementing the latest relevant good practice in radiolysis and ECP modelling. This also includes providing further evidence to demonstrate the responsiveness of UK ABWR to HWC. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-01: No internally pumped Boiling Water Reactor (like UK ABWR) has ever undertaken long-term continued operations using Hydrogen Water Chemistry. To demonstrate the responsiveness of UK ABWR to Hydrogen Water Chemistry under the operating conditions chosen by a future licensee, and to address limitations in the evidence provided during GDA on this topic, the licensee shall:

- Develop and implement an adequate radiolysis and electrochemical corrosion potential model to demonstrate UK ABWR can be controlled within the limits and conditions defined in the safety case.
- Produce evidence to demonstrate adequate consideration has been given to the impact of coolant flow conditions on hydrogen mixing, platinum deposition, and the resultant electrochemical corrosion potential.

142. Notwithstanding the above points, Hitachi-GE's radiolysis modelling does produce several important outputs. For defined locations around the RCS, it provides calculated concentrations for hydrogen, hydrogen peroxide (the most important oxidant produced) and total oxidant. The model also provides calculated ECP values for these locations. Ref. 47 also depicts the distributions of calculated ECP values around the UK ABWR RPV, with and without the operating chemistry of HWC and OLNK, to demonstrate which regions of the plant are, and which are not "protected" from SCC by the operating chemistry.
143. To evaluate the validity of the data contained in Ref. 47, my TSC conducted an independent set of confirmatory analysis using the EPRI BWR Vessel and Internals Application (BWRVIA) computer software for radiolysis and ECP analysis (or VIA model). This model calculates oxidant levels, molar ratio ($[\text{H}_2]/([\text{O}_2] + 0.5[\text{H}_2\text{O}_2])$) and

ECP around a BWR circuit and has been used extensively for both US and non-US BWR systems that operate using HWC. It has also been used to model the effectiveness of HWC for a J-ABWR plant. The model therefore has a very good pedigree. Some of the key conclusions from my TSC's independent radiolysis and ECP modelling (Ref. 48) are as follows:

- The initial core design for the UK ABWR suggests the downcomer region will experience smaller dose rates than the J-ABWR, so HWC would be less effective.
- Uncertainty in the dose rate data provided by Hitachi-GE (as previously described) had consequences for the predicted chemistry behaviour in the downcomer region for both models. The predictions of the BWRVIA model half way along the downcomer differed significantly from those of the Hitachi-GE model. The BWRVIA model prediction of oxidant removal in this region by feedwater hydrogen was that the process is significantly less efficient half way along the downcomer, however;
- Lower plenum (i.e. the bottom of the RPV) chemistry behaviour was similar in both models, which indicates much more recombination occurs in the bottom half of the downcomer in the BWRVIA model compared to the Hitachi-GE model. These differences highlight the importance of having accurate input dose rate data to perform radiolysis and ECP calculations. **This has already been captured as part of Assessment Finding, AF-ABWR-RC-01, above.**

144. In terms of the assessment of Hitachi-GE's argument that UK ABWR operating with a feedwater hydrogen concentration of 0.2 ppm (as assumed in Ref. 47) provides mitigation against SCC, it is useful to compare the calculated hydrogen: oxidant molar ratios in each of the models at a point in the RCS known as the "S1 carry-under point". This point is a location near the top of the downcomer which corresponds to the bottom of the shroud head dome in the BWRVIA model, or the top of the downcomer in Hitachi-GE's model. If the molar ratio at this point is great than two, the molar ratio will remain greater than two in all subsequent parts of the coolant flow path until boiling occurs. This is therefore a key parameter in determining the effectiveness of HWC and how much hydrogen is needed in the feedwater. BWR plants in the US which operate using HWC and platinum injection (i.e. the entire US BWR fleet) calculate the molar ratio at this point and then choose a feedwater hydrogen value that gives a molar ratio of four. This provides the plant with some operational margin in preventing SCC. In Ref. 48, my TSC derived data for the BWRVIA calculated molar ratio at this point, for a range of feedwater hydrogen concentrations, using both Hitachi-GE's and their independently calculated dose rates for the initial and equilibrium core designs at the BOC and EOC. The results show that the feedwater hydrogen concentration needed:

- Varies between 0.16 and 0.23 ppm[†], depending on the conditions and assumptions.
- Is comparable between the initial and equilibrium core designs.
- Is not changed significantly depending on the dose rate data used, varying only by 0.01 ppm between the two sets, and;
- Increases from the BOC to EOC, varying by up to 0.05 ppm.

145. This behaviour is consistent with the trends observed for existing BWRs which implement HWC, suggesting there should be no fundamental differences in the behaviour of UK ABWR.

146. The other very important point to consider when looking at the outputs of the radiolysis modelling, are the chemistry and ECP predictions in the parts of the plant where Hitachi-GE have claimed the operating chemistry needs to mitigate SCC. These are

[†] The feedwater hydrogen operating range specified in the generic safety case is 0.15-0.30 ppm.

the SS piping in the RHR and CUW systems and the bottom head region of the RPV (also referred to as the lower plenum). Both Hitachi-GE and my TSC calculate the water leaving through the “N8 nozzle”[‡] to have a molar ratio greater than two for typical HWC conditions, meaning there is excess hydrogen over oxidant entering these parts of the plant and SCC mitigation by the “environment” (i.e. the operating chemistry) would be expected. There are some important caveats associated with these conclusions:

- My TSC’s calculations are based on the assumption the “N8 nozzle” is located at the top of the downcomer, but this is not clear from the information provided by Hitachi-GE during GDA. The location of the nozzle is important, because as previously explained; the effectiveness of mixing with the main coolant flow is a very important factor for the efficiency of the recombination reaction.
- This is based on the assumption that platinum has managed to reach and coat the SS surfaces of this part of the plant. Hitachi-GE has assumed that the platinum loading is sufficient at all locations in the CUW system for the surfaces to be fully catalytic (Ref. 48).

147. Overall, based on these considerations, for GDA I am satisfied with Hitachi-GE’s claim that the operating chemistry offers SCC mitigation for SS piping in the CUW and RHR systems. This is based on the two assumptions described above. These do not undermine the overall generic safety case presented, but they are important assumptions that will require the necessary clarification and evidence to be provided by a future licensee, to demonstrate they are reasonable. The feedwater hydrogen demand for UK ABWR may change based on these assumptions. **This has been captured as part of Assessment Finding, AF-ABWR-RC-01, above.**
148. Hitachi-GE’s other main claim on the impact the operating chemistry has on SCC mitigation, is that it offers SCC “protection” of the RPV bottom head. Figure 7 shows some of the chemistry predictions (total oxidant, hydrogen and hydrogen peroxide) for this region of the plant calculated using Hitachi-GE’s and the BWRVIA radiolysis model. It also shows the resultant, calculated, ECP values.

[‡] The “N8 nozzle” is the point at which the RHR and CUW systems take flow from the RPV.

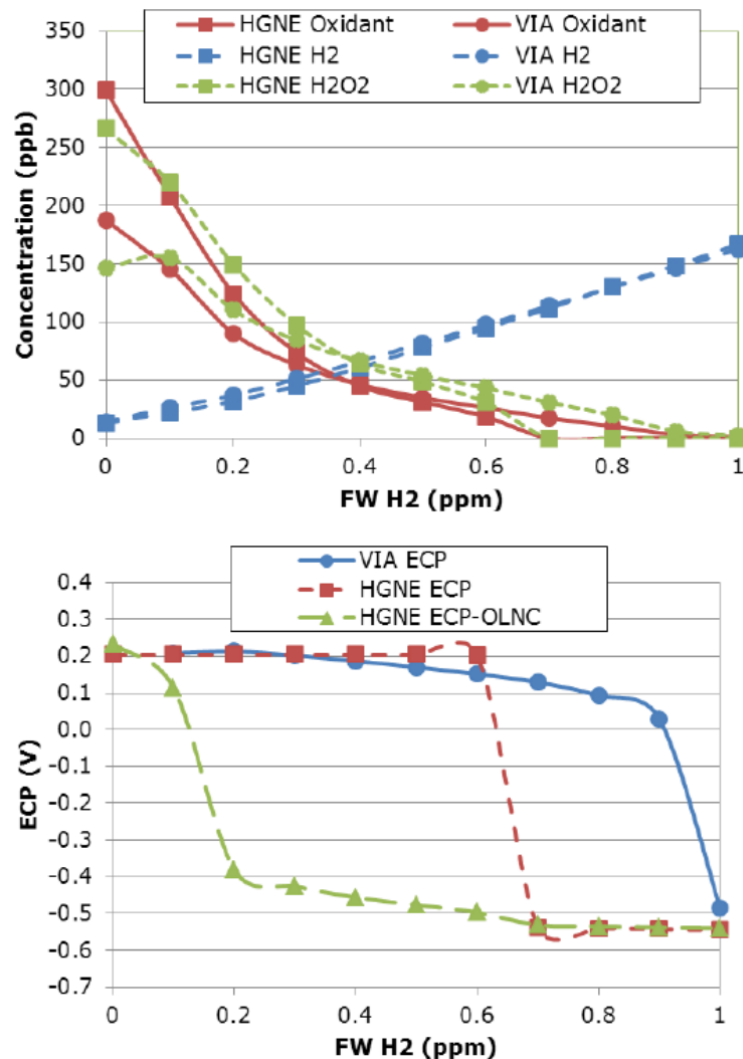


Figure 7: Comparison of Hitachi-GE and BWRVIA calculated chemistry parameters and ECP using Hitachi-GE dose rate data for the RPV bottom, initial core at BOC (Ref. 48)

149. Figure 7 demonstrates both models predict similar chemistry in the bottom head of the RPV, although the BWRVIA data predicts that oxidant and hydrogen peroxide concentrations may be lower. Both data sets indicate UK ABWR will be responsive to HWC in the bottom head region of the RPV. However, looking at the predicted ECP values in isolation could lead to the conclusion that the bottom head region of the RPV may not be “protected” from SCC. The calculated ECP values indicate the BWRVIA model is less responsive to HWC in the bottom head of the RPV. However, BWRVIA does not calculate ECP values plants which inject platinum, where the ECP is largely determined by platinum deposited on the surfaces. In effect, BWRVIA under predicts the expected ECP under these conditions. The BWRVIA results are therefore not directly applicable for the planned UK ABWR operating chemistry in this region of the plant. Putting this to one side, these data do support Hitachi-GE’s claim that the operating chemistry provides SCC mitigation in the bottom head of the RPV.
150. This further reinforces the need for representative ECP and platinum deposition physical monitoring locations within the UK ABWR design (considered in more detail in Section 4.3.5.3 of my report). Figure 7 also further highlights the importance of understanding how flow patterns around the RPV may impact the calculated ECPs and how the RIPs are modelled in Hitachi-GE’s radiolysis calculations. **This has been captured as part of Assessment Finding, AF-ABWR-RC-01, above.**

151. To summarise my overall conclusions regarding Hitachi-GE's approach to radiolysis and ECP modelling:
- Fundamentally, Hitachi-GE's radiolysis model is fit-for-purpose for a generic safety case. However, there are several very important aspects and uncertainties (captured as part of Assessment Finding, **AF-ABWR-RC-01**) associated with the current approach which the future licensee will need to address, to be able to move the current approach to radiolysis and ECP modelling forward to a more suitable position for an operating regime. In the context of SAP ECH.3, radiolysis and ECP modelling appears to be a very important part of demonstrating adequate control of the operating chemistry, but, for the purposes of a generic safety case, I am largely satisfied the expectations of this SAP and SAP ECH.1 have been met.
 - Radiolysis and ECP modelling and the HWC mini-test conducted at KK-7 indicate the ABWR will be responsive to HWC. These data support Hitachi-GE's specific claims regarding which regions of the UK ABWR, the operating chemistry will provide mitigation against SCC for.
152. From Ref. 47 the validation status of Hitachi-GE's radiolysis and ECP is unclear. Some of the uncertainties associated with Hitachi-GE's current approach emphasises the need to benchmark any radiolysis and ECP model used for UK ABWR, against plant chemistry data. Most US BWRs do this when using the BWRVIA model (Ref. 48). This includes using an extensive range of plant data, over a wide range of BWR designs, core designs and operating chemistry regimes. For GDA, therefore, I have been unable to judge the extent to which Hitachi-GE's radiolysis and ECP model satisfies the expectations of some of the relevant SAPs on validation of models, namely SAPs AV. 2 and AV. 3. Some benchmarking with KK-7 data has been performed, but this is just a starting point. It is clear a significant amount of work will need to be done by the licensee in this area. **This has been captured as part of Assessment Finding, AF-ABWR-RC-01, above.**
153. Ref. 32 also identifies a number of important constraints which could impact the effectiveness of OLNLC. These include factors relating to the coverage, distribution and size of the noble metal particles which become deposited on wetted RCS surfaces. Clearly, for OLNLC to be effective there are therefore constraints associated with each of these parameters. Ref. 32 identifies the timing of OLNLC during the fuel cycle and ability to monitor the deposition and distribution behaviour of platinum inside the reactor, once it has been injected, as being important "controls". Ref. 32 states OLNLC should not be conducted under the following conditions:
- Without feedwater hydrogen injection being in service. The basis for this requirement is that BWR plant experience has shown that without HWC, SCC cracks will continue to grow unmitigated. Also, the presence of HWC during OLNLC is beneficial for platinum deposition and durability because a more compact, tighter oxide is formed. Therefore, it is important to maintain hydrogen injection throughout the OLNLC application period.
 - Within the first 60 days of the fuel cycle. The basis for this requirement is essentially to allow "fresh" fuel added to the core during the previous outage to form a protective oxide layer and avoid direct contact of platinum on "fresh", un-oxidised fuel surfaces.
 - Within three months or less of a planned outage. The basis for this requirement is that BWR plant experience has shown conducting OLNLC too close to a planned outage can result in increased moisture carryover to the turbine plant, with higher outage rates caused by an increase in ⁶⁰Co transported with the moisture to the balance of plant.
 - If the reactor is running at <85% power. This requirement is also captured in Ref. 36.

154. These are all important constraints, consistent with plant OPEX and based on the experience of currently operating BWRs. While it is overwhelmingly positive that the generic safety case considers each of the above factors, what is less clear is how these inter-related measures and other parameters would all be “controlled” in practice. In RQ-ABWR-1496 (Ref. 49) I therefore asked Hitachi-GE to clarify the relationship between these measures and how the future operator would be expected to use Hitachi-GE’s radiolysis and ECP model to safely operate the plant. In response, part of the clarification provided by Hitachi-GE stated:
- The feedwater hydrogen demand for UK ABWR will be based on the outputs of radiolysis and ECP calculations performed using Hitachi-GE’s radiolysis model.
 - The feedwater hydrogen demand for UK ABWR will be calculated on a conservative basis, to achieve a hydrogen to oxidants molar ratio greater than three at the “most conservative location”. The “most conservative location” for UK ABWR is upstream of the downcomer.
 - The validity of Hitachi-GE’s radiolysis and ECP model will be verified using calculations performed using the EPRI’s BWRVIA computer code.
155. On this basis it appears Hitachi-GE’s approach will be very similar to the approach adopted by US BWRs which implement HWC. Hitachi-GE did not state the precise location of the “most conservative location” which they refer to. However, using BWRVIA terminology, it would appear to be in a similar location to the “S1 carry-under point”. Another similarity in Hitachi-GE’s approach is to provide some operating margin when calculating the effectiveness of HWC, although Hitachi-GE uses a molar ratio of hydrogen to oxidants greater than three, whereas plants using the BWRVIA model select a molar ratio of hydrogen to oxidants greater than four. During GDA, Hitachi-GE did not explain why a different molar ratio has been selected for UK ABWR. For example, whether there are differences in plant design between UK ABWR and other non-internally pumped BWRs which operate using HWC, which mean selecting a lower molar ratio for UK ABWR is justified. This does not undermine the overall generic safety case presented, but is an important area which will need to be addressed. **This has been captured as part of Assessment Finding, AF-ABWR-RC-01, above.**
156. Considering the controls required to safely implement HWC and OLNC, Hitachi-GE has identified a number of limits and conditions (or ORs) as part of the generic safety case. The identified ORs relating to HWC and OLNC are given in Annex 5 of my report. In summary, they include ORs on feedwater and reactor water hydrogen (and availability), reactor water platinum, ECP (including hydrogen to oxygen molar ratio), and also capture some of the restrictions on performing OLNC (as described previously). With the exception of the upper limits defined for feedwater hydrogen (related to radioactivity) and the amount of platinum injected (related fuel integrity), all of these ORs relate to the mitigation of materials degradation. In the majority of cases, I am satisfied Hitachi-GE has recognised the importance of maintaining adequate control over most of these parameters, and also with the good alignment the derived ORs have with the arguments and evidence provided in the generic safety case. However, there are some aspects of the ORs related to implementing HWC and OLNC I am not satisfied with, because:
- The minimum feedwater hydrogen “limit” is set based on the requirement of the related OR that the reactor water hydrogen to oxygen molar ratio is greater than two. This removes the need to have ECP as a control parameter, because the requirement to have the molar ratio in the reactor water greater than two means that any location where this condition is met will exhibit ECP <-230 mV (vs. SHE). This is standard practice for plants which run on HWC. However, it is not clear from Ref. 20 if the molar ratio itself is a control parameter, or whether it should be. Furthermore, it isn’t clear whether the basis of setting the molar ratio of hydrogen to oxygen is strictly correct. Normal practice for BWRs

appears to set this requirement based on the molar ratio of hydrogen to oxidants (i.e. oxygen, but also hydrogen peroxide).

- Based on the radiolysis modelling performed by my TSC, UK ABWR may be expected to operate at the lower end of the feedwater hydrogen range specified in the generic safety case. The lowest values calculated (0.17 ppm) remain slightly higher than Hitachi-GE's lower "limit" of 0.15 ppm. This difference is due to the Hitachi-GE value using a molar ratio of two and my TCS' calculations using a value of four (at the "S1 carry-under" point). This lower "limit" value may therefore change.
- The lower platinum "limit" of 20 g per year corresponds to the minimum amount required to achieve $0.05 \mu\text{g cm}^{-2}$ coverage of the RINs, assuming it is equally dispersed. I am not sure this OR is appropriate, firstly because it would be very difficult to develop adequate surveillance requirements for and secondly, because it seems very optimistic.
- The condition of feedwater hydrogen availability being >90 % does not appear appropriate (i.e. why is it acceptable at all for feedwater hydrogen injection to be unavailable during plant operations); considering the claims placed on the operating chemistry to mitigate SCC, the UK ABWR plant design (welded RINs) and Hitachi-GE's conclusion this reduces relevant risks SFAIRP.
- While the requirement not to conduct OLNC within a 60 day fuel conditioning period is recognised, the other important constraints identified by Ref. 32 are not; namely HWC in service, reactor power > 85%, and not within three months of a planned outage.

157. Overall, I am therefore not content that the ORs provided for HWC and OLNC in the generic safety case constitute a sufficient basis for the future licensee to develop their own set of operationally focussed ORs. They do not appear to enable the future operator to comprehensively demonstrate the UK ABWR operating chemistry can be adequately controlled within the requirements defined by the generic safety case. Considering the overall importance of defining adequate ORs, I expect a future licensee to address these gaps in the UK ABWR generic safety case. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-02: Defining and justifying adequate quantitative limits and conditions (Operating Rules) to execute the necessary control over a range of chemistry parameters important to safety is fundamental. During GDA, ONR's chemistry assessment established there are numerous gaps in the chemistry-based Operating Rules identified as the outputs from the UK ABWR generic safety case; including missing parameters and inappropriate values. To address these gaps, the licensee shall review and update the UK ABWR safety case to ensure an adequate set of chemistry-based Operating Rules are defined and justified.

Impact of Impurity Ingress on Materials Degradation

158. For BWRs, irrespective of the operating chemistry decision, the requirement to minimise impurities to negligible levels (and operate close to the theoretical conductivity of pure water) always remains. However, during plant operation, the reactor water accumulates impurities due to the ingress of soluble and insoluble metallic and non-metallic species, mainly through the CFDW system. In terms of materials degradation, the main concern is with soluble non-metallic species, such as chloride and sulphate. This is recognised by Hitachi-GE in the claims made in Chapter 23 of the PCSR (Ref. 11):

- "[RC SC1.2] *Corrosion (such as SCC, FAC) and corrosion rates will be mitigated by control of impurities by operation of the clean-up systems*".

159. *Topic Report on Impurity Ingress* (Ref. 30) presents information on potential impurity sources, risks to plant components, mitigation measures and OPEX of impurity ingress events in BWRs. This considers all potential risks from impurities, not only materials degradation. In general, any increase in impurities will increase the risk of materials degradation. The approach adopted in Ref. 30 to explain the impacts of this is reasonable, however the evidence presented is very much a summary. There is no detailed consideration of the specific impact of impurities under the operating chemistry conditions specified for UK ABWR. However, the report correctly identifies that chloride and sulphate are the most important impurities to minimise in the RCS. Such impurities can, if tolerated, significantly impact on the integrity of the RCS and associated systems due to the production of a corrosive environment which leads to degradation mechanisms such as SCC, pitting and crevice corrosion in austenitic SSs and NBAs, which comprise the RCS and associated SSCs.
160. With respect to SCC of materials in BWR operating environments, it is well documented that sulphate is particularly detrimental (Ref. 44 and 45). Sulphates have a pronounced impact on both crack initiation and growth rates in SSs, and can enhance the rates for Carbon Steel (CS) and low alloy steels (LAS). The increase is most pronounced under NWC conditions. Under HWC conditions the increase is not as significant for SSs, although the effect still occurs if the sulphate concentration is high and the ion is present in the acid form. It is not clear if the same benefit of HWC is seen for NBAs. This is consistent with the evidence presented in Ref. 30, and elsewhere in the generic safety case. The effect for SS is exemplified by Figure 8, below. The important conclusion is that maintaining low ECPs using HWC and OLCN does not remove the requirement to maintain low levels of sulphate in the reactor water, in particular for areas where the operating chemistry cannot provide SCC “protection”.

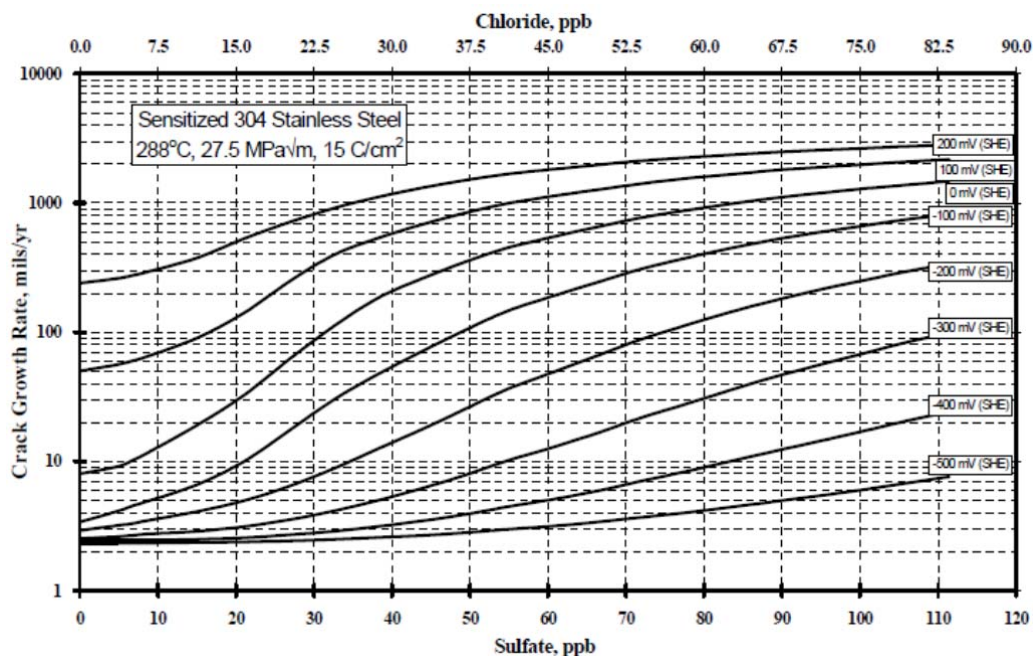


Figure 8: Predicted SCC crack growth rates as a function of sulphate or chloride ion concentration (Ref. 45 EPRI)

161. As also depicted in Figure 8, chloride has a similar detrimental impact on SCC rates. An important promoter of the impact of chloride is oxygen (or corrosion potential), which acts synergistically. With very low levels of oxygen (1 ppb), even extreme concentrations of chloride can be tolerated. While Figure 8 predicts a similar impact to sulphate, experimental testing concludes that the impact of chloride is in fact less (i.e.

- a higher concentration is required). However, chloride increases the risk of all types of corrosion, particularly pitting and/or crevice corrosion. As with sulphate, operating using HWC and OLNC reduces the sensitivity to chloride ingress events in areas of the plant where mitigation is provided, but this does not remove the need to maintain strict control over chloride.
162. It is also worth noting the impact of copper. Owing to the mechanism by which copper effects the recombination reaction of oxygen and hydrogen on SS surfaces, it can have a detrimental impact on the effectiveness of HWC and OLNC. The impact of copper on SCC mitigation is not discussed in Ref. 30, but Ref. 44 identifies that reactor water copper concentrations > 10 ppb are necessary before the effectiveness of SCC mitigation by HWC begins to be compromised. Copper also has detrimental impacts on fuel cladding, as discussed later in Section 4.3.4.1 of my report.
163. Other anions such as chromate and nitrate which are not stable under reducing conditions (i.e. HWC) have only a minimal effect on IGSCC. Cations such as sodium also appear to have minimal effect.
164. Given the overall importance of maintaining adequate control over certain impurities, Ref. 20 therefore identifies a number of ORs for UK ABWR related to the control of impurities in both feedwater and reactor water. ORs of direct relevance are given in Annex 5 of my report. In summary, they include ORs associated with direct conductivity, reactor water chloride and sulphate concentrations and feedwater copper concentrations.
165. Direct conductivity measurements provide early warning and protection in ensuring that corrosive species contained in the coolant water are minimised. For this reason, most operational BWRs include conductivity as a continuously monitored parameter, although as indicated for UK ABWR, this does not obviate the need to measure specific ions of significance (i.e. chloride and sulphate). With respect to the scope of the parameters related to impurities selected as the basis for the ORs for UK ABWR, I am generally content, because the parameters selected align with plant OPEX and the wider generic safety case. However, I am not content with the limit values suggested for chloride and sulphate in the generic safety case, because:
- In Ref. 20, the upper limit for chloride is set based on a single set of SCC experiments for SS. This does not consider the full impact of chloride, nor the range of environmental conditions expected for UK ABWR. Importantly, operating with 600 ppb chloride, even under HWC conditions, would seem to be within a regime where SCC can be expected. In effect, the mitigating effects of HWC have been removed. Notably this value is off-scale in Figure 8. This is also inconsistent with current industry water chemistry guidelines (Ref. 45).
 - The chloride upper limit is then used as the basis for setting the limit for sulphate, as it is assumed that they exhibit a similar dependence. Notwithstanding the above, this is inconsistent with experimental data and also seems to allow concentrations of sulphate to be present in a known region of high SCC susceptibility.
 - The direct conductivity limit is calculated based on operation with 600 ppb of chloride. Given this parameter is used as a general measure for water quality, no attempt is made to relate the value to any other impurities. Again, it is not clear that operating with such high conductivity values, irrespective of the impurity, would be acceptable.
166. I am not satisfied these ORs represent an adequate set of limits for impurities, within which the UK ABWR can be operated, to be able to demonstrate the requirements identified in the underlying generic safety case are achieved, to minimise the risk of materials degradation. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02, above.**

167. With respect to impurity control in feedwater, Hitachi-GE argues it is acceptable for the UK ABWR generic safety case not to provide limit values for other impurity control parameters in feedwater, because limit values are already set for reactor water. This general approach is taken for other BWRs (Ref. 44), but the important difference is that a suitable limit value for feedwater direct conductivity is specified. This is because adequate control of impurities in feedwater is the first line of defence preventing their ingress into the reactor. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

Impact of Feedwater Oxygen Injection on Materials Degradation

168. As described previously, the production of oxygen in a BWR is inevitable due to the radiolysis and boiling which occurs within the reactor core. Typical levels for a NWC plant are approximately 150 - 250 ppb DO in the reactor water. The adoption of HWC and OLNC and the minimisation of impurities, act synergistically to mitigate the impact of DO on SCC.
169. For sections of the RCS which comprise the CFDW system the main materials degradation threat is FAC. FAC is not relevant for the RINs or RPV because the materials selected are resistant, owing to their composition. The main parameters which influence FAC include: material composition, temperature, DO concentration, and the flow velocity of coolant through the system piping and components. pH is also a factor for FAC, but the UK ABWR chemistry regime does not include chemical addition for adjustment of the pH. Considering the inherent design of BWR plants, this is the approach that has to be adopted for all operational BWRs. The main parameters of relevance to chemistry which can therefore affect FAC are material composition and DO concentration. Some aspects of the materials selection for the CFDW system, and the purpose of the enhanced protection provided by oxygen injection into the CFDW system, to mitigate the risk of FAC, is justified by Hitachi-GE in, *Topic Report on Degradation Mechanisms – Flow-Accelerated Corrosion and Erosion-Corrosion* (Ref. 50). This part of my assessment therefore considers the claims made by Hitachi-GE on DO control in the CFDW system.
170. Chapter 23 of the PCSR (Ref. 11) recognises the significance of mitigating FAC control in the CFDW system by making a specific safety claim on the UK ABWR operating chemistry:
- “[RC SC2.1] *Corrosion such as FAC and pitting corrosion will be reduced by oxygen control*”.
171. For UK ABWR, the concentration of DO in the condensate water is expected to be typically less than 10 ppb during at-power operation, as it is efficiently removed by the vacuum de-aeration performed by the main condenser (Ref. 50). This low DO concentration, accompanied by a high coolant flow velocity, is what poses the risk of FAC. This is exacerbated by the operating temperatures within the CFDW system, which are around the values which lead to maximum FAC rates. The purpose of oxygen injection for UK ABWR is therefore to increase the DO concentration to a level which mitigates the risk of FAC, without causing other detriments. The fundamental requirement specified in Ref. 50 is that a feedwater DO concentration of ≥ 15 ppb is sufficient to provide FAC mitigation. In practice, the degree of FAC mitigation provided will vary with numerous other parameters. Hitachi-GE’s assumption is though, that all other factors being equal, an order of magnitude reduction in FAC rates will be achieved at DO concentrations ≥ 15 ppb.
172. In principal, the addition of oxygen to mitigate FAC is a common practice used by numerous nuclear and conventional power plants worldwide (for example, see Ref. 51). The strategy adopted by Hitachi-GE is therefore quite clearly reasonable. However, the rationale provided by Hitachi-GE to justify a DO concentration ≥ 15 ppb is

adequate to satisfy claim [RC SC2.1], is much less robust. Hitachi-GE bases the threshold value of 15 ppb, and the corresponding claimed reduction in observed FAC rates, on a single paper from 1972. No attempt is made to relate this important assumption to the conditions in UK ABWR. This is very important, because inevitably there will be variations in the extent of the FAC mitigation provided by the operating chemistry, across different locations within the UK ABWR plant design. This is why operating plants model such systems to identify SSCs and pipework sections susceptible to FAC. I also note:

- Other relevant information (for example Refs. 44 and 45) indicates a higher level of DO is required to provide assurances that FAC mitigation by the operating chemistry is achieved.
- All BWRs operating to the EPRI's BWR Chemistry Water Chemistry Guidelines (Ref. 50) do so with a minimum feedwater DO concentration >15 ppb.
- All Nordic BWRs operate with a minimum DO concentration of ≥ 40 ppb. This was in response to FAC discovered in CFDW system piping relatively recently, even when plant(s) were operating with a minimum feedwater DO concentration ≥ 30 ppb (Ref. 43).

173. The above OPEX represents the practices conducted at well over half of the world's currently operating BWR fleet. On this basis and the evidence provided by Hitachi-GE during GDA, I am not satisfied that operating at ≥ 15 ppb DO would provide assurances of effective mitigation for FAC. It is also important to note here that the minimum threshold value of DO ≥ 15 ppb is one of the main criteria Hitachi-GE use in their materials selection guidelines (Ref. 52) to assess the risk of FAC. This threshold is therefore a key deciding factor in whether or not more corrosion resistant materials are selected for UK ABWR SSCs. My assessment of the adequacy of this threshold value, in this context, is also considered in Section 4.3.2.3 of my report, during my assessment of Hitachi-GE's justification for the materials selected for the Heater Drain system.

174. Based on claim [RC SC2.1], and the associated evidence provided in the generic safety case, several ORs related to feedwater DO concentrations are identified by Hitachi-GE. They are listed in Annex 5 of my report. Considering the nature of the claims made on DO, it is important that Hitachi-GE has identified these as control parameters. However, as explained above, I am not satisfied with the lower "limit" value (15 ppb) defined for this parameter. Furthermore, for the upper limit value (500 ppb), Ref. 20 explains this is set based on pitting corrosion concerns for CS piping in the CFDW system. What does not appear to be considered is the impact of such high levels of feedwater DO on the reactor water; which would appear to be a more limiting concern. **This has been captured as part of Assessment Finding AF-ABWR-RC-02, above.**

Impact of DZO Injection on Materials Degradation

175. DZO injection involves the continuous addition of low levels of soluble zinc to the reactor coolant water. As described earlier, for both BWRs and PWRs, this is primarily a measure applied to mitigate radioactivity and worker ORE. However, sufficient quantities of zinc have been shown to have benefits in mitigating SCC in SS and NBAs.

176. *Topic Report on Zinc Injection* (Ref. 31) provides an overview of initial work performed to investigate the potential beneficial impact of DZO on SCC propagation rates. The overall conclusion from these data is that, while there may be some small benefits from performing DZO injection in particularly susceptible conditions, to achieve a significant general benefit in reducing SCC crack growth rates would require a much higher

concentration[§] of reactor water DZO than is required for dose rate reduction. This is consistent with PWR experience, and appears to apply irrespective of other water chemistry controls, such as NWC or HWC. In effect this confirms that the only practicable chemical mitigation for SCC in BWR environments is to reduce the ECP. Little work has been performed examining SCC initiation rates, but it is likely that the same trend is observed. Importantly, Ref. 31 also provides evidence that there is no detriment to materials degradation behaviour from DZO injection.

177. This is consistent with the claims made in Chapter 23 of the PCSR (Ref. 11). Claims [RC SC1] and [RC SC2] both relate to the non-detrimental effect of DZO injection on SSC integrity. Importantly, Hitachi-GE does not make any explicit claims regarding any benefits DZO injection may bring in mitigating SCC.

Impact of Commissioning Chemistry on Materials Degradation

178. As described in Section 4.3.1.3, during commissioning Hitachi-GE makes important claims regarding maintaining the operating chemistry within set limits, to ensure the risk of materials degradation is reduced SFAIRP, particularly in relation to SCC and FAC. For the materials and commissioning chemistry topic, during GDA my assessment therefore considered whether claims [RC SC1] and [RC SC2] are adequately justified. As previously explained (Section 4.3.1.3), *Topic Report on Commissioning Chemistry* (Ref. 38) identifies three specific phases of commissioning for UK ABWR: ≤ 100 °C without fuel, ≥ 100 °C without fuel and ≥ 100 °C with fuel.
179. For the first phase (≤ 100 °C without fuel) and second phases (≥ 100 °C without fuel), Hitachi-GE states the most important SSCs to consider are the RPV pressure boundary and RINs. The use of high purity, deoxygenated, demineralised water is specified for both phases. The water quality to be applied during this phase will be determined by the future licensee, but Hitachi-GE expects this to be within the requirements specified within the *Water Quality Specification* (Ref. 20).
180. Given the “new” at-power operating chemistry proposed for UK ABWR the important decision during commissioning, which respect to mitigating materials degradation, is when and how to apply, a low ECP environment. Considering this difference, in Ref. 38, Hitachi-GE examines further potential improvements that could be implemented for UK ABWR during phase one commissioning, including dosing hydrazine, trisodium phosphate, or utilising Low Temperature NobleChem™. While these options are not foreclosed by the UK ABWR generic design, Hitachi-GE considers their application to be unjustified at present given the uncertainties and potential detriments associated with them. I am content this is a reasonable position for GDA.
181. For phase two commissioning, the position is slightly more complicated. Firstly, because of the increased temperatures. Above 100 °C the risk of SCC increases considerably if water quality control is not adequately maintained. In response, one of the main differences for phase two is that, to minimise the concentration of DO, Hitachi-GE proposes to use nitrogen blanketing wherever possible. The second complication is the availability of clean-up systems to be able to meet the required water quality specifications, and also the availability of plant systems to inject chemicals. Clean-up capabilities in the form of the CUW system only become available during phase two of commissioning. Considering these complications and with the aim of achieving a low ECP environment in mind, Hitachi-GE also considers a range of options and technologies which may be able to meet this aim during phase two commissioning. Based on an examination of potential benefits, detriments and uncertainties associated with these options, Hitachi-GE’s overall conclusion is that

[§] This may require several tens or hundreds of ppb of DZO, or at least an order of magnitude increase.

- some may be beneficial, but further work is required by a future licensee, including seeking additional OPEX and potentially undertaking research and development.
182. While I consider this is a reasonable approach for this stage of the project and Ref. 38 to provide a useful starting point for a future licensee to begin any such further work, I do not agree with Hitachi-GE's current conclusion that the "ALARP option" for phase two commissioning is to use pure water, with the CUW system being available for clean-up. This is mainly because I do not agree with the assumptions and technical argument Hitachi-GE have put forward regarding the injection of platinum.
183. For phase three of commissioning (≥ 100 °C with fuel) the risk of materials degradation due to SCC of the RINs remains, but the first steam flows will also become established, meaning other parts of the RCS will be used for the first time. This brings with it additional degradation risks, notably FAC in the steam and CFDW systems. Positively, in response Hitachi-GE proposes to adopt oxygen injection into the feedwater as soon as practicable during phase three of commissioning. For SCC mitigation, the following measures are proposed:
- Use of high purity, deoxygenated, demineralised water. This also includes using nitrogen blanketing where possible to further minimise the DO concentration.
 - Perform de-aeration operations during nuclear heating to lower the DO concentration in the reactor water to <200 ppb.
 - Bringing the CUW system into service to remove impurities.
 - Implement HWC at 5-10% thermal power (with OLNC after 60 days of operation).
184. I regard the above as broadly being a positive approach adopted by Hitachi-GE as it demonstrates a graded approach, where risk mitigation is increased as the environmental conditions evolve and UK ABWR plant systems become available.
185. With respect to the requirement to perform de-aeration operations, Ref. 38 states, "*there is no set criteria value for the de-aeration operation*", and states the DO concentration of <200 ppb stated above, as being derived on the basis that this is a typical DO concentration for reactor water during at-power operations. Hitachi-GE also argue this specification has been achieved by operating ABWR plants without the need for more stringent controls being in place. In addition, for UK ABWR, Hitachi-GE argue that other improvements have been applied, including improved material choices and stress improvements countermeasures. Despite these improvements, I do not agree with Hitachi-GE's assertion that this very loose control is sufficient. In effect, phase three commissioning is the very first reactor start-up. My assessment in the next sub-section which covers the impact of start-up and shutdown chemistry on materials degradation, therefore presents my overall conclusion on this aspect of the generic safety case.
186. Similar to the point above regarding the control of de-aeration operations, the choice of when to bring HWC into service is also a matter where the requirements of commissioning and (first) start-up chemistry control overlap. I have therefore assessed Hitachi-GE's case regarding implementing HWC in the sub-section on start-up and shutdown chemistry, below. For the purposes of feeding into my overall conclusion on commissioning chemistry here though, I note Hitachi-GE's decision to implement HWC at 5-10% reactor thermal powers is consistent with US BWR good practice and represents an improvement over typical Japanese practice, which is to commence HWC at 100% reactor power.
187. Overall, with respect to the impact of materials degradation and chemistry control during commissioning, I consider Hitachi-GE has provided an adequate justification during GDA as part of the generic safety case. Hitachi-GE's work has indicated that a

range of reasonably practicable options are not foreclosed by the UK ABWR generic design, and has identified the relative importance of these options, by undertaking a considered and structured review. By its very nature, this is a topic where much greater development of the assumptions made by Hitachi-GE during GDA and important future decisions will need to be made by a future licensee, as knowledge in the field continues to emerge. This is all the more important given the “new” operating chemistry proposed for UK ABWR. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-03: Chemistry control options implemented during UK ABWR commissioning may have a significant impact on the longer-term safety performance of the plant, including the generation and transport of radioactivity, amongst other factors. No Boiling Water Reactor has ever commissioned for the operating chemistry proposed for UK ABWR, nor operated using this chemistry from the very first fuel cycle. In this regard, the UK ABWR generic safety case therefore presents several options which require further work, to explore their relative benefits and detriments, before a final decision can be made by the future licensee. The licensee shall review and update the relevant conclusions in the UK ABWR safety case, and supplement them using their own further work on this topic, to demonstrate chemistry control implemented during commissioning reduces relevant risks SFAIRP. In particular, this shall include a much more comprehensive justification for when to implement: hydrogen, noble metals and zinc injection.

Impact of Start-up and Shutdown Chemistry on Materials Degradation

188. Given the “new” at-power operating chemistry proposed for UK ABWR, the environmental conditions experienced during start-up and shutdown operations change dramatically. This has very important ramifications for materials degradation threats. For example, the changes in temperatures, redox conditions and likelihood of impurity ingress events during these phases of operation, have important consequences on the ability to maintain adequately low ECP values and therefore mitigate the risk of SCC. This is recognised by Hitachi-GE in Chapter 23 of the PCSR (Ref. 11) as part of claims [RC SC1] and [RC SC2], but more specifically in claim:
- “[RC SC1.3] *Stress Corrosion Cracking (SCC) propagation will be mitigated during startup by minimising the reactor water oxygen*”.
189. In addition, claims [RC SC1.1], [RC SC1.2], [RC SC2.1] and [RC SC2.2] (see Annex 4) also apply to start-up and shutdown operations. These are important claims and it is positive to see them feature at the very top level of Hitachi-GE’s hierarchy of safety case documentation.
190. *Topic Report on Start-up and Shutdown Chemistry* (Ref. 36) is the main safety case document that supports these claims. It reviews the current industry guidelines and relevant good practices for normal start-up and shutdown processes for BWR refuelling outages, or after a trip. It covers the RCS and provides a more detailed assessment of the options and conclusions from the output of Ref. 27, as well as providing supplementary information. Start-up and shutdown chemistry was considered by my TSC in Ref. 37, and I have also considered relevant industry approaches, such as Ref. 44, 45 and 53 as part of my assessment. This part of my assessment of start-up and shutdown chemistry is restricted to the materials degradation aspects. Radioactivity, fuel integrity and chemistry control systems assessed in Sections 4.3.3, 4.3.4 and 4.3.5, respectively.
191. In basic terms, the approach defined by Hitachi-GE for material degradation considerations in Ref. 36 is to maintain the at-power operating chemistry conditions for as long as possible into the shutdown and start-up sequences. Impurities are minimised as much as possible throughout, using the CUW system and CPS. For shutdown operations, this means that both the hydrogen and feedwater oxygen

injection systems are kept in service until the low feedwater flow trip occurs. For both of these systems the precise point at which this will occur is not yet determined, as this depends on detailed system design. However, Ref. 36 does indicate that for UK ABWR reducing conditions will be maintained to lower power levels (and hence lower temperatures and for less time) than both Japanese and US plants. I welcome this practice. I consider this to be more important for SCC mitigation, given the rates and significance of this degradation mechanism and the fact that FAC rates should be lower at shutdown. The decision on when to stop hydrogen injection will also relate to other controls during this phase, including impurity levels. I would expect a future licensee to justify the decision on when to cease hydrogen injection during a reactor shutdown as part of normal business, in the site-specific phase.

192. While not related to material degradation, but directly related to when hydrogen injection may be stopped (or started), it is notable that Ref. 36 states, "*It is concluded that residual risk for vulnerable areas [of flammable gases] is reduced by the whole process of plant design for the UK ABWR and specific chemistry measurements/controls are not required during shutdown*". I consider the impact of this claim on plant design as part of my assessment of radiolysis gases in Section 4.3.6. I also note that my TSC's independent review (Ref. 37) identified that other BWRs which run HWC place requirements on the need to monitor for potential flammable atmospheres, and therefore place limits on hydrogen concentrations which need to be achieved before the RPV head removal sequence can commence. I do not consider that the generic safety case provides a convincing set of arguments and evidence to clearly justify why similar controls to manage the potential for flammable atmospheres in UK ABWR are not required. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**
193. During the outage a number of operational practices are proposed for UK ABWR, all of which, Hitachi-GE claim, will result in the reduction of detrimental impurities in the reactor and feedwater:
- operation of the CUW system and Fuel Pool Cooling and Clean-up System (FPC);
 - non-regeneration of the Condensate Demineraliser (CD) ion exchange resins; and
 - storage of the CD resins in de-aerated water and draining before putting back in service.
194. I am satisfied with the above proposed operational practices and that their importance is given suitable visibility in the generic safety case.
195. During a start-up the intention is to reach normal at-power operating chemistry conditions as soon as practicable, thus minimising the time spent in a potentially vulnerable environment. This is a valid approach, consistent with industry experience and practice. To do this Hitachi-GE specify a number of operations, including explicitly for materials degradation:
- De-aeration of the condensate water prior to < 200 ppb prior to start-up.
 - Commencing hydrogen injection at 5-10% power; and
 - Commencing feedwater oxygen injection at 5-10% power.
196. The arguments and evidence provided in Ref. 36 to justify these choices are the same as those given in Ref. 38 for commissioning chemistry. My assessment in the previous section of my report is therefore also relevant here, in particular regarding the adequacy of the < 200 ppb DO limit and the choice regarding starting HWC at 5-10%

power, as opposed to implementing the practice of Early Hydrogen Water Chemistry (EHWC)**.

197. The former point was also specifically considered by my TSC as part of their review (Ref. 37). It is common practice for BWR reactor water quality specifications to be presented as two discrete sets, defined by a threshold temperature (normally around 100°C). This recognises the importance of the relationship between impurities, oxygen and operating temperature and the impact this has on the risk of SCC to structural materials. Presenting specifications in this way (with a condition on temperature) prevents the plant operator from proceeding to the next, higher temperature stage in the start-up sequence before the DO concentration has reached an acceptably low value. I therefore do not agree with Hitachi-GE's argument (with minimal supporting evidence) that operating with up to 200 ppb DO, at temperatures above 100 °C is adequate to reduce risks of SCC SFAIRP. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02, above.**
198. For hydrogen injection the position is slightly more complicated. A number of BWRs have implemented EHWC. Hitachi-GE's case for GDA is that the decision whether to implement EHWC is for a licensee to make and that injecting hydrogen between 5-10% reactor powers is the "ALARP option". The generic safety case argues it is not necessary to implement EHWC for each start-up because it is an emerging technology and the UK ABWR design includes other countermeasures to reduce the risk of SCC through appropriate materials choices and other stress relief techniques. This argument, however, neglects the increased stresses during start-up compared to at-power conditions, and the likelihood of higher levels of impurities. In addition, as part of the work on commissioning chemistry, the possibility of conducting EHWC is discounted on the basis that OLNK is restricted by a fuel conditioning period of > 60 days. This would only apply during phase three of commissioning (≥ 100 °C with fuel). It should not preclude prior platinum injection during phase two of commissioning.
199. Furthermore, Ref. 38 states the UK ABWR design does not foreclose the adoption of EHWC, and in fact the modifications required to implement it would be relatively minor. As EHWC is an emerging technology, it may become relevant good practice in the future. I would therefore expect a future licensee to review the decision and assumptions made in the generic safety case by Hitachi-GE regarding the applicability of EHWC to UK ABWR. Based on this, and a review of relevant plant OPEX, I would expect a licensee to decide whether implementing EHWC is necessary to ensure the UK ABWR operating chemistry continues to reduce relevant risks SFAIRP. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-04: The UK ABWR operating chemistry proposed for reactor start-up operations involves injecting hydrogen at 5-10% reactor power to mitigate the risk of stress corrosion cracking. During GDA, Hitachi-GE asserted injecting hydrogen at reactor powers lower than 5-10%, so called Early Hydrogen Water Chemistry (a further risk reduction measure), is a decision for a future licensee to make. The licensee shall review relevant plant operating experience and decide whether implementing Early Hydrogen Chemistry is necessary for UK ABWR, to ensure relevant risks are reduced SFAIRP.

200. To reflect the importance of executing sufficient chemistry control over certain parameters during start-up and shutdown to minimise and control materials degradation risks, several limits and conditions are identified as chemistry-based ORs in the generic safety case. These are listed in Annex 5 of my report and include:

** Early Hydrogen Water Chemistry (EHWC) refers to the practice of injecting sufficient hydrogen during reactor start-up at low reactor powers and temperatures, to achieve a hydrogen:oxidant molar ratio of at least two at wetted RCS surfaces where platinum has deposited.

reactor and feedwater direct conductivity, reactor water chloride and sulphate concentrations, reactor and feedwater DO and reactor and feedwater dissolved hydrogen. What is clear is that many of the ORs defined for start-up and shutdown are based heavily on their at-power equivalents. Many of the limit values are identical, only the expected values appear to be higher. My assessment of the at-power equivalent ORs is therefore equally applicable here too. More fundamentally, I am not convinced this approach is correct. It implies less strict control is required during these transient periods. Intuitively, I would expect stricter controls (or at least equivalent) to apply during start-up and shutdown. Furthermore, some of the more specific gaps I have identified in the ORs include:

- Several control parameters do not have associated limits.
- The expected values for chloride and sulphate are approximately an order of magnitude higher than other industry guidelines.
- There is no recognition of the significance of temperature in the requirements for impurities and DO.
- It is not clear how the requirement to determine DO in the reactor water is ≤ 200 ppb prior to start-up will be achieved.
- Several parameters will not be applied for the full period (notably hydrogen, DZO and oxygen) but the ORs do not appear to reflect this.
- It is not clear if other parameters need to be more explicitly considered, for example copper in the feedwater during start-up; and
- No ORs are given for the CFDW system during outages.

201. I am not satisfied that the relevant ORs defined by the generic safety case are adequate to enable claims [RC SC1] and [RC SC2] to be satisfied for start-up and shutdown operations. In addition, as previously mentioned, the requirement to control the potential for flammable atmospheres is not captured by the ORs derived. Collectively, these are important gaps in the generic safety case and will require work from a future licensee to resolve them. Nevertheless, I am content this does not undermine the generic plant design. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

202. Remaining on the topic of ORs, the *Water Quality Specification* (Ref. 20) specifically considers start-up and shutdown (coolant ≥ 100 °C, power $\leq 100\%$) and outage conditions (coolant ≤ 100 °C), separately. These definitions appear to be somewhat ambiguous. For example, Ref. 36 defines shutdown as, “*shutdown is the period after commencing the reducing the reactor thermal power and until the temperature of the coolant becomes less than 100 °C*”. It is not clear if this means that any operations with less than 100% power are covered by this mode (e.g. 95% power). It is notable that these definitions are different to those provided in other parts of the generic PCSR (Ref. 11). They also differ to other similar chemistry-based guidelines, for example Ref. 45. There needs to be certainty over the terminology used in the UK ABWR safety case, to ensure the necessary chemistry controls claimed are implemented by the future plant operator when they need to be. I expect the future licensee to review and update the definitions given for chemistry-based operating modes in the UK ABWR safety case, to apply a consistent set of terminology. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-05: During GDA, ONR’s assessment of the UK ABWR generic safety case across several topics, established there is an inconsistent approach to defining reactor operating modes. The chemistry-based operating modes defined are also inconsistent with equivalent international guidelines. This could lead to inadequate chemistry control being implemented by the future plant operator. The licensee shall review and update the definitions provided for chemistry-based operating modes in the UK ABWR safety case, and apply a coherent and consistent set of terminology, to ensure future plant operations will reduce relevant risks SFAIRP.

Mitigation of Materials Degradation by the Operating Chemistry Summary

203. In the context of materials degradation risks, the main change proposed for the UK ABWR operating chemistry compared to the reference plant is the adoption of HWC and OLNLC, to mitigate the risk of SCC. This is an important decision taken by Hitachi-GE and reflects the significance of this potential degradation mechanism and the impact the operating environment (chemistry) can have. This is reflected in both the quantity and quality of information provided in the generic safety case on these matters. In this regard, I am satisfied the generic safety case prepared by Hitachi-GE is a suitable starting point for a future licensee to be able to develop the case further. This is where the largest amount of work will be required moving forwards; creating a robust operations-based safety case. Given the change to the operating chemistry proposed for UK ABWR, this is perhaps not surprising. Nevertheless, I am satisfied an adequate justification for this change has been provided. Overall, considering my detailed assessment presented above for this sub-topic, in the context of a generic safety case, I am satisfied the expectations of the relevant SAPs (ECH.1, ECH.2, EMC.2 and EMC.3) have been met.
204. During my assessment of this sub-topic I have raised five Assessment Findings, **AF-ABWR-RC-01, -02, -03, -04 and -05** and no minor shortfalls. Perhaps the most significant Assessment Finding raised in this sub-topic is **AF-ABWR-RC-02**, which is concerned with the adequacy of chemistry-based ORs. There are important gaps in the chemistry-based ORs derived in the generic safety case which a future licensee will need to address. Although these gaps are important, the nature of the gaps, however, does not undermine the UK ABWR generic design.

4.3.2.2 Hitachi-GE's Approach to Justifying Materials Choices

205. As explained in the introduction to Section 4.3.2, the following section presents my assessment of the second of three sub-topics I have considered for my overall assessment of materials degradation for UK ABWR.
206. There are many lower-tier generic safety case documents referenced from the PCSR which provide the evidence to justify the basis for the material choices proposed for UK ABWR. Before describing my assessment of them, it is useful to understand the safety case approach and structure developed by Hitachi-GE in this regard. In response to RO-ABWR-0035 (Ref. 25) Hitachi-GE provided a suite of documentation to justify the materials selected for UK ABWR. This is shown schematically by Figure 9. Excluding the PCSR, their hierarchy leads to three tiers of documentation.

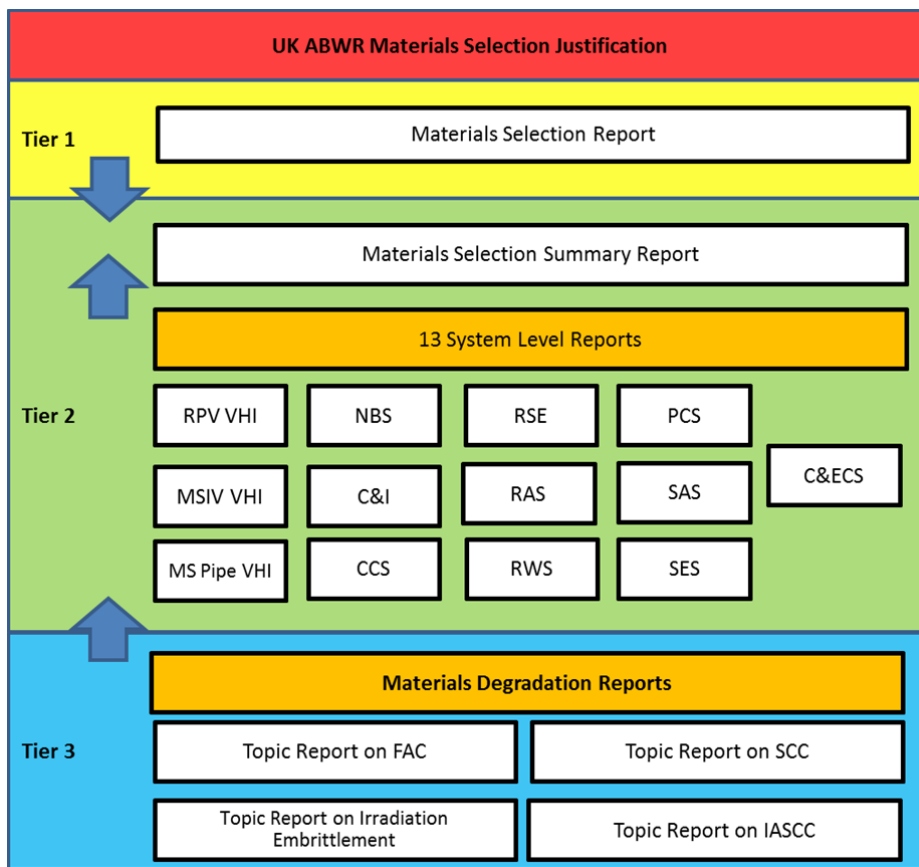


Figure 9: Overview of Hitachi-GE's submission structure to justify the materials selected for UK ABWR

207. The first tier comprises a single document, *The Materials Selection Report* (Ref. 52). This is essentially the head-document in the suite and defines the processes and guidelines by which materials selection decisions are made for UK ABWR SSCs in the balance of plant and nuclear steam supply system. Materials selection for fuel assemblies and control rods, and also for radwaste systems are excluded from the scope of Ref. 52. These are covered elsewhere in the generic safety case.
208. In Ref. 52, Hitachi-GE say materials are selected for UK ABWR to maintain the safety and reliability of SSCs throughout the design life of the plant. They say their materials selection process considers factors that impact the integrity of SSCs, worker dose and wider plant safety concerns. These requirements are then broken down into a series of initial factors, or requirements, to be considered when selecting materials. The stated approach is to, in general, base the materials selection for UK ABWR on materials used in existing plants where manufacturing issues are well understood and there is a significant amount of OPEX. The basic policy for UK ABWR is therefore:
- In principal to select CS.
 - If CS is shown to be unsuitable for the operating environment, consider using a more highly alloyed material.
 - Austenitic SSs are used for RINs and SSCs in contact with reactor water to minimise corrosion, radioactivity from corrosion products and the effects of radiation damage.
 - NBAs and precipitation-hardened SS are used where high strength is required.
209. Hitachi-GE's approach to materials selection essentially involves undertaking an initial general screening exercise to identify suitable materials for a particular application. Screening covers a number of matters, from design requirements, source term through

- to in-service degradation. This involves using a series of flow charts with relevant factors which need to be considered for materials in different operating environments. The output of following the flow chart through to completion is the selected material class.
210. Ref. 52 also provides specific guidance on the approach to be adopted for UK ABWR where the use of more corrosion resistant materials is required. For austenitic SSs, Hitachi-GE identifies a number of specific types (grades) of material that should be selected for UK ABWR, based on a consideration of the specific operational environment and materials requirements. For austenitic SS and NBAs, Ref. 52 also states the importance of selecting low cobalt material in applications where the SSC makes a particularly large contribution to cobalt release, and therefore can have a significant impact on worker doses. The guidelines also cover specific approaches where seawater is present and the application of titanium and copper alloys in the design.
211. The approaches described above are those adopted by Hitachi-GE to consider materials selection up to the point of justifying materials choices made for existing Japanese BWRs and J-ABWR. Ref. 52 then stipulates that having reached this point, to complete materials selection justifications for UK ABWR, an ALARP evaluation must be carried out. Section 4 of Ref. 52 explains Hitachi-GE's ALARP methodology and provides guidance on how materials selection ALARP justifications for UK ABWR should be conducted.
212. Considering the large number of SSCs which comprise the UK ABWR plant as a whole, Ref. 52 explains the need to group them together to make the ALARP demonstration for materials selection more manageable and proportionate. Ref. 52 describes how the grouping was carried out and its outcome. For GDA, the scope of the materials selection decisions covered for UK ABWR considers SSCs identified by Hitachi-GE as being "representative".
213. Looking at Figure 9, Hitachi-GE's second tier of documentation essentially comprises the outcome of applying the materials selection guidelines and ALARP methodology described in Ref. 52, to the identified "representative SSCs". The second tier comprises 14 documents in total – 13 "systems level reports" and a *Materials Selection Summary Report* (Ref. 54). This latter document is indeed nothing more than a summary. I have sample assessed a number of the "systems level reports" (Ref. 55) as part of my assessment of this sub-topic, which follows.
214. Tier three comprises four separate reports which address specific degradation mechanisms that need to be safely managed by the UK ABWR design. The submissions comprise TRs covering: FAC (Ref. 50), SCC (Ref. 56), Irradiation Assisted Stress Corrosion Cracking (IASCC) (Ref. 57) and Neutron Irradiation Embrittlement (Ref. 58). These TRs provide further evidence to supplement the more limited set of evidence Hitachi-GE use in their tier 2 documents as input information into their materials selection ALARP justifications.

Hitachi-GE's SSC Grouping

215. In this part of my assessment I consider the general approach adopted by Hitachi-GE for the grouping of SSCs. Ref. 52 provides an explanation of Hitachi-GE's approach to grouping UK ABWR SSCs together:
- Hitachi-GE identifies UK ABWR systems by their system code and screens all of them based on a multi-disciplinary approach, to ensure the widest possible scope of SSCs are initially considered for grouping. Ref. 52 states the need to balance a number of competing factors influenced by: structural integrity,

- chemistry, radiation protection and radwaste and decommissioning considerations, when making materials selection decisions.
- Once target systems have been identified and others screened out, individual target components within the systems screened in are then selected. The target components are then “categorised” into one of seven categories according to factors such as their type, product form or function.
 - Once target components are categorised, Hitachi-GE give further consideration to a number of conditions/factors for these components including, but not limited to: design conditions, degradation risks, OPEX, form and size, operating conditions etc. Where Hitachi-GE judges these factors to be similar for a number of components, they are grouped together on this basis. Hitachi-GE then selects the representative SSC for that particular group within the category based on the conditions they judge to be the most limiting for the group.

The outcome of Hitachi-GE’s grouping exercise is too large to reproduce here, but is presented in Ref. 52. Based on my assessment of Ref. 52 and the scope and approach of Hitachi-GE’s grouping methodology, as described above, I am satisfied for GDA that a suitable and sufficient set of “representative SSCs” has been identified. “Representative SSCs” identified by Hitachi-GE are much wider than only considering Very High Integrity (VHI) and Class 1 components; other relevant Class 3 components (e.g. CFDW system main piping) whose materials selection decisions also need to be adequately justified, from a chemistry perspective, are included.

Hitachi-GE’s Materials Selection ALARP Methodology

216. Hitachi-GE’s materials selection guidelines also include an ALARP methodology. This is based on the generic GDA ALARP guidance produced by Hitachi-GE (Ref. 21). In summary, the process includes 5 key steps which broadly cover: risk identification (i.e. the most relevant materials degradation threats), consideration of relevant good practice and baseline material selections, undertaking options scoring against relevant criteria and comparing the output of options scoring to identify the “ALARP material”. As part of the methodology, Hitachi-GE also makes some key assumptions:
- The UK ABWR operating chemistry is HWC, OLNC and DZO injection.
 - Where SCC is identified as a degradation threat, it is assumed polishing and peening is performed.
 - Weld metals are considered explicitly for some specific SSCs. In all other cases, weld metal selection is optimised based on the outcome of the ALARP assessment for the base metal.
217. Applying relevant parts of the guidance contained in ONR’s ALARP TAG (Ref. 4), I am satisfied Hitachi-GE’s process meets its intent. Ref. 52 clearly calls for a consideration of further measures, through a robust consideration of other potential options. Furthermore, I consider Hitachi-GE has identified a suitable range of criteria to compare options against one another. Based on information contained in Ref. 52, derivation of these criteria has clearly been informed by an appropriate consideration of a range of authoritative sources of relevant OPEX and guidance on materials degradation issues. From a chemistry perspective, selection of criterion such as “*release of detrimental material*” and “*radiological dose*” also provides confidence the scope of Hitachi-GE’s methodology is appropriate. Including these criteria ensures important issues such as the influence of materials selection decisions on the generation and transport of radioactivity are considered.
218. Overall, I am content Hitachi-GE’s materials selection methodology is fit-for-purpose for GDA. As materials justifications developed during GDA, I raised numerous RQs (RQ-ABWR-0743, -0744, -0747 and -1085 (Ref. 49)) asking about the presentation, scope and application of the materials selection ALARP process. Adequate information

to respond to all of my queries was eventually provided by Hitachi-GE in the relevant materials selection reports submitted during GDA.

219. The more fundamental consideration for my overall assessment of the adequacy of the materials selection process was to examine Hitachi-GE's application of it, to judge whether it generates robust and justified materials selection outcomes. This is the focus of the rest of my assessment of Hitachi-GE's responses to RO-ABWR-0035 (Ref. 25), presented below.

Carbon Steel – General Considerations

220. The basis of the materials selection philosophy for UK ABWR is essentially to use CS unless it is shown CS is unable to meet the material requirements for the applicable environmental conditions. This overarching philosophy means CS is selected either exclusively, or applied in large parts of a significant number of UK ABWR systems. The most significant degradation mechanism affecting CS in nuclear power plants, particularly in terms of safety, is FAC (Ref. 40). The other main degradation threat to CS is general corrosion, which leads to a loss of material thickness.
221. One of the main concerns with general corrosion is also the impact the released corrosion products (mainly iron for CS) have on the operating chemistry. For a BWR, this includes the importance of this impact on managing and controlling worker doses and fuel deposits. Ref. 52 appears to recognise this impact, specifically by considering "*release of detrimental material*" as a relevant criterion in materials choices. This shows, at a principles level, the scope of Hitachi-GE's materials selection guidelines are broader than just considering structural integrity based concerns, which is both necessary and positive.
222. In terms of FAC, Ref. 52 recognises the importance of this specific degradation threat to CS, by providing a specific flowchart and list of criteria to consider and assess the FAC threat for a particular system. Again, I consider this is positive and demonstrates this aspect of Hitachi-GE's approach to materials selection aligns with the available OPEX (Ref. 40).
223. For CS, whilst Ref. 52 demonstrates an appropriate understanding of the relevant degradation threats, I do not consider using the guidance contained in it alone, would result in adequate justifications, from a chemistry perspective, for the materials selected for several UK ABWR systems. This is because, consistent with the expectations of SAP EKP.1, the threat of FAC in systems fabricated from CS can largely be eliminated by design through appropriate materials selection (for example, by selecting LAS, or an equivalent material). This may then remove the need to manage and mitigate the effects of FAC by placing requirements on the need to implement systems to provide active chemistry control. The good practice of eliminating the degradation risk by appropriate materials selection at the design stage is what I would regard as the starting point for a materials selection ALARP justification. In certain circumstances, this may not be consistent with Hitachi-GE's starting point for materials selection for the plant.
224. I have therefore focused a large part of my assessment of UK ABWR systems where CS has been selected, with the aim of considering whether this is the most appropriate material to select for these systems on balance, especially considering the important change made to the operating chemistry specified for UK ABWR, compared to J-ABWR. A number of systems fabricated from CS aren't directly relevant to my assessment. Apart from the CUW system, those that are relevant (CFDW, HD and RHR systems), are covered by corresponding actions I raised on RO-ABWR-0035 (Ref. 25) during GDA. Hitachi-GE's extensive use of CS in all of these systems was the root cause for these actions being raised.

Low Alloy Steel – General Considerations

225. Ref. 52 specifies LAS for certain pipework and vessels in the steam system. The RPV itself is also fabricated from high strength grades of LAS. With respect to steam system components, the selection process is also provided in Ref. 52. Essentially, LAS has been selected where FAC is considered a threat to CS components. The specification of 1.25% chromium LAS for these SSCs means that the predominant degradation mechanism, FAC, is potentially mitigated, as such materials only experience very low rates of FAC. The use of LAS in steam system components is consistent with international operating experience. SSCs that have previously suffered FAC in BWR plants are specified in 1.25% chromium LAS for UK ABWR, which should negate similar failures. (Ref. 40).
226. Although general corrosion and associated loss of wall thickness could occur in LAS materials, Ref. 52 states that this will be accounted for in the design thickness of applicable pipework. I consider this approach to be in line with good practice (Ref. 40).
227. The main degradation threats for LAS in BWR plants are general corrosion, environmental fatigue and irradiation embrittlement (Ref. 40). Based on the nature of these mechanisms and therefore, SSCs where LAS is specified for use in UK ABWR; large parts of the plant where LAS is selected are beyond the scope of my chemistry assessment. For example, this applies to the RPV in particular. The RPV is a Very High Integrity component and is therefore covered in detail by ONR's structural integrity assessment report (Ref. 59).
228. For UK ABWR systems already fabricated from LAS which are within the scope of my assessment (mainly, steam system components, excluding the RPV), I consider the basis of the guidance contained in Ref. 52 to be consistent with good practice. This aspect of Hitachi-GE's materials selection demonstrates that measures to effectively "eliminate" the risk of FAC in these systems, by selecting appropriate materials, results in an inherently safer design. This is consistent with the expectations of SAP EKP.1. For these systems there is therefore no requirement to make any claims on the operating chemistry and/or to implement active chemistry control measures to mitigate the risk of FAC.

Austenitic Stainless Steels – General Considerations

229. Ref. 52 explicitly states austenitic SSs will be used for UK ABWR RINs and associated components that are in direct contact with water, in order to minimise general corrosion and the production of radioactive corrosion products. This is particularly important for the RINs and RPV inner cladding, which are almost exclusively SS. Hitachi-GE's materials selection guidelines (Ref. 52) also provide flow charts and tables for the selection of particular grades of SS depending on the expected environment and component purpose. This leads to the extensive use of SS throughout the UK ABWR design where corrosion would otherwise be a particular threat.
230. A review of the available OPEX (Ref. 40) shows the most significant degradation issue that has affected SS components in BWRs is IGSCC. The threat of IGSCC in BWRs has been tackled by implementing extensive materials and chemistry development programmes. As previously assessed in some detail in Section 4.3.2.1, based on the selection of HWC as part of the UK ABWR operating chemistry, it is clear Hitachi-GE are implementing good practice in their consideration of the influence the operating environment has on the potential for SCC in UK ABWR.
231. In terms of materials development, nuclear grades of Type 304 and 316 SSs have been developed. Low carbon variants of the main alloys, 304L and 316L, have been developed to limit the carbon available to form carbides with chromium at grain boundaries (the process which occurs during the sensitisation of SSs). In order to

compensate for the loss of strength associated with the reduction in carbon content, nitrogen strengthened grades such as 316LN (or 316NG – nuclear grade) have also been developed (Ref. 40). SSs can also be stabilised by adding niobium or other strong carbide-forming elements, such as titanium. Highly stabilised grades such as 347 and 321 are often employed in BWR RPV internals, such as the top guide and core plate. However, some examples of IGSCC have been observed in these grades, for example: where high weld residual stresses and irradiation have led to IASCC, or where a high carbon content and low stabilisation ratio have combined with weld residual stress to promote IGSCC (Ref. 40).

232. Hitachi-GE's materials selection guidelines appear to fully reflect these materials developments and the available OPEX regarding their performance. To address the issue of thermal sensitisation, the approach in Ref. 52 is to specify low carbon SSs with a carbon content of $\leq 0.020\%$ (i.e. Nuclear Grade) in applications where pipework or components require welding and during their operational lifetime, the pipework or component will be in contact with water at a temperature $>100^{\circ}\text{C}$. In addition, where assurances regarding tensile strength are also required Hitachi-GE's approach is to select SSs with nitrogen addition i.e. Type 304(NG) and Type 316(NG). On this basis I am satisfied Hitachi-GE has selected modern, good practice grades of austenitic SS for UK ABWR (Ref. 40).
233. Ref. 52 also recognises the importance of controlling some of the other factors which influence SCC susceptibility, including cold work. Detailed assessment of the adequacy of these measures is beyond the scope of my chemistry assessment, however, for the purposes of my assessment I have noted here that Hitachi-GE are implementing other good practice measures to reduce the SCC susceptibility of UK ABWR SS components.
234. Overall, I consider Hitachi-GE is taking necessary and practicable measures to positively influence all of the above factors, to reduce the overall SCC susceptibility of UK ABWR SS components. Positively, I also note Ref. 52 specifically refers to the importance of selecting low cobalt material in applications where the SSC makes a particularly large contribution to cobalt release, and therefore can have a significant impact on worker doses.
235. From a chemistry perspective, I therefore accept the high-level principles of Hitachi-GE's case for the need to use austenitic SS alloys in UK ABWR. I consider their materials selection practices and associated control procedures are consistent with what is regarded as good practice for BWRs (Ref. 40). I also judge the expectations of SAPs ECH.1, EAD.1 and EAD.2 have been met in this context. I have sample assessed some aspects of Hitachi-GE's justification for austenitic SS use for specific SSCs in my more detailed assessment, which follows.

Nickel Base Alloys – General Considerations

236. Although the use of NBAs in UK ABWR is much more limited compared to other material classes, they are applied in nuclear safety critical locations both within the RPV and as part of the RPV assembly itself. The main component which uses NBA is the shroud support, but lower sections of the RPV are also clad in this material. OPEX (Ref. 40) shows the main degradation threat to consider when using NBAs in BWRs is IGSCC. Some types and grades of the material are particularly susceptible.
237. Similar to SSs, reducing the carbon content and adding carbide forming elements such as niobium, titanium and/or tantalum also helps prevent sensitisation and limit the propensity for IGSCC in NBAs. Based on these developments, the Alloy 600 family of materials continues to be used in BWRs, with controls in place to mitigate degradation threats. This includes minimising cold work in a similar manner to SSs (Ref. 52). The

evidence suggests that only limited benefit can be achieved through chemistry control alone.

238. Considering the material choice alone, Hitachi-GE's materials selection guidelines (Ref. 52) appear to be consistent with the balance of international OPEX, in that they identify SCC as the most important degradation issue for NBAs. Furthermore, it is clear that by selecting the niobium-modified Alloy 600: N-bar ≥ 12 version of the material (Ref. 52), Hitachi-GE has selected a good practice, modern specification for Alloy 600.
239. Moving to the associated weld metals for Alloy 600 base metal, Alloy 182 and 82, the OPEX (Ref. 40) clearly shows Alloy 182 has not performed well in BWR environments. Alloy 182 has experienced a number of incidences of IGSCC, including the discovery of over 300 SCC cracks in shroud support welds at a Japanese BWR in 1999, and numerous SCC cracks have been observed in these welds at other BWRs (Refs. 40). SCC of Alloy 182 is a known issue under oxidising conditions. The performance of these weldments is expected to be improved under HWC conditions. Alloy 82 has a higher chromium content than Alloy 182 and has performed better with respect to IGSCC. Ref. 52 noted that no Alloy 82 in-service failures had occurred to date (2011). Like the case for Alloy 600 base metal, Ref. 52 is consistent with the OPEX. Alloy 182 will not be used by Hitachi-GE in UK ABWR. The baseline choice of weld metal for NBA in the design is niobium-controlled Alloy 82: N-bar ≥ 12 .
240. Alloy X-750 is another type of NBA used in BWR applications, particularly more widely in older BWRs. OPEX shows this alloy can be susceptible to IGSCC under BWR conditions, especially when specified in the equalised and aged temper (Ref. 40). In RQ-ABWR-1324 (Ref. 49) I asked Hitachi-GE to clarify whether and where Alloy X-750 is used in UK ABWR and the approach taken to specifying heat treatment conditions. Hitachi-GE's response explained several UK ABWR components are fabricated from Alloy X-750. A description of typical heat treatment conditions that may be specified was also provided, which specifically stated specifying Alloy X-750 in the equalised and aged temper should be avoided. On this basis, I am content with the position for GDA and judge that the process of controlling and specifying particular heat treatments should be progressed under normal business by a future licensee.
241. Overall, from a chemistry perspective, I accept the high-level principles of Hitachi-GE's case for the need to use the NBA class of materials in UK ABWR. I consider Hitachi-GE's materials selection practices and the majority of their associated control procedures are consistent with what is regarded as good practice for BWRs (Ref. 40). I also judge the expectations of SAPs ECH.1, EAD.1 and EAD.2 have been met in this context. I have sampled Hitachi-GE's justification for using NBAs for specific SSCs in my more detailed assessment, which follows.

Summary for Hitachi-GE's Approach to Justifying UK ABWR Material Choices

242. During GDA Hitachi-GE developed a comprehensive suite of documentation to justify the material choices for UK ABWR. This was of particular importance given the change in operating chemistry for UK ABWR. The net result is that it is clear from the generic safety case what the basis for the material choices are, what risks are being addressed and how the various mitigations, including chemistry, have been considered.

4.3.2.3 Assessment of Residual Materials Degradation Risks for UK ABWR SSCs

243. In this part of my assessment I consider whether the material degradation risks for a sample of "representative SSCs" have been reduced SFAIRP, based on the combination of the operating chemistry and material choices. My sample assessment focussed on SSCs where one, or all, of the following criterion apply: operating BWRs have historically experienced materials degradation, Hitachi-GE has identified changes

for UK ABWR, and the safety case claims on the operating chemistry’s ability to mitigate the risk of materials degradation are significant.

- 244. To justify the likelihood and consequences of materials degradation threats for UK ABWR SSCs, Hitachi-GE perform several risk ranking exercises for several materials degradation threats. My assessment in this sub-section is therefore structured accordingly. Firstly, I consider the adequacy of Hitachi-GE’s approach to the risk ranking performed for UK ABWR SSCs. Secondly, based on the criteria given above; I assess Hitachi-GE’s materials selection justifications for a selection of the risk ranked SSCs.

Risk Ranking of UK ABWR SSCs by Material Degradation Risks

- 245. Hitachi-GE’s approach to UK ABWR SSC risk ranking is described within four “*materials degradation reports*”, each of which covers one of the following degradation threats: FAC (Ref. 50), SCC (Ref. 56), IASCC (Ref. 57) and neutron irradiation embrittlement (Ref. 58). These are the “*tier three*” reports shown in Figure 9. Considering claims [RC SC1.3] and [RC SC2.1], the two most relevant degradation threats to consider are FAC and SCC.
- 246. To support my assessment of this topic, I also asked my TSC to review Hitachi-GE’s approach based on their knowledge of BWR materials degradation and chemistry issues. The output from their review is reported in Ref. 41 and 42. My TSC’s independent work has informed my assessment that follows.
- 247. In summary, Ref. 50 and 56 – 58 are more detailed materials degradation risk assessments for the UK ABWR. For FAC, SCC and IASCC the approaches taken in each of the separate reports are essentially the same. A risk ranking for UK ABWR SSCs is undertaken to be able to identify and then “quantify” the level of residual risk remaining in the UK ABWR design for a given degradation threat; after specific, relevant countermeasures to reduce the risk of that particular degradation mechanism occurring, have been applied. For a given SSC the approach is to define the “risk” posed by a particular degradation threat by semi-quantitatively evaluating the degradation susceptibility and the safety significance for the subject SSC, separately. The outcomes of undertaking these two independent exercises are then multiplied together to estimate an overall risk rank, from 1-4, for the particular degradation mechanism and SSC under study. This is illustrated by the matrix presented in Figure 10 below.

		Degradation susceptibility Rank			
		High	Medium	Low	Very low
Safety significance	Class 1	Rank 1	Rank 2	Rank 3	Rank 4
	Class 2				
	Class 3				

Figure 10: Degradation risk ranking matrix used in Hitachi-GE’s risk assessment methodology (Ref. 50)

- 248. According to Hitachi-GE’s definition, regardless of the degradation mechanism under consideration, risk ranks 1-4 are described as follows:

- SSCs identified as “Rank 1” represent an unacceptable level of residual risk and the risk of degradation would result in a “significant probability” that the SSC could not deliver its safety function. The UK ABWR generic safety case does not provide any guidance on the appropriate way to manage the residual risk arising from SSCs identified as “Rank 1”.
 - “Rank 2” is defined as having a “low probability” of the SSC not being able to deliver its claimed safety function.
 - “Rank 3”, a “very low probability”. Hitachi-GE says the appropriate management strategy for SSCs in “Rank 2” and “Rank 3” is to manage the risk of degradation by inspection. Hitachi-GE says the inspection frequency for “Rank 3” SSCs should be lower and reduced in scope, compared to the inspection requirements for “Rank 2” SSCs.
 - “Rank 4” SSCs are defined as having no identified credible materials degradation threat that could lead to loss of the safety function.
249. Hitachi-GE’s overall risk ranking process is a relatively straight forward procedure that has many similarities to existing methodologies which are commonly used by the nuclear industry for assessing the risk of materials degradation in BWRs. Hitachi-GE’s process is not identical to these procedures, but it is built on very similar principles and therefore well proven procedures for ranking the residual risk of degradation for different components and parts of systems (Ref. 41 and 42). In this regard, I am satisfied the process is appropriate and fit-for-purpose for a generic safety case.
250. The important part of the risk ranking methodology is how the “*degradation susceptibility rank*” (shown across the top of the matrix in Figure 10) for a particular SSC is determined. This is a convoluted process using flowcharts and tables, but the output is essentially a degradation (corrosion) rate, which is then compared against a set of pre-defined degradation (corrosion) rates for each rank. In effect these corrosion rates are simply an aggregate numerical value, and do not necessarily represent an actual corrosion rate that may be experienced in operation; they are simply a means for comparison. I queried some aspects of this approach in RQ-ABWR-1461 and -1465 (Ref. 49) and also requested my TSC to undertake an independent review of this particular aspect of Hitachi-GE’s methodology. My TSC did not raise any significant objections (Ref. 41 and 42) and I was satisfied with Hitachi-GE’s responses to my RQs.
251. Having assessed Hitachi-GE’s overall risk ranking methodology, the following two sub-sections present my detailed assessment of Hitachi-GE’s specific application of the methodology to the degradation threats of FAC and SCC, in Ref. 50 and Ref. 56, respectively.

Topic Report on Flow Accelerated Corrosion

252. Ref. 50 contains Hitachi-GE’s risk assessment for FAC and erosion-corrosion (EC). The main conclusions of the report are:
- No SSCs were identified as “Rank 1”.
 - Only one SSC was identified as “Rank 2”. Part of the CS piping in the Turbine Gland Steam System.
 - Several were identified as “Rank 3”. The majority of these were CS piping segments in the HD system and the remainder were CS SSCs either in the CUW or CFDW systems; and
 - The majority of SSCs were identified as “Rank 4”.
253. While the title of Ref. 50 suggests the report covers the degradation mechanisms of FAC and EC, the main focus is FAC. There is no corresponding risk ranking for EC. During GDA I raised RQ-ABWR-1479 (Ref. 49) to seek clarification on how this specific degradation threat is addressed by Hitachi-GE. The response provides

arguments for why FAC is bounding for EC and explains that based on this assumption, Ref. 50 uses the terms FAC and EC interchangeably. This means the FAC risk ranking for UK ABWR is in fact the EC risk ranking, and vice versa. Based on this explanation, I am not entirely satisfied Hitachi-GE's approach to FAC and EC risk ranking for UK ABWR SSCs is entirely reasonable. This is because although the degradation threats of FAC and EC are mechanistically similar and may occur in similar environments, they are not the same. EC is very often erroneously combined with FAC (Ref. 40). This may lead to a similar risk ranking for FAC and EC, but their different mechanisms may give rise to important differences in the risk locations identified for the plant for each different degradation threat. As further work will be undertaken by a future licensee to develop detailed inspection and monitoring programmes for FAC and EC, I am satisfied with Hitachi-GE's assumption for GDA, however, **I consider this to be a Minor Shortfall:**

MS-ABWR-RC-03: The UK ABWR generic safety case makes the assumption that materials degradation risks for flow accelerated corrosion and erosion-corrosion are equivalent. Based on this assumption, no separate risk ranking for erosion-corrosion is performed. These degradation mechanisms are similar, but not the same. The licensee should revise and update the UK ABWR safety case to include a separate risk ranking, or similar risk assessment, for erosion-corrosion.

254. In Ref. 50, Hitachi-GE's FAC risk ranking methodology also relies on an assumption made about the threshold DO concentration which provides mitigation for FAC in CS piping. This value is the same concentration (15 ppb) I have previously introduced and assessed in Section 4.3.2.1 of my report. There, I have already concluded 15 ppb is an inappropriate threshold value to select. During my assessment of Ref. 50, I therefore asked Hitachi-GE (RQ-ABWR-1327 (Ref. 49)) whether they were able to provide any additional plant-based evidence (i.e. the results of any FAC inspection and monitoring programmes performed for the J-ABWR reference plant), to support their assertions that 15 ppb is an appropriate threshold value to claim the operating chemistry mitigates FAC. During GDA Hitachi-GE was unable to provide this information. This would have been a robust source of evidence to validate the outcome of Hitachi-GE's FAC risk ranking exercise. Given the safety case presented during GDA is at the generic stage, I am satisfied with the evidence (in the form of the FAC risk ranking) presented for this phase of the project. However, **I consider this to be a Minor Shortfall which a future licensee should address:**

MS-ABWR-RC-04: The UK ABWR generic safety case does not utilise any operating experience from the J-ABWR reference plant, as evidence to support the key safety case claims made on the operating chemistry's ability to mitigate the risk of flow accelerated corrosion, nor in support of the corresponding risk assessment. The licensee should revise and update the UK ABWR safety case to draw upon the findings of any flow accelerated corrosion inspection and monitoring programmes performed for the J-ABWR reference plant.

255. During my assessment of Ref. 50 I also asked (RQ-ABWR-1465 (Ref. 49)) Hitachi-GE to undertake a sensitivity analysis to examine the impact of selecting a DO threshold of 15 ppb, on the outcome of the FAC risk ranking. In response, Hitachi-GE reconsidered their FAC risk ranking assuming the DO threshold of 30 ppb. As already explained in Section 4.3.2.1 of my report, this value is consistent with the EPRI's guidelines (Ref. 45). This analysis showed four regions in the HD system were at an increased risk of FAC, and their respective risk rankings were elevated from "3" to "2". During GDA, Hitachi-GE stated a higher DO threshold could have resulted in a different material being selected for these four locations. Ref. 50 does not reflect the outcome of this work and risk ranks of "3" have been retained for these regions of the HD system. Furthermore, the DO threshold value of 15 ppb is also retained. Given the UK ABWR SSC which appears most sensitive to this assumption is the HD system, my overall

conclusion on the acceptability of this for GDA is presented in the corresponding part of Section 4.3.2.3 of my report, below, where I assess certain aspects of Hitachi-GE's materials selection justification for the HD system in much more detail.

256. As well as the assumptions made on DO, other work presented in Ref. 50 directly relevant to my chemistry assessment concerns Hitachi-GE's justification for suitable materials selection and compositions (i.e. the chromium content), to mitigate the risk of FAC. For UK ABWR, this is most relevant for the CFDW system. I have therefore assessed these aspects of Ref. 50 in the corresponding sub-section of my report, below, which assesses Hitachi-GE's materials selection justification for the CFDW system.
257. Overall, based on my assessment of Ref. 50 presented above, and as supported by the independent work of my TSC, I am broadly satisfied Hitachi-GE's FAC/EC risk ranking is fit-for-purpose for a generic safety case. I have identified two Minor Shortfalls, **MS-ABWR-RC-03 and -04**, which I expect a future licensee to give further consideration to.

Topic Report on Stress Corrosion Cracking

258. In a similar manner to Ref. 50, Ref. 56 provides Hitachi-GE's risk assessment for SCC for UK ABWR. Consistent with the approach taken for FAC, the SCC risk ranking undertaken for UK ABWR SSCs is done so on a grouped basis. I consider Hitachi-GE's process for grouping UK ABWR SSCs is appropriate and fit-for-purpose. The process involves considering a wide range of authoritative guidance on BWR materials and component degradation, a suitable range of OPEX, specific design conditions for SSCs and certain key threshold conditions required for SSC to occur. Unlike Ref. 50, from a chemistry perspective, I consider the threshold conditions Hitachi-GE have identified for SCC appropriate. This is consistent with the independent views of my TSC (Ref. 41).
259. I note that the SCC risk ranking is only performed for SSCs made of austenitic SS and NBAs and therefore excludes LAS and CS. Based on the mechanistic understanding of SCC, I consider this to be an acceptable simplification to make for GDA (Ref. 41), although I note the risk of SCC in these materials, especially LAS, would not necessarily fall under "rank four". I would expect a future licensee to complete the risk evaluation and counter measures necessary for LAS as part of normal business, as they develop their detailed inspection and monitoring programmes for SCC.
260. The main focus of my assessment of Ref. 56 was the assumptions Hitachi-GE use to assess the impact of the operating chemistry on the overall risk of SCC for UK ABWR. In summary, Hitachi-GE assumes either NWC or HWC conditions (i.e. in ECP terms either ≥ -230 or ≤ -230 mV (vs. SHE), respectively). As I have concluded in Section 4.3.2.1, above, this ECP threshold value is reasonable. The distinction Hitachi-GE use in their SCC risk ranking is to assume a 10-fold decrease in SCC crack growth rates for HWC. While defining such precise differences is difficult, I consider this is an acceptable simplification for the purposes of the SCC risk ranking performed by Hitachi-GE. This conclusion was also independently supported by my TSC's work (Ref. 41). To be able to determine when to use the NWC or HWC SCC crack growth rates in the risk ranking exercise, Ref. 56 makes direct use of the outputs from Hitachi-GE's radiolysis and ECP modelling (Ref. 47). I have already previously assessed (Section 4.3.2.1) the adequacy of Hitachi-GE's radiolysis and ECP modelling and concluded the approach is fit-for-purpose for a generic safety case.
261. In addition, there are also two other important assumptions Hitachi-GE make in their approach to SCC risk ranking which suggests their approach is appropriate and suitably conservative:

- No claim is made on the potential benefits of HWC on SCC initiation, only crack growth rates.
 - For SSCs where stress improvement countermeasures are applied, any associated improvements in resistance to SCC initiation are not credited.
262. Based on the application of this approach, Ref. 56 shows no UK ABWR SSCs are identified as “Rank 1”, however there are a significant number identified as “Rank 2”, several as “Rank 3” and the remainder are “rank four”. Compared to Ref. 50, the outcome shows the residual risk of SCC for UK ABWR is higher than FAC. This isn’t particularly surprising given the risk of FAC can effectively be “eliminated” by selecting materials which are “immune” to this degradation threat. The prevalence of SCC in nuclear power plants has therefore been much reduced, but it can, and still does occur under certain specific conditions. This fact appears to be reflected in the output of Hitachi-GE’s SCC risk ranking.
263. Although there are a significant number of SCC risk “Rank 2” and “Rank 3” SSCs identified for UK ABWR, from a chemistry perspective, I am satisfied that in the majority of cases Hitachi-GE has demonstrated during GDA that all reasonably practicable measures to mitigate the risk of SCC in UK ABWR, have been taken. This includes the consideration of materials selection, stress improvement techniques and environmental countermeasures. As informed by my TSC’s review of Ref. 56 (Ref. 41), the residual risk of SCC posed to UK ABWR SSCs is consistent with the outcome that would be achieved for most other BWRs, and in many cases Hitachi-GE has made additional improvements to reduce the risk of SCC further. This leads me to conclude that I am satisfied with Hitachi-GE’s claim that the most appropriate way to manage the residual risk of SCC for UK ABWR is by developing and implementing suitable SCC inspection and monitoring programmes, as already undertaken by existing BWR plants. This is a matter of normal business for a future licensee to progress.
264. The one very important exception to my above conclusion is the impact of Hitachi-GE’s OR (presented in Annex 5 at the end of my report) concerning feedwater hydrogen availability (>90%) on the risk of SCC. I have already assessed the adequacy of this OR (section 4.3.1.2.1) and judged it is not appropriate. In Ref. 56, Hitachi-GE consider the impact of this OR by undertaking a sensitivity analysis to evaluate what impact the permitted operation of running UK ABWR without HWC has on the risk of SCC. The outcome is that some SSCs see an increase in their SCC risk ranking. One of the most important and relevant aspects for chemistry, is the impact this has on SS sections of Class 1 piping in the CUW and RHR systems. This is because the operating chemistry is the only SCC mitigation measure claimed for this part of the plant; no other SCC risk reduction measures are identified in the generic safety case for these parts of the plant (RQ-ABWR-1416 and 1508 (Ref. 49)). Furthermore, mitigation of SCC in CUW system SS piping is one of the two specific claims made by Hitachi-GE for where the operating chemistry *needs* to be the most effective (RQ-ABWR-1324 (Ref. 49)). This is one of the main contributory factors supporting my conclusion in Section 4.3.2.1 that I do not consider this is an appropriate OR for UK ABWR. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**
265. This concludes my assessment of Hitachi-GE’s risk ranking for UK ABWR “representative SSCs”. The following sub-sections of Section 4.3.2.3 present my detailed assessment of Hitachi-GE’s materials selections for several of the ranked “representative SSCs” which from a chemistry perspective, are most significant.

RINs - Core Shroud, Shroud Support and CRD Stub Tubes

266. Based on past experience, the core shroud is one of the BWR components in particular that has suffered the most from materials degradation threats, predominately SCC. Hitachi-GE also claim the UK ABWR operating chemistry will mitigate SCC in the core shroud. The core shroud in a BWR surrounds the core and provides support to

the top guide and core plate, to ensure fuel assemblies are kept in a horizontal position. The shroud also separates the different coolant flow paths in the RPV. Considering the functions the core shroud provides, Hitachi-GE has identified it as a Class 1 component. They have selected Type 316L (NG) SS forging, including the specification of low-cobalt ($\leq 0.05\%$) material for the core shroud. Ref. 55 also states that further measures to reduce SCC susceptibility, such as polishing or peening, will be implemented around core shroud welds. Based on this materials selection, the additional countermeasures Hitachi-GE are implementing to tackle the potential for SCC, and my assessment of the outputs of Hitachi-GE's radiolysis and ECP modelling (Section 4.3.2.1), I consider Hitachi-GE has adequately justified the materials selected for the UK ABWR core shroud.

267. As well as the core shroud itself, the shroud support is also a very important component because it takes the weight of many of the RINs, including not only the shroud, but the top guide and steam separator. The shroud support has therefore been identified as a Class 1 component by Hitachi-GE. As well as the shroud support, the CRDs are also located in the bottom head region of the RPV. The control rods are inserted from the bottom in a BWR; therefore UK ABWR has a correspondingly complex geometry in this part the RPV. To reduce the influence of local stresses in this part of the plant, CRD stub tubes are used to separate the CRD housing welds and the RPV bottom head. This means the CRD stub tubes form part of the pressure boundary, and they are therefore also identified as a Class 1 component by Hitachi-GE.
268. In terms of wrought materials selection, Hitachi-GE's overall decision for the shroud support and CRD stub tubes is to select N-bar controlled Alloy 600, for both. N-bar controlled Alloy 82 has been selected as the weld metal (Ref. 55). Hitachi-GE's overall case is to select these materials in combination with implementing stress improvement techniques and claiming the operating chemistry mitigates SCC in this part of the plant.
269. Considering the complexity of the geometry in the bottom head region of the RPV, in tandem with Hitachi-GE's specific claims on the operating chemistry's ability to mitigate SCC in this location, it is important to ensure the potential for crevices has been eliminated in the design. This avoids the potential for "aggressive" local chemistry conditions to develop. During GDA, I therefore asked Hitachi-GE (RQ-ABWR-1324 (Ref. 49)) to explain the measures taken in the UK ABWR design to avoid/minimise crevices in the bottom head region of the RPV. Using a report produced by the EPRI (Ref. 45), Hitachi-GE provided technical information to demonstrate crevices in the bottom head region of the UK ABWR RPV have been eliminated by design. On this basis, I accept the arguments presented by Hitachi-GE. Building on this conclusion, based on my earlier conclusions in Section 4.3.2.2, where I assess the general considerations for Hitachi-GE's use of NBAs in UK ABWR, I am satisfied with Hitachi-GE's materials selection justification for the wrought materials used for the UK ABWR shroud support and CRD stub tubes.
270. For weld metals, Hitachi-GE's justification considers the case for these separately, and in doing so, makes a number of important assumptions. In the bottom head region of the plant there are a number of key welds made between the core shroud, shroud support and the RPV. Hitachi-GE selects the same weld metal (Alloy 82) for all of these welds. In doing so, one of Hitachi-GE's key assumptions is that all of the welds are equivalent. During GDA I asked (RQ-ABWR-0746 (Ref. 49)) Hitachi-GE why this assumption is appropriate, because the stresses at each weld location and therefore susceptibility to SCC may be different. Hitachi-GE's response explained their materials selection justification is based on several conservative assumptions; namely that the residual stress condition for all welds is tensile stress, and no suitable stress relief techniques are implemented. On this basis I conclude this aspect of Hitachi-GE's justification for the weld metal(s) selected is adequate. I also specifically asked (RQ-ABWR-0746 (Ref. 49)) why and how the use of Alloy 52 had been dismissed, mainly

because Alloy 52 has been used in BWR applications before as an overlay material for nozzles. Hitachi-GE argue the previous performance of Alloy 52 in BWR applications is not directly applicable, because these overlays were performed on components which have completely different geometries compared to the subject weld locations in UK ABWR. Furthermore, Hitachi-GE also argue Alloy 52 has inferior welding properties to Alloy 82. On this basis, from a chemistry perspective, I am satisfied with Hitachi-GE's justification to select Alloy 82 as the weld metal for the welds in this region of the plant.

271. In summary, from an overall chemistry perspective, I consider Hitachi-GE has selected good practice, modern grades of material for the core shroud, shroud support, CRD stub tubes and associated weld metals. I also consider they have taken all other reasonably practicable measures to mitigate the risk of SCC in this part of the plant. I judge the expectations of SAPs ECH.1, EAD.1 and EAD.2 have been met.

CFDW System Main Piping

272. The choice of materials selected for the CFDW system main piping has an important impact on the operating chemistry applied to UK ABWR because the corrosion products (mainly iron) released from the majority of the system are fed directly forward to the RPV, influencing the generation and transport of radioactivity and the nature and quantity of fuel deposits. The CFDW system makes a major contribution to the total amount of iron expected to be present in UK ABWR feedwater. In addition, depending on the materials selected, it may place specific requirements to make claims on the operating chemistry to mitigate materials degradation threats and/or other risks.
273. Hitachi-GE's overall approach to justifying the materials selected for the CFDW system main piping is to split the system up into two main areas based on the classifications they have assigned to the piping. As shown in Figure 11 below, Class 1 piping is located inside the RCCV and forms part of the pressure boundary. Hitachi-GE has designated all other CFDW system main piping outside of the RCCV as Class 3.

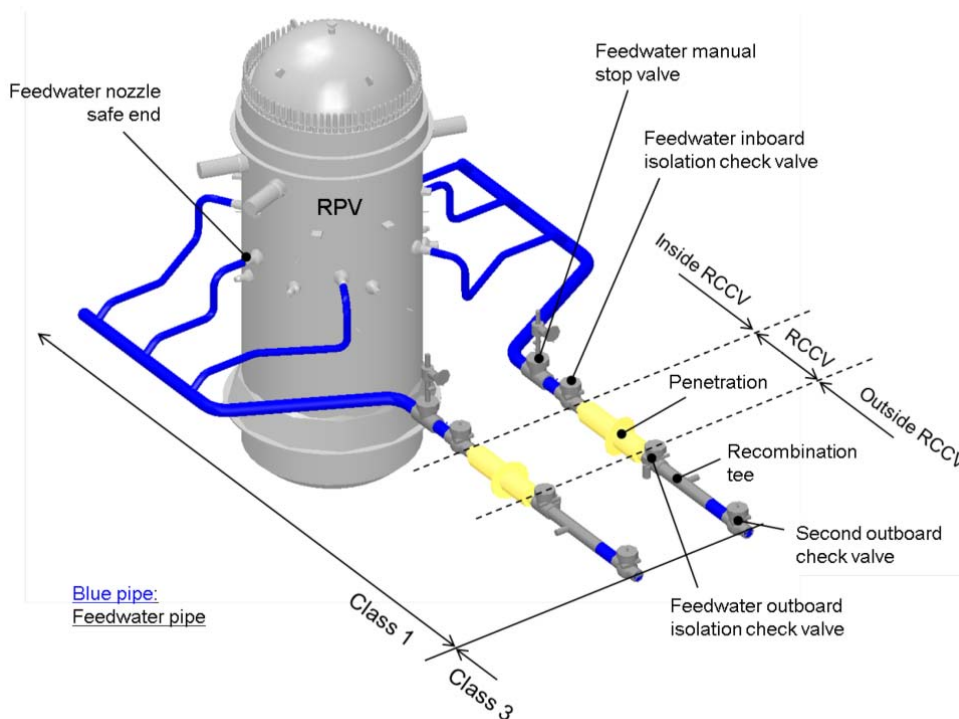


Figure 11: Segregation between Class 1 and Class 3 main feedwater piping in the UK ABWR (Ref. 55)

274. Assessing the adequacy of Hitachi-GE's classification of the piping is not within the scope of my chemistry assessment, but I am satisfied that splitting the system up in this way is a reasonable approach to take for the purposes of justifying the materials selected. Based on this, Hitachi-GE's principal decision is to select two different materials for the CFDW system main piping:
- Class 1 main piping will use LAS (chromium $\geq 1.25\%$); and
 - all Class 3 main piping will use CS with a specified and controlled minimum chromium content $\geq 0.1\%$.
275. Hitachi-GE argue the difference in nuclear safety consequences which would result from failure of these sections of piping (as reflected in their classification) is the biggest factor in justifying why it is appropriate to select different classes of materials. Their detailed rationale for selecting LAS for Class 1 piping is that structural integrity concerns dominate. In Ref. 50 and Ref. 55, Hitachi-GE argue that given the extent of Class 1 piping in the CFDW system is relatively small compared to Class 3 piping, and the fact that Class 1 piping is located in high dose rate areas inside the RCCV, the through-life cost savings of selecting LAS for Class 1 piping would offset the one-off cost. Hitachi-GE contends the same argument does not apply to Class 3 piping. By selecting LAS, Hitachi-GE's approach also recognises the need to implement Post Weld Heat Treatment (PWHT), and they argue this is readily achievable for Class 1 locations. From a chemistry perspective, I therefore agree with the arguments put forward by Hitachi-GE. I consider selecting LAS for Class 1 CFDW system main piping represents the use of a good practice material.
276. Ref. 55 argues there are several detriments and important technical reasons which mean using LAS, or another equivalently corrosion resistant material, for Class 3 sections of CFDW system main piping is neither necessary nor reasonably practicable. Hitachi-GE's key arguments are:
- Using LAS requires PWHT to be conducted. Given the larger pipe diameters and much larger volume of material required to fabricate Class 3 sections of the CFDW system, compared to the Class 1 sections, this would have a significant impact on the manufacturing schedule and introduce other integrity risks.
 - Japanese OPEX shows that when feedwater oxygen is properly controlled, the risk of FAC in CS can be adequately controlled and corrosion rates can be managed within corrosion allowances.
 - In most cases (except SS), using a different material would still require feedwater oxygen injection to be implemented for the purposes of minimising the release of iron from the system and the impact on radiological source term.
 - Selecting LAS would only offer a small benefit in terms of its overall impact on corrosion products released from the CFDW system.
 - Changing the material for all Class 3 main piping would incur a large monetary cost.
277. Using information made available to ONR by our TSCs (Ref. 43) I was able to determine that the CFDW systems for a large number of operational BWRs are fabricated from CS. Some plants use different materials based on whether the main piping is located inside or outside of the containment – SS is selected inside and CS outside, for integrity reasons. This particular approach provides support for Hitachi-GE's rationale for selecting two different material classes for CFDW system main piping.
278. For existing BWRs, the approach to mitigating the risk of FAC in the CFDW system is to inject oxygen into the feedwater and perform targeted in-service inspection, informed by evaluating the risk of FAC using various proprietary modelling codes. The OPEX shows (Ref. 43) this continues to be a largely successful strategy for managing the risk of FAC in the CFDW system for these plants; but it does demonstrate that

where feedwater oxygen is not properly controlled, FAC in BWR CFDW systems fabricated from CS can still be a problem. Based on the information made available to me (Ref. 43), the experience at existing BWR plants which operate with HWC (largely US plants) shows that injecting hydrogen into the feedwater can have the potential to cause a significant amount of iron to be released from the CFDW system. On this basis I am satisfied with Hitachi-GE's technical arguments regarding the necessity for oxygen injection, regardless of the material selected.

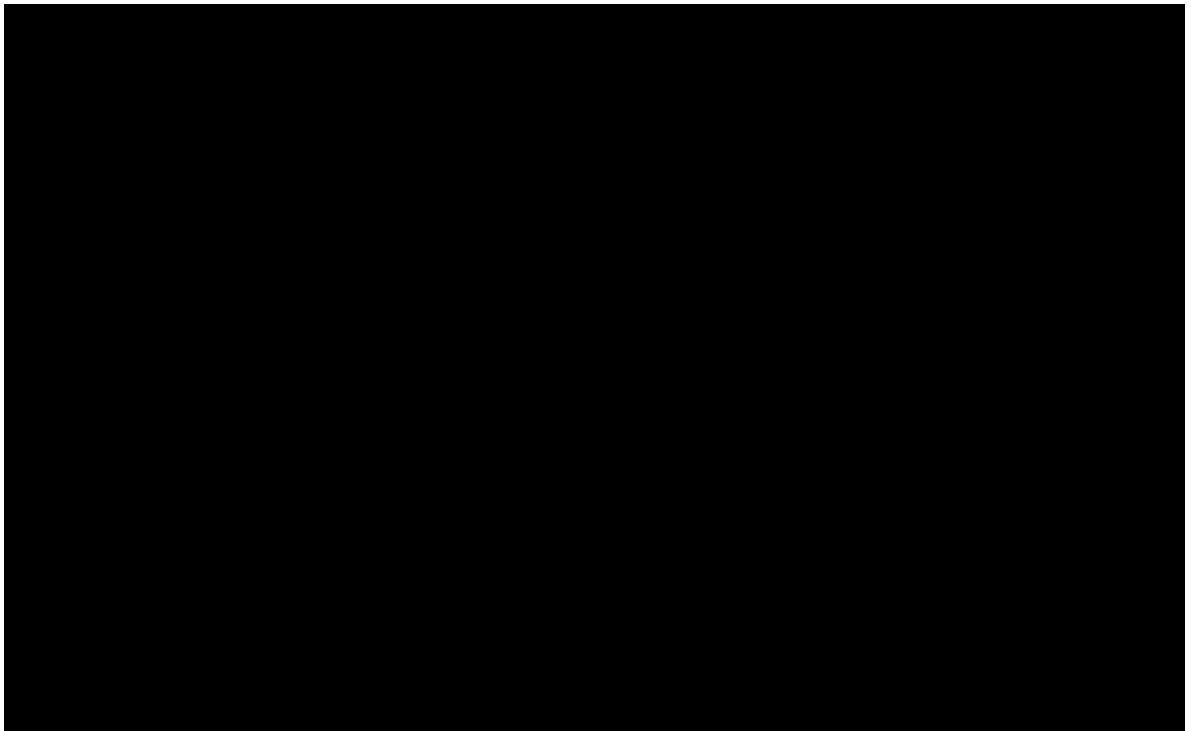
279. Overall, based on my consideration of the practices adopted for existing BWRs, I am satisfied, from a chemistry perspective, there is suitable and sufficient evidence to support Hitachi-GE's justification for the materials classes proposed for CFDW system main piping. The materials selected for UK ABWR are justified, reasonably practicable improvements compared to the majority of the existing BWR fleet, including J-ABWR. The important caveat is that feedwater oxygen injection still needs to be properly controlled. This has been assessed previously in Section 4.3.2.1 of my report, where I have concluded the lower "limit" value (15 ppb) selected by Hitachi-GE is not appropriate. **This is captured as part of Assessment Finding, AF-ABWR-RC-02.**
280. As well as implementing adequate control over oxygen injection, based on the case presented by Hitachi-GE, the other key requirement is ensuring the minimum levels of chromium specified for CS in Class 3 CFDW system main piping, are adequate. The general trend is for the FAC resistance of a material to increase as the chromium content becomes higher. For GDA, Hitachi-GE has specified the minimum level of chromium required in CS used in CFDW system main pipework as $\geq 0.1\%$. I agree in principal that specifying a minimum chromium content in the CS will provide a net benefit for FAC susceptibility. However, I am less convinced the specific level of chromium ($\geq 0.1\%$) Hitachi-GE has proposed during GDA, is the most appropriate lower "limit" that will provide sufficient margin to mitigate FAC. The precise level of chromium where the desired level of FAC performance is actually achieved will be determined by a number of factors (Ref. 42). In Ref. 50, Hitachi-GE consider the case for two potential options of CS with different ranges of chromium specified: 0.10-0.20% and $>0.20\%$, $<0.40\%$. In summary, Hitachi-GE assert selecting CS with a minimum chromium content in the range 0.10-0.20% represents the "baseline material" for the UK ABWR CFDW system, because:
- Specifying plate material with $>0.2\%$ chromium would be classed as a "special order" by suppliers, attracting additional costs and implications for schedule.
 - There may be technical issues with welding material with $>0.2\%$ chromium and code compliance issues.
 - There would be a limited benefit in terms of FAC rates when selecting material $>0.2\%$ chromium.
281. The balance of the arguments presented in Hitachi-GE's evaluation are directly related to choices a future licensee may wish to make, which could have a significant impact on the overall decision regarding the precise minimum chromium content to be specified. In addition, the position regarding the availability and therefore "special order" status of the materials may reasonably change over time, or alter, if different suppliers were selected. Given the importance of the chromium content of CS to the overall case presented by Hitachi-GE for mitigating FAC in the CFDW system, I therefore expect a future licensee to re-visit and further justify this position. Furthermore, Hitachi-GE assume a very specific benefit (in terms of FAC rates) will be gained by selecting a minimum chromium level of $\geq 0.1\%$. During GDA, aside from the information provided above, Hitachi-GE did not present a robust case to show this level of chromium provides enough margin to account for the potential variabilities in FAC susceptibility that should reasonably be expected across the range of systems, operating environments, and at different stages in the operating lifecycle of the UK ABWR. This conclusion is also supported by the independent work undertaken by TSC. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-06: To mitigate the risk of flow accelerated corrosion, the material of construction proposed for UK ABWR Class 3 Condensate and Feedwater system main piping is carbon steel, specified with a minimum level of residual chromium. The final decision regarding the appropriate minimum level of chromium to specify will depend on licensee design choices, and operator specific choices related to materials product form, and the selection of materials suppliers. Based on all of these considerations, the licensee shall justify the minimum level of residual chromium specified, reduces relevant risks SFAIRP.

282. In summary, Hitachi-GE worked hard during GDA to justify the materials selected for the UK ABWR CFDW system main piping. This has resulted in some important reasonably practicable changes. I am content with the case presented for GDA, however, from a chemistry perspective; I consider the relevant SAPs ECH.1, EAD.1 and EAD.2 have been partially satisfied at this point in time. This is because I have raised Assessment Finding, **AF-ABWR-RC-06**. Resolving this Assessment Finding which will specifically address gaps I have identified in the UK ABWR generic safety case presented by Hitachi-GE, which are relevant to the overall case to ensure the risk of FAC in the CFDW system is adequately controlled.

Heater Drain System Main Piping

283. In UK ABWR, the HD system recovers and recycles the waste steam/water that has been fed to the MSR and main feedwater heaters (FWHs) and utilises the drained steam/water to heat the feedwater, thereby increasing the plant's thermal efficiency. The HD system therefore experiences a relatively wide range of operating conditions including temperatures ranging from 58 to 280°C, single and two phase flows, high flow velocities and a range of DO concentrations. These conditions and the materials traditionally selected for the system mean FAC is the principal materials degradation threat of concern. A schematic of the UK ABWR HD system in UK ABWR is shown in Figure 12.



284. With respect to the design and arrangement of the Heater Drains (HDs), UK ABWR is a forward pumped^{††} BWR. Forward pumping has particularly important consequences for the operating chemistry because any insoluble corrosion products (predominately iron-based) present in the HDs are largely not removed by the CPS and directly enter the reactor. The basis of the UK ABWR design and generic safety case is that it will be a “low iron” plant. The concentrations of iron present in a BWR can have very important consequences for its safety performance, including risks to fuel integrity and increased worker doses (assessed in much more detail in Section 4.3.3 of my report). The materials selected for the HD System are therefore particularly important in this regard because they are one of the major sources of iron into the RPV.

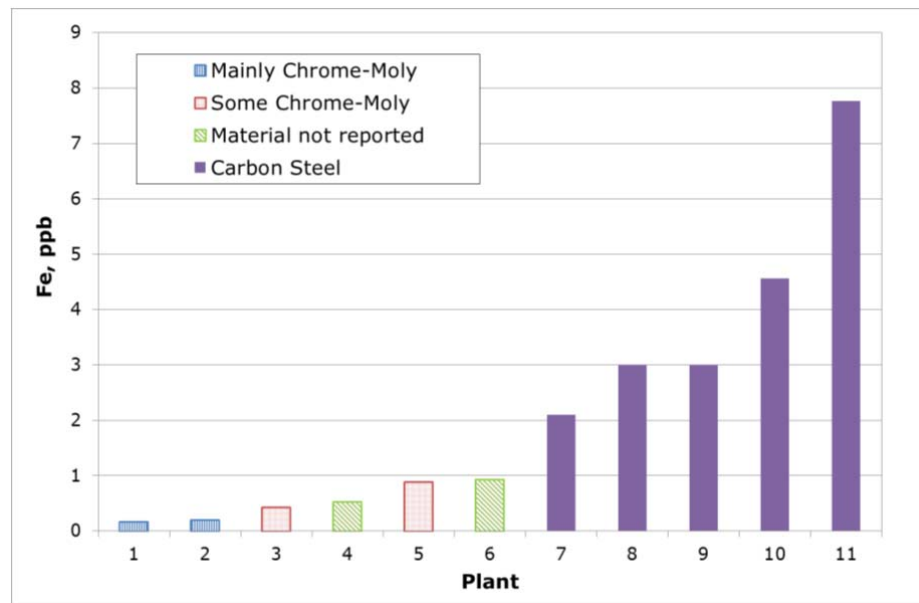


Figure 13: Feedwater iron concentrations in “forward pumped” BWR plants with different alloy selections (Ref. 60)

285. Figure 13 plots the feedwater iron concentrations for several BWR plants which are designed with forward pumped HDs. It shows the information as a function of the materials selected for the extraction drain, moisture separator and high pressure heater drain piping. These data clearly show forward pumped BWR plants operating with CS components in this part of the plant generally have higher concentrations of feedwater iron. During GDA, I also asked my TSCs to provide independent information (Ref. 43) on materials selection for the HD system for other BWRs. Based on the information provided, of those BWRs which were originally designed using CS for HD system main piping, primarily in response to FAC issues, many have gradually replaced large sections of piping with more corrosion resistant materials.
286. Ref. 55 contains Hitachi-GE’s materials selection justification for the HD system main piping for UK ABWR. Their approach is consistent with that undertaken for other systems. The HD system is split into a (relatively large) number of individual piping sections (or locations) based on criteria which include environmental conditions, and whether the sections are located upstream, or downstream of the Condensate Purification System (CPS), more specifically the Condensate Demineraliser (CD). The piping locations are identified above in Figure 12 using bracketed numbers. Broadly

^{††} The term forward pumped refers to the arrangement of the heater drains. In forward pumped BWRs a proportion of the flow from the heater drains is not recycled back to the Condensate Purification System for clean-up. It is fed directly forwards as feedwater to the reactor.

though, as shown in Figure 12, in UK ABWR the High Pressure Heater Drain (HPHD) is downstream from the CD and the Low Pressure Heater Drain (LPHD) is upstream.

287. For piping locations upstream (before) of the CPS (the CD), one of Hitachi-GE's key underpinning assumptions in their materials selection justification is that the impact of iron-based corrosion products from these regions of the HD system is negligible, because they are removed by the CPS. My assessment of the design of the CPS is presented later in my report, in Section 4.3.5.2. In brief though, the CD is not designed to remove insoluble corrosion products (which any corrosion products from the HD system will predominately be). The CD removes soluble species. The part of the CPS which functions to remove insoluble impurities is the Condensate Filter (CF). The very important point here is that neither the HPHDs nor the LPHDs in UK ABWR are recycled back for clean-up by the CF in the CPS. For this reason, I consider Hitachi-GE's assumption regarding the un-importance of corrosion products from the HPHD is not robust. This is an important discriminatory factor in Hitachi-GE's overall materials selection justification for the HD system, but the impact of the iron-based corrosion products is largely concerned with the generation and transport of radioactivity. I have therefore assessed this particular aspect of Hitachi-GE's justification for the HD system materials in much more detail in Section 4.3.3.3 of my report.
288. For this section of my report, dealing with materials degradation threats, the two most relevant criteria in Hitachi-GE's materials selection methodology for the HD system relate to when to select LAS. These criteria are namely, when the DO concentration is <15 ppb and where there is two-phase flow. These criteria are essentially all about mitigating the risk of FAC in the system. If either one of these two conditions is met, Hitachi-GE's methodology stipulates LAS should be selected.
289. Applying the above approach and criteria, as shown in Figure 12, the materials proposed for the UK ABWR HD system include relatively large sections fabricated from CS and LAS. This materials selection is largely the same as J-ABWR, with the exception of the following main changes (for locations as identified in Figure 12):
- Locations 16 and 20 (main piping leading from the Low Pressure Drain Tank (LPDT)) use LAS (J-ABWR uses CS); and
 - Location 30 (main piping the after High Pressure Drain Tank (HPDT)) uses CS with $\geq 0.1\%$ chromium (J-ABWR uses CS).
290. While it is positive that Hitachi-GE has recognised the need to make improvements to the UK ABWR design, the question is whether this particular combination of LAS and CS represent relevant good practice and reduces relevant risks SFAIRP. This is especially important because:
- OPEX shows the concentration of DO in the HD system is very important to determining the overall risk of FAC occurring.
 - The practice for US BWRs is to continue to replace sections of CS piping in the HD system if degradation is found during in-service inspections.
 - A smaller number (five) of forward pumped BWRs (in Sweden) do not use CS at all in the HD system. One uses SS and the other four use LAS (Ref. 43).
291. The basis for Hitachi-GE's decision to implement LAS in Locations 16 and 20 is a set of calculations and analysis which estimates the DO concentrations expected in various regions of the HD system. This includes considering OPEX from KK-6 and -7, the impact HWC will have on main steam DO concentrations for UK ABWR (they will reduce relative to NWC conditions, see Section 4.3.6 of my report) and the potential variation allowed by the relevant ORs defined in the generic safety case (principally for feedwater hydrogen). I haven't assessed these calculations in detail, given the inherent uncertainty in such analysis. However, the outcome is that Locations 16 and 20 are determined to have a DO concentration of 9 ppb. This value is lower than Hitachi-GE's

threshold value of 15 ppb used in their materials selection criteria, and therefore LAS is selected. Even on this limited basis, I judge that this change appears to be necessary for UK ABWR and I consider it to be a positive example of Hitachi-GE demonstrating the operating chemistry choice and materials selection decisions for UK ABWR are being taken in tandem; demonstrating the design is being optimised.

292. This also demonstrates the importance of the DO threshold selected. As explained previously in Section 4.3.2.1 of my report, I do not consider a DO concentration of 15 ppb is appropriate to be able to claim the UK ABWR operating chemistry mitigates FAC. Using Hitachi-GE's method for the HD system, if DO is estimated to be below this value, it triggers the selection of LAS. This means a significant part of the sensitivity of the outcome of Hitachi-GE's materials selection decisions for the HD system, relies on the acceptability of the DO threshold set at 15 ppb. As also previously introduced in Section 4.3.2.2 during my assessment of Hitachi-GE's FAC risk ranking for UK ABWR SSCs, during GDA, I therefore asked Hitachi-GE (RQ-ABWR-1465 (Ref. 49)) to undertake a sensitivity analysis. Setting the DO threshold for FAC mitigation to 30 ppb, consistent with the value recommended by the EPRI (Ref. 45), and applying Hitachi-GE's methodology, this results in four further locations where LAS would be selected:
- Locations 7 and 8 (main piping leading from the HPDT); and
 - Locations 13 and 14 (sections of main piping leading from Low Pressure Feedwater Heater #5 to the main condenser and Low Pressure Drain Tank (LPDT)).
293. Further materials changes may also be deemed necessary, if uncertainty is factored into any defined criteria. Irrespective of any decision over the precise criteria to be applied, what is clear is that the scope of the locations in the HD system identified by Hitachi-GE as being at increased risk of FAC, during GDA, does not appear to be robustly justified.
294. Another important point is that Locations 16 and 20 are in a very similar location to Locations 7 and 8, in terms of their environment, and therefore susceptibility to FAC. However, the materials for Locations 7 and 8 were not changed for GDA, while the materials for Locations 16 and 20 were changed (to LAS). Locations 16 and 20 are located in the LPHD, whereas Locations 7 and 8 are in the HPHD. As previously described, the LPHD is cleaned-up using the CD, so applying Hitachi-GE's assumption regarding the negligible impact of corrosion products from the LPHD because of this clean-up, this means it is actually be more important to change materials for Locations 7 and 8. However, for GDA, the materials for Locations 16 and 20 were changed instead. I consider this further calls into question the robustness of Hitachi-GE's materials selection decisions for the UK ABWR HD system.
295. Taking all of these factors into account, I am not satisfied that the material selection choice for the HD piping in UK ABWR adequately accounts for FAC risks. I judge that a future licensee will need to prepare a robust and appropriately justified case. Agreeing an appropriate set of materials classes has been selected to construct the plant from is fundamental. At present, even considering the risks to integrity alone, I consider the case presented by Hitachi-GE to justify the materials selected for the HD system is not robust. It does not provide an adequate demonstration to be able to justify all reasonably practicable steps have been taken to apply suitably corrosion resistant materials throughout the HD system, to mitigate the risk of FAC SFAIRP. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-07: During GDA, ONR's chemistry assessment established that some of Hitachi-GE's materials selection decisions for the Heater Drain system are sensitive to one of the important assumptions underpinning their materials selection criteria. The assumption is about the minimum acceptable levels of dissolved oxygen

necessary to mitigate the risk of flow accelerated corrosion. ONR concluded this assumption is not robust. Based on this conclusion, the licensee shall review, revise and update their materials selection criteria for the Heater Drain system. To demonstrate relevant risks are reduced SFAIRP, the revised criteria should be applied to justify whether more corrosion resistant materials should be selected for some parts of the Heater Drain system, where carbon steel is currently proposed.

296. This section only considers the integrity aspects of Hitachi-GE's justification for the materials selected for the HD system main piping. As already explained, I have considered the aspects of the case relating to the generation and transport of radioactivity separately, in Section 4.3.3.3 of my report. This is where my overarching judgement on the overall acceptability of Hitachi-GE's case presented for GDA for the materials selected for the HD system, is presented.

RHR System Main Piping

297. The RHR system performs a number of functions across normal operations and during fault and accident conditions. Its main role during normal operations is to provide cooling to remove decay heat in preparation for the outage. During accident and fault conditions, following a LOCA, it provides cooling and maintains the water level in the reactor core as part of a number of systems. The RHR system in the UK ABWR has three divisions and is therefore a substantial system within in its own right. Figure 14 gives a high-level overview of the system.

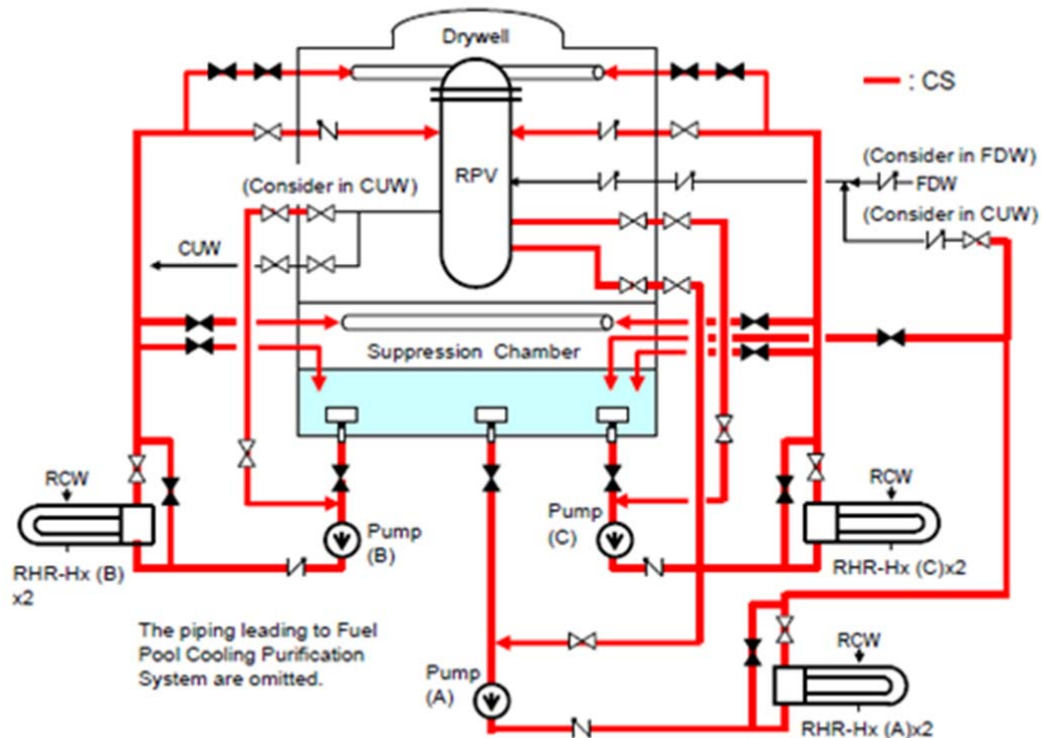


Figure GB3-29 Results of ALARP assessment for RHR pipe

Figure 14: Materials selected for the UK ABWR RHR System main piping (Ref. 55)

298. During GDA Hitachi-GE applied their materials selection ALARP methodology (Ref. 52) and selected CS for the RHR system main piping. From a chemistry perspective, because the RHR system is used during shutdown, one of the principal hazards which needs to be safely managed is the deposition of radioactivity (principally ^{60}Co) on the piping. This is because when a BWR is shutdown there is a "spike" in the level of soluble and insoluble radioactivity contained in the reactor coolant. The impact of the

RHR material selection on radioactivity is considered in greater detail in Section 4.3.3.3 of my report. This includes my assessment of the claims Hitachi-GE make on the importance of several operational practices which Hitachi-GE include in the UK ABWR generic safety case, which need to be implemented to ensure the generation and transport of radioactivity in the RHR system is minimised. I only consider the materials degradation risks here. Some of Hitachi-GE's key arguments of relevance to materials degradation, which underpin their decision are as follows:

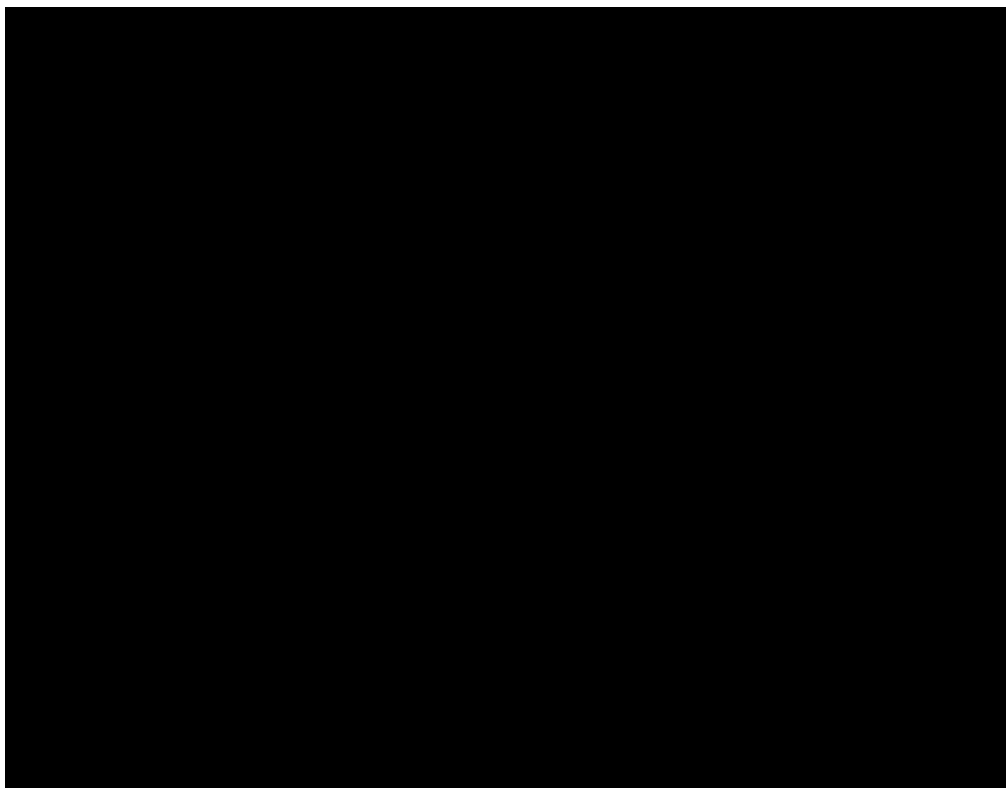
- There are important operational differences between different BWR designs which influence materials selection decisions.
- If SS is selected, the risk of thermal fatigue in the RHR system becomes significant.

299. To inform my assessment I again made use of my TSC's to identify materials selected by other BWRs for their RHR systems (Ref. 43). Based on the information made available, it is clear that using CS appears to be the preference; the exception is to use SS. The information also confirmed there are subtle, yet very important design differences between BWR plants. For example, the major difference between the UK ABWR and Swedish BWRs (which use SS for the RHR system) is that the Swedish plants continually operate large parts of the RHR system during normal operations, owing to the specific design of the RHR and CUW systems for these plants. For the UK ABWR design and largely for US BWRs (which use CS), almost all parts of the RHR system remain in standby mode during normal operations. This difference means that for Swedish plants, large parts of the RHR system are continually in contact with hot reactor water and the radioactivity it contains. In these plants thermal fatigue is not a significant concern, but radioactivity deposition is, hence the move to SS.
300. Selecting SS for the RHR system introduces the risk of SCC, that would otherwise be absent if CS were selected. This is reflected in Hitachi-GE's options scoring in Ref. 55, but is not specifically drawn out by them as one of the main arguments for not adopting SS for the RHR system in UK ABWR. Hitachi-GE's focus is that they believe using SS would introduce thermal fatigue as a significant concern. Assessing the detail of Hitachi-GE's evaluation of the likelihood of thermal fatigue in the RHR system is beyond the scope of my chemistry assessment, but this is clearly a relevant risk. Also, OPEX clearly shows IGSCC is a concern in BWR RHR systems fabricated from SS. The OPEX also provides further supporting justification for the observed wide-spread application of CS in BWR RHR systems.
301. Hitachi-GE also claims the relatively short time which the RHR system is in operation is an important deciding factor for the material selected. Considering the main materials degradation threats associated with using CS are general corrosion and FAC, I consider this is a reasonable assertion to make. General corrosion can be managed in the design by specifying a suitable corrosion allowance. For FAC, from a chemistry perspective, I consider that the environmental conditions expected in the RHR system suggest the risk of FAC should be minimal.
302. Overall, purely from a chemistry impact on materials integrity perspective, I am satisfied Ref. 55 provides an adequately justified position for the selection of CS for the RHR system main piping. I have reached this judgement on the basis that, purely from a material degradation viewpoint, SS poses higher levels of overall risk than CS for both thermal fatigue and SCC.
303. I have reached this conclusion on the basis that the identified operational practices for the RHR system, which I have assessed in Section 4.3.3.5, are also implemented. Given the selection of CS based on materials integrity concerns, these are fundamental requirements to be able to demonstrate an overall ALARP balance has been found in the materials selection process for the UK ABWR RHR system. The same point also applies to the application of suitable surface treatment techniques to

mitigate the release of corrosion products and/or deposition of radioactivity in the RHR system. I note the selection of CS means measures such as Electro-Polishing, which could potentially be applied if SS were selected, are rendered not possible. Given the significant contribution the RHR system makes to collective worker doses, I would therefore also expect a suitable surface treatment to be adequately justified, for Hitachi-GE to be able to claim the selection of CS for the RHR system is the overall "ALARP solution". All of these matters are assessed in more detail in Section 4.3.3.3 of my report.

CUW System Main Piping

304. Ref. 55 contains Hitachi-GE's justification for the materials selected for the CUW system. In the context of materials selection, during GDA, I focussed my assessment (RQ-ABWR-1087 and 1332 (Ref. 49)) on CUW system main piping. To undertake their materials selection justification, Hitachi-GE first splits the system up into seven discrete sections based on the considerations of safety classification and the operating environment each section of main piping is exposed to. The approach to splitting the system up in this way appears reasonable, but has important consequences, as I have described below.
305. I agree with Hitachi-GE's position in Ref. 55, where they consider the two most important materials degradation threats for CUW system main piping are FAC and SCC. The outcome of Hitachi-GE's consideration is to select three different materials for CUW system main piping: CS, LAS and SS. Figure 15 shows the sections of CUW system main piping where these different materials are proposed.



306. Comparing the materials selected for the CUW system, there are significant differences between the reference plant and the UK ABWR design; J-ABWR uses CS throughout. The main factor driving Hitachi-GE's decision to change the materials for the UK ABWR CUW system is the change in the operating chemistry.
307. More specifically, Hitachi-GE has selected SS owing to concerns about an increase in the deposition of ^{60}Co (and therefore increased worker doses), which might occur in

high temperature parts of the CUW system fabricated from CS. Hitachi-GE claim the impact of DZO injection will mitigate ^{60}Co pick-up on SS surfaces. This has driven Hitachi-GE's decision to change the majority of CUW system main piping inside the containment and leading up to the regenerative heat exchanger, to SS. The decision to use LAS is largely driven by concerns about FAC, as a result of the DO concentration in the reactor water decreasing, because of HWC. This has driven Hitachi-GE's decision to select LAS for the CUW system main piping between the regenerative heat exchangers and the feedwater line and the non-regenerative heat exchangers. I agree the risk of FAC is important to mitigate in this region of the CUW system.

308. For the regions of the CUW system where LAS is selected, Hitachi-GE's decision not to select SS is based on an assumption about the impact of temperature on ^{60}Co deposition behaviour. In Ref. 55 Hitachi-GE argues that below 150°C the behaviour of CS and SS, with respect to ^{60}Co pick-up, is the same. The operating temperatures where LAS is proposed are approximately 110°C . Whilst I agree there is an important correlation between radioactivity deposition and temperature, I have also noted Hitachi-GE's assumption regarding ^{60}Co pick-up and the threshold temperature selected, may be an oversimplification. I am not convinced a robust case has been presented during GDA to explain why SS isn't selected for this region of the CUW system, considering the material is also essentially "immune" from FAC.
309. CS is retained in the CUW system after the heat exchangers and before and after the CUW system F/Ds because the operating temperature is below 60°C , and Hitachi-GE take credit for the F/D's ability to remove ^{60}Co . Using Hitachi-GE's methodology, this means there are minimal concerns regarding FAC and ^{60}Co deposition in this region of the system, which drives their decision to retain CS. Similarly, as shown in Figure 15 above, CS is retained for the regenerative heat exchanger bypass line and also the spray line from the CUW system leading to the RPV top head.
310. Overall, from a chemistry perspective, I consider Hitachi-GE's decision to select SS to mitigate the deposition of ^{60}Co on CUW system main piping is a very positive step. I agree with Hitachi-GE's rationale for the need to select this material. This decision is a very positive example of how Hitachi-GE have taken the J-ABWR design and made optimised materials selection decisions for UK ABWR, based on the impact of selecting a different operating chemistry.
311. Despite this conclusion, within the detail of the specific materials selected for each region of the CUW system, I am not convinced the generic safety case has provided a comprehensive consideration of the potential impact of selecting several different materials, and the detriments this may bring. For example, considering the relevant risks being mitigated, it is not immediately obvious to me what the benefits of using LAS over SS are. Furthermore, I consider the impact of retaining CS after the heat exchangers and around the F/Ds needs further thought. In the absence of more detailed information being available during GDA, Hitachi-GE's materials selection justifications for the CUW system (like others) are based on certain key assumptions regarding the likelihood and severity of materials degradation threats and other key relevant factors (i.e. ^{60}Co deposition). As a future licensee produces more detailed information to better understand these relevant risks, I expect the key assumptions made by Hitachi-GE in the generic safety case in this regard, and by inference, the basis of the decision to use LAS and CS in the CUW system, to be re-examined. Considering the CUW system is continuously operated and is therefore continuously exposed to the radioactivity contained in the reactor water, the importance of appropriate materials selection in this system on reducing worker doses during outages is particularly pronounced. **I consider this is an Assessment Finding:**

AF-ABWR-RC-08: During GDA, as a result of the proposed change to the operating chemistry for UK ABWR, changes were also made to the materials selected for the reactor water clean-up system. Detailed analyses of materials degradation threats were unavailable during GDA, meaning it was necessary for Hitachi-GE to make certain important assumptions to underpin the materials selected. Based on the more detailed materials degradation analysis that will need to be conducted by the licensee; the licensee shall re-consider the use of low alloy steel and carbon steel for reactor water clean-up system main piping, to determine whether selecting more corrosion resistant materials, reduces relevant risks SFAIRP.

RPV Bottom Drain Line

312. The “bottom drain line” (BDL) is a UK ABWR design feature located in the bottom head of the RPV. It is part of the CUW system and carries a proportion of the flow of reactor water taken from the RPV and cleaned-up by the CUW system during normal operations. The line is shown in green in the corresponding part of Figure 15, above. During GDA I raised RO-ABWR-0034 (Ref. 25), asking Hitachi-GE to provide a suitable justification for the inclusion of a BDL in the UK ABWR design, because:
- Information provided to ONR by Hitachi-GE earlier in GDA indicated the BDL was a significant contributor to operator doses during outage maintenance activities for J-ABWR plants.
 - Earlier in GDA, Hitachi-GE were unable to clearly articulate the safety and or operational functions and purpose(s) of the BDL, particularly given it is a penetration to the RPV.
 - A number of operational BWRs have either permanently closed the BDL owing to concerns regarding operator dose and/or materials degradation, or purposefully designed out the BDL. (The latter point is directly applicable to internally pumped BWR designs, like UK ABWR).
313. My assessment of the chemistry aspects of Hitachi-GE’s case for the BDL mainly concerned examining their justification for the combined impact of the materials selected and the “new” operating chemistry proposed for UK ABWR. During GDA Hitachi-GE decided to justify the materials selected for the BDL separately to the rest of the CUW system. Their detailed materials selection justification for the BDL was therefore not presented as part of their wider suite of safety case documentation for materials. The TR, *ALARP Consideration on RPV Bottom Drain Line* (Ref. 61), is the UK ABWR generic safety case document which contains Hitachi-GE’s complete justification for the BDL, including the materials selected.
314. Ref. 61 is comprehensive. It identifies and explains the principal safety and operational functions the BDL performs. It also undertakes a thorough review of relevant worldwide OPEX to identify a credible list of alternative options to deliver these functions and identifies the relative benefits and detriments of each. The TR presents the outcome and rationale for scoring of these options (undertaken during a workshop) and a sensitivity analysis of the outcome. The outcome of this work is that the UK ABWR generic design will retain the BDL and in doing so incorporates several measures which are claimed to ensure risks are reduced SFAIRP, including:
- The material selected for the BDL will be changed from CS (used in the J-ABWR) to LAS (for the section of piping immediately below the RPV).
 - Changing the UK ABWR operating chemistry from NWC (J-ABWR) to HWC, OLN and DZO injection.
 - Implementing remote inspection techniques to further reduce work doses and manage conventional health and safety risks.
315. Overall, I consider the approach adopted by Hitachi-GE in Ref. 61 is very comprehensive and I am satisfied it meets the relevant parts of ONR’s guidance on

demonstrating ALARP (Ref. 4). One of the overarching principles in Ref. 4 is that the ALARP demonstration must be fit-for-purpose, which means its breadth and depth should be commensurate with the magnitude of the risk(s) under consideration. Considering the relevant risks associated with the BDL are relatively low (RQ-ABWR-1110 (Ref. 49)) compared to the overall spectrum of risk for the UK ABWR design, in this regard, I would say Ref. 61 is particularly sophisticated. The approach adopted by Hitachi-GE goes above and beyond what I would normally expect to see for these circumstances.

316. A key part of Hitachi-GE's conclusion to retain the BDL in the UK ABWR generic design is to change the material to LAS, which is mainly driven by Hitachi-GE's argument that implementing HWC increases the risk of FAC in the BDL for UK ABWR (if fabricated from CS). In support of my assessment of the appropriateness of this assertion, as part of my TSC's work on radiolysis modelling, I asked them to undertake an independent assessment of the likelihood of FAC in the BDL for UK ABWR (Ref. 60). The FAC analysis performed by my TSC suggests when implementing HWC, the BDL is not protected from FAC by oxygen for most of its length during normal operations (Ref. 60). The results of my TSC's independent analysis are broadly consistent with Hitachi-GE's corresponding analysis for the oxidant concentrations expected in the CUW system, presented in Ref. 47. This is the evidence Hitachi-GE use to support their claim that the BDL in UK ABWR (if fabricated from CS) is at risk of FAC. There are a number of contributory factors which lead to FAC occurring. Oxidant concentrations are a key factor, but not the only one determining the overall risk of FAC. My TSC's independent analysis of oxidant concentrations in the BDL does suggest though that Hitachi-GE's claim that the UK ABWR BDL is at risk of FAC (if fabricated from CS), and that a material change to LAS is a suitable countermeasure to address this risk.
317. Hitachi-GE worked very hard on this topic during GDA. The overall basis of Hitachi-GE's case for changing the material for the UK ABWR BDL is that retaining CS, in combination with the "new" operating chemistry proposed for UK ABWR, increases the risk of both FAC and potentially increases worker doses. Hitachi-GE claim changing to LAS helps reduce and minimise the risk of FAC and would not make worker doses any worse. On this basis, from a chemistry perspective, I am satisfied Hitachi-GE has presented a justified case for selecting LAS for the UK ABWR BDL. My conclusion is also supported by independent advice provided by my TSCs (Ref. 60). From a chemistry perspective, based on the reasoning provided above, I am satisfied the broad expectations of SAPs EKP.1, ECH.1, EAD.1 and EAD.2 have been fulfilled in this context.

Summary for Sample Assessment of Residual Materials Degradation Risks for UK ABWR SSCs

318. Overall, based on my sample assessment of a number of important (to chemistry) "representative SSCs", I am content the materials choices are consistent with the claims and evidence presented in the generic safety case, regarding the impact of the operating chemistry on materials degradation risks. Furthermore, I am content Hitachi-GE has provided suitable and evidence to justify the main materials gradation risks for these "representative SSCs" are reduced SFAIRP. As part of this work, Hitachi-GE has also identified several positive, necessary and reasonably practicable improvements to the materials selected for the UK ABWR design:
- Selecting more corrosion resistant materials (SS and LAS) for high-temperature sections of the CUW system.
 - Selecting LAS for Class 1 regions of CFDW system main piping inside the RCCV.
 - Selecting CS with a specified minimum chromium content ($\geq 0.1\%$) for Class 3 CFDW system main piping; and

- Moderately expanding the use of LAS for HD system main piping in certain parts of the system.
319. These changes are demonstrable evidence that Hitachi-GE is optimising the UK ABWR generic design to take account of both the increased 60 year design life of the plant and the different operating chemistry proposed.
320. During my assessment presented in this section of my report I have also raised three Assessment Findings, **AF-ABWR-RC-06, -07 and -08** and two Minor Shortfalls, **MS-ABWR-RC-03 and -04**.

4.3.2.4 Materials Degradation Summary

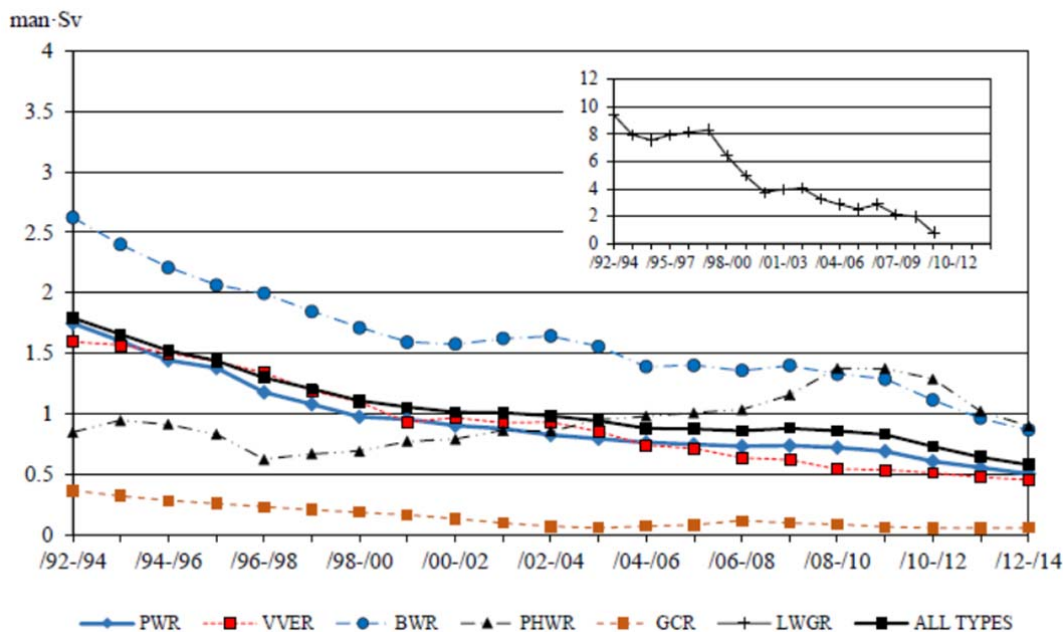
321. During GDA Hitachi-GE worked very hard on these aspects of the generic safety case and produced a substantial amount of documentation. The scope and volume of the work presented is perhaps best described as a “safety case within a safety case”. Hitachi-GE’s documentation suite is presented as a hierarchy. This does help make these aspects of the generic safety case more usable; however, the volume of work presented also means the documentation can be particularly repetitive in places, which can make extracting the relevant information more challenging. This is a presentational matter rather than a technical one, which can be addressed by the future licensee, to rationalise the documentation into a more usable and operationally focussed suite of documents.
322. The important point for GDA is that it is clear what the main materials degradation threats are for UK ABWR SSCs are and where and how, the operating chemistry is claimed to provide mitigation. The selection of HWC and OLNC, in tandem with several materials changes, has been demonstrated by Hitachi-GE to be a significant part of providing assurances that relevant risks are reduced SFAIRP.
323. In the main, I consider Hitachi-GE has devised an adequate materials selection process for GDA and I consider they have provided a breadth and depth of evidence which is commensurate with the requirements of a generic safety case. The reservations I had during GDA for this topic largely concerned Hitachi-GE’s application of their methodology, rather than the methodology itself, to justify the materials selected for a limited number of specific and important UK ABWR SSCs. This is reflected in the output of my assessment for the materials degradation topic in the form of 8 Assessment Findings, **AF-ABWR-RC-01 to -08** and two **Minor Shortfalls, MS-ABWR-02 and -03**. This means that although I am broadly satisfied any further changes implemented by a future licensee in this area would not undermine the UK ABWR generic design; there is still a significant amount of work to complete.
324. From a chemistry control perspective, one of the main areas for further development will be in relation to the definition of suitable limiting values for the identified ORs. At this stage, I do not consider what has been suggested is adequate to ensure risks are reduced SFAIRP. Nevertheless, I am satisfied these matters can be resolved by a future licensee without undermining the UK ABWR generic design.
325. My overall judgement against the SAPs is that in the context of a generic safety case, I am broadly satisfied the expectations of SAPs EKP.1, ECH.1, EAD.1 and EAD.2 have been met.

4.3.3 Radioactivity

326. Like all nuclear processes, the generation of radioactivity is an inherent consequence of the operation of a BWR. However, there are steps that can be taken, both during design and operation, that can minimise the generation of such radioactivity. ONR expect that radioactivity should be reduced SFAIRP, and an important part of the

assessment conducted during GDA has been to provide assurance that this has, or can, be achieved by the UK ABWR design.

- 327. The importance of such measures on safety is demonstrated by the fact that the radioactivity carried by the coolant of a BWR is a principal source of ORE and routine radioactive wastes as well as a potential source term in accidents. Most ORE in a BWR occurs during shutdown periods when intrusive maintenance and refuelling takes place, where up to 90% of the ORE is due to activated corrosion products (dominated by cobalt radioisotopes) (Ref. 62).
- 328. The control of coolant chemistry, selection of appropriate materials and operating practices are all important to the minimisation of radioactivity, dose rates, and ultimately to radioactive waste management and discharges.
- 329. Historically there has been a great variation in levels of radioactivity generated by different reactors, and even between BWRs. Figure 16 below illustrates the 3-yearly average world-wide annual worker doses (in man.Sv) by reactor type (Ref. 63), over around the last decade. While significant improvement have, and continue, to be made in BWRs in reducing ORE, they remain amongst some of the highest dose plants worldwide. Ref. 63 also contains the same data, but split by country. While it is difficult to directly compare results of this nature due to differences in: plant design and the number of plants and cycle history, these data show the general trend towards lower ORE seen in recent years for BWRs. It also demonstrates the large variations that are possible. The average is around 1 man.Sv and the maximum around 4 man.Sv.



Key: PWR – Pressurised Water Reactor; VVER – Voda-Vodyanoi Energetichesky Reaktor (Russian PWR); BWR – Boiling Water Reactor; PHWR – Pressurised Heavy Water Reactor; GCR – Gas Cooled Reactor; LWGR – Light Water Graphite Reactor

Figure 16: 3-year rolling average collective dose per reactor for all operating reactors by reactor type, 1992-2014 (Ref. 63)

- 330. Considering the significance of the hazards associated with radioactivity, and the importance of chemistry related aspects in determining the scale of these, I sampled a number of aspects related to this as part of my assessment. These broadly align with:

- defining and justifying the amount of radioactivity expected to be present within the UK ABWR design; and
 - demonstrating the impact of the material choices, operating chemistry and operating practices on radioactivity in the plant to show that these reduce radioactivity SFAIRP.
331. The main SAPs (Ref. 2) considered for this part of my assessment are those relating to the adequacy of the safety case (FP.4, SC.4, SC.5 and SC.6), consideration given to chemistry effects (ECH.1 and ECH.2) and those relating to minimisation of operator doses (EHT.5, RP.1) and radioactive waste (RW.2). Importantly, these cover the expectations that radioactivity is controlled and minimised, including giving sufficient consideration to chemistry related effects. In assessing these aspects of the safety case I have also considered the guidance provided within the associated TAGs (Ref. 4).

4.3.3.1 Defining and Justifying the Quantities of Radioactivity

332. The assessment of the quantities and control of radioactivity within the UK ABWR design was one of the first topics sampled during GDA. Failure to adequately define or justify the quantity of radioactivity could ultimately mean that the design, operations or controls specified may not be soundly based. It would also prove difficult to demonstrate that associated risks have been reduced SFAIRP. The Step 2 reactor chemistry assessment report (Ref. 64) discusses these aspects in more detail.
333. The original basis of the UK ABWR source terms was a report summarising industry experience up to 1973 (Ref. 65). Importantly this did not consider the chemistry or materials proposed for UK ABWR, nor even similar designs of BWR. To address these gaps the regulators (ONR and the Environment Agency) jointly raised RO-ABWR-0006 (Ref. 25) in April 2014. This RO was associated with:
- the definition and justification for the source terms for UK ABWR, including how these are used;
 - the demonstration of the impact of the material choices, operating chemistry and operating practices on radioactivity in the plant and to show that these reduce radioactivity SFAIRP; and
 - the demonstration that the information is adequately managed and controlled throughout the safety and environmental cases.
334. This RO considered all of the aspects bulleted above, including the definition (Action 1) and supporting evidence that was considered necessary to justify (Action 2) the amount of radioactivity in UK ABWR. Other actions under RO-ABWR-0006 deal with the management of source term information and the justification that radioactivity is reduced SFAIRP, and these are discussed separately in subsequent sections of my assessment. It is important to note that the scope of this part of my assessment is for normal operations. The amounts and behaviour of radioactivity during accidents is different, and is assessed separately in Section 4.7 of my assessment.
335. The original responses to Actions 1 and 2 were received during January 2015. These responses are not discussed in detail here, except to note that the approach within these reports was almost exclusively based on theoretical calculations, contained very limited justification and no recourse was made to plant Operating Experience (OPEX). The assessment of these submissions indicated a number of significant shortfalls in terms of meeting ONR's expectations. Given the significance of the identified regulatory shortfalls the definition and justification for the UK ABWR source terms during normal operations was escalated to a Regulatory Issue, RI-ABWR-0001 (Ref. 66), consistent with *Guidance to Requesting Parties* (Ref. 1) in June 2015. The expectations for the RI were essentially the same as those for RO-ABWR-0006 Actions 1 and 2.

Hitachi-GE's Responses

336. In response Hitachi-GE provided a suite of documentation which defines and justifies the concentration of radionuclides around the UK ABWR plant during all modes of normal operations (Refs 67 to 78). The basis, derivation methodology and calculated values are presented in these documents, as shown in Figure 17. The document structure uses a tiered approach that includes a high-level strategy report, an overriding manual, supporting reports and value data sets. This figure also shows how the various documents are linked.

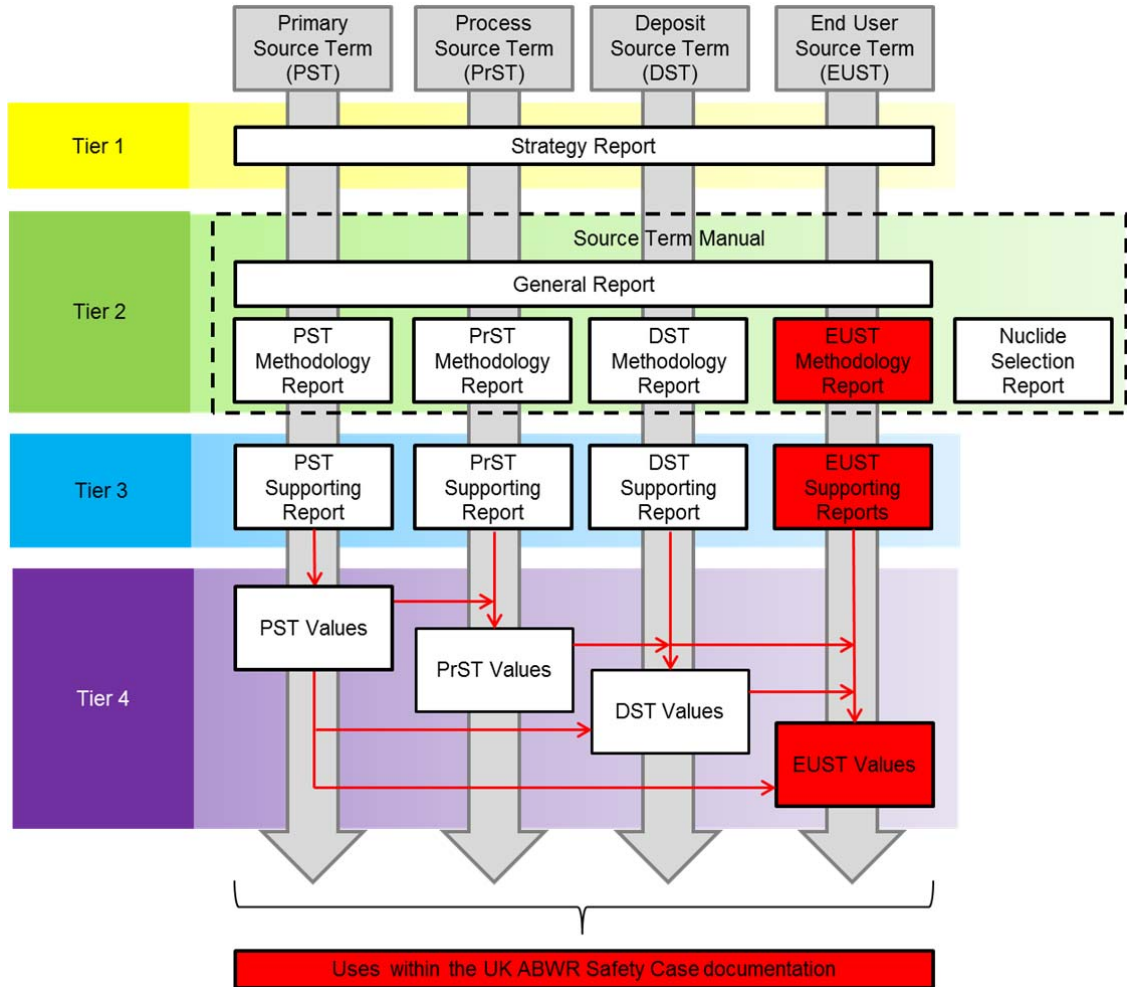


Figure 17: Overview of the RI-ABWR-0001 Submission Structure

337. Further details on this approach, and the contents of the various submissions are provided in my assessment report produced to justify closure of the RI (Ref. 79), so is not repeated in detail here. It is sufficient to note here that:

- Collectively the responses include information on radioactivity within the reactor, water and gaseous auxiliary systems as well as deposited on piping surfaces and fuel cladding.
- This considers all of the different categories of radionuclides (Corrosion Products (CPs), Activation Products (APs), Fission Products (FPs) and Actinides (ActPs)).
- The responses include information on the OPEX data selection methodology, key assumption sets, derivation and justification data and key radionuclide selection methodology, plus other information necessary to adequately define and justify the radionuclide concentrations.

- Hitachi-GE considers all phases of the operational cycle (i.e. system start-up, power operation, normal hot stand-by, shutdown and outage). Not every phase is explicitly defined however, as the RPs approach is to simplify by using bounding values for some phases of operation.
 - Given the nature of BWRs, namely with boiling of the coolant, it is important that the phase change is considered appropriately. The approach taken is to define the concentration of a given nuclide according to its solubility and volatility. This accounts for the fractions of radionuclides which remain in the water and that which is transported with the steam.
 - A range of derived values are calculated, including Best Estimate (BE), Design Basis (DB) and Cycle Average (CA) values. These cover the full range of requirements in the safety case. Further details are provided in Ref. 79.
338. There are some differences, but the basis of the approach adopted to define the amount of radioactivity is to use statistical analysis of OPEX data gathered from existing BWR plants. Hitachi-GE select a range of OPEX to provide the broadest data set that they judge pertinent to the design and operation of the UK ABWR (influencing factors include proposed water chemistry and material selection). The limitation with this approach is that the OPEX data of sufficient quality and providence is typically limited to a sub-set of the most significant radionuclides, which does not cover all of the nuclides necessary for UK ABWR. These gaps in the OPEX data are filled using verified and validated models, computer codes and supporting calculations. The underlying assumptions and parameters are defined and justified within the various supporting reports, alongside the OPEX selection methodology.
339. Hitachi-GE contends that the use of OPEX for derivation provides a degree of justification. However, additional independent justification for the source term values is also provided. The approach to this justification varies much more than for the definition aspect, including from nuclide to nuclide, but some combination of the following methodologies are employed, in this order of preference:
- further OPEX data analysis using additional (independent) plant information;
 - first principle computer codes and modelling calculations incorporating the proposed neutron flux, water chemistry and material selection for the UK ABWR;
 - published literature; or
 - sensitivity analysis.

ONR's Assessment

340. Considering the significance of RI-ABWR-0001 (Ref. 66), my complete assessment of Hitachi-GE's responses to it was reported separately in December 2016, when the RI was formally closed (Ref. 79). These responses were aimed at providing a suitable and sufficient definition and justification for the amount of radioactivity likely to be present within UK ABWR. As such, that part of my GDA assessment is not repeated in detail here. It is summarised below, alongside further developments and responses of relevance submitted by Hitachi-GE in the latter stages of GDA, since my assessment of the responses to RI-ABWR-0001 (Ref. 66) concluded. Ref. 79 considers Hitachi-GE's: scope, approach, methodology and radionuclide concentrations defined for UK ABWR. The main conclusions of my assessment were:
- The scope and approach adopted by Hitachi-GE in responding to the RI was adequate.
 - The use of relevant plant operating experience, utilising the broadest data set that is considered pertinent, gives confidence in the defined values. Where suitable data does not exist, recourse is made to other methods in a satisfactory manner.

- Throughout the development of the responses, suitable and sufficient consideration has been given to safety, including all significant radionuclides that exist in the systems expected to contain radioactivity throughout the envisaged operational states.
 - The defined UK ABWR source terms includes all appropriate sources of radioactivity within the plant, covering mobile and fixed sources, and considers how the nature and quantities of radioactivity within the plant may evolve over time.
 - Variations in radioactivity due to the different operational phases of the plant both in the short-term and long-term are appropriately considered, covering the entire fuel cycle.
 - Both Best Estimate (BE) and Design Basis (DB) values are defined, representing an expected and more conservative estimate for the likely levels of radioactivity within UK ABWR. The BE values derived represent a reasonable estimate, for nuclear safety case purposes, of the likely performance and behaviour of UK ABWR. A set of conservative DB values have been derived which should be suitable for use in the safety case.
 - Where uncertainties still remain due to the methodologies, assumptions or approach these would not have a significant impact on the derived values.
 - The derived values are further justified using additional OPEX, calculation, literature and sensitivity analysis. An adequate and proportionate degree of supporting evidence has been provided, which is focussed on those nuclides of highest safety significance.
 - While the responses have been updated several times throughout my assessment, sufficient evidence has been documented to capture and understand the basis of the UK ABWR source terms should this need to be revisited in the future.
341. Overall, Ref. 79 concludes that the defined UK ABWR source terms are fit for purpose in making the UK ABWR safety case. I judged that while further changes may still occur, these should only be minor in nature. It is notable that, in the interim, there have been no significant changes necessary to the source term documents (Refs 67 to 78), and hence these conclusions remain valid.
342. While my assessment at the time of RI closure identified a number of “*Residual Matters*”, none of these were considered significant enough to prevent closure of the RI at that time. However, a number of these have been subsequently progressed by Hitachi-GE as part of their on-going safety case development and it is therefore worth reviewing these now to confirm if they have been resolved, or need to be further captured in some way. Annex 7 summarises where these “*Residual Matters*” have been considered in this or other Step 4 assessment reports for UK ABWR, noting that one to four were matters for ONR to consider and these have therefore been completed by production of this assessment report. In summary, I consider that all of these Residual Matters have been satisfactorily addressed by Hitachi-GE since my original assessment of RI-ABWR-0001 (Ref. 79). They do not undermine any of my conclusions and add further weight to the arguments and evidence presented within the safety case. While number 12 has not been reflected within the submissions, I am content this is not a significant omission. However, I do consider that this matter should be addressed by the future licensee. **I consider this to be a Minor Shortfall:**

MS-ABWR-RC-05: ^{16}N is the main radiation source during at-power operating conditions for UK ABWR. During GDA, Hitachi-GE was unable to reference plant operating experience to justify the expected concentrations of this nuclide, due to confidentiality issues. The licensee should review and update the UK ABWR safety case to include additional evidence to justify the defined ^{16}N concentrations in both reactor water and steam.

4.3.3.2 Hitachi-GE's Approach to Reducing Radioactivity

343. Chapter 23 of the PCSR (Ref. 11) recognises the importance of the impact of chemistry related matters on the generation or accumulation of radioactivity in UK ABWR; making several explicit and a number of implicit claims. The explicit claims all fall under the "top claims" in this area, namely that:
- "[RC SC5] *The UK ABWR reactor chemistry regime will ensure that the source term radiological dose to the worker is ALARP by optimising materials selection, operating chemistry and operating practices*";
 - "[RC SC6] *The UK ABWR reactor chemistry regime will ensure that the radionuclide releases and exposure to public is ALARP*"; and
 - "[RC SC7] *The UK ABWR reactor chemistry regime will ensure that the radionuclide releases and exposure to worker is ALARP*".
344. Specific claims made under these include:
- "[RC SC5.1] *The source Term will be minimised to ALARP in association with material selection*";
 - "[RC SC5.2] *The source term will be minimised to ALARP in association with operating practices*";
 - "[RC SC5.3] *The source term will be minimised to ALARP in association with operating chemistry*";
 - "[RC SC5.4] *Commissioning activities will be optimised to ensure that the longer term radiation exposure will be minimised to ALARP*";
 - "[RC SC5.5] *Source of activation products loaded from condensate and feedwater system to the reactor will be minimised to ALARP by the operation procedure of the condensate purification system*";
 - "[RC SC6.1] *The amount of the activation product (¹⁶N) transferred to the steam will be maintained low by the control of HWC+OLNC*"; and
 - "[RC SC7.1] *Radionuclides in the reactor water will be kept below levels that results in increased radionuclide release and exposure, and minimised to ALARP in normal operations by CUW and OG system operations*".
345. There are also a number of claims made under "top claim" [RC SC3] (see Annex 4), which is concerned with fuel integrity. However, these claims are also relevant to radioactivity, as maintaining the fuel cladding integrity is the barrier to releases of radioactivity into the reactor coolant. The impact of the operating chemistry on fuel is a discrete topic in its own right and is therefore assessed in detail in Section 4.3.1.4 of my report.
346. In general, I am content that the above claims reflect the aspects where chemistry may influence radioactivity within a BWR, although I do note that the claims themselves are often not clear or specific enough. This is a presentational issue with the PCSR, rather than a flaw in the underpinning arguments or evidence. Importantly the overall objective, of reducing radioactivity within the plant SFAIRP, features prominently within the generic safety case.
347. A key document submitted by Hitachi-GE to explain their approach for reducing radioactivity is *Topic Report on Radioactivity Behaviour in UK ABWR* (Ref. 80). It provides a summary of Hitachi-GE's fundamental understanding of the behaviour of radioactivity and its control in UK ABWR. The approach taken by Hitachi-GE is to treat each category of radionuclide (i.e. CPs, APs and FPs/ActPs), separately. Further sub-division is necessary on some occasions below this categorisation when the behaviour of individual nuclides differs significantly; in particular for the APs. In providing this understanding the report:

- Provides a high-level overview of how radioactivity is generated in the reactor core, transported around the plant systems and then deposited on the inner surfaces of equipment and pipework.
- Describes the fundamental management principles employed for controlling the above processes.
- Identifies the most important factors in controlling radioactivity and ranks them in order of importance; and
- Explicitly makes the links between specific safety claims and operating conditions to the relevant evidence presented in other document submissions and the wider generic safety case.

348. As is evident by the fourth bullet point above, the report itself is a high level summary of the detailed evidence that can be found in other reports. Ref. 80 provides the hierarchy of documentation, which together, comprise the safety case for radioactivity within UK ABWR. This is shown in Figure 18.

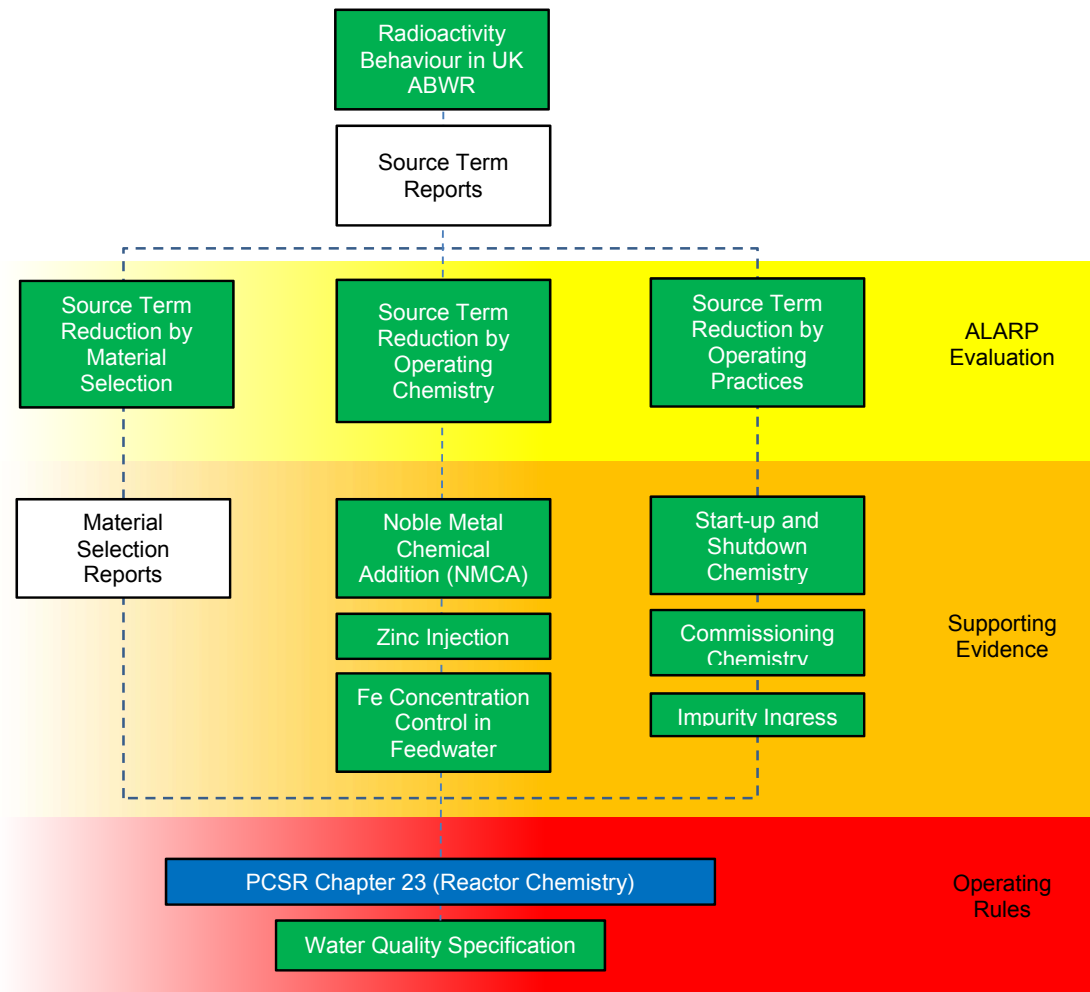


Figure 18: Document map for radioactivity within the UK ABWR generic safety case

349. An important and valuable aspect of Ref. 80 is that it provides the fundamental behaviour of radioactivity expected for UK ABWR. Considering the inherent design of BWRs, namely the boiling which occurs within the core, the chemical environment and the relatively long residence time within the core region, directly influence this behaviour. Specific aspects of this behaviour are assessed in Ref. 79, and are further discussed in subsequent sections of my assessment.

- 350. It is clear that this behaviour is reflected within the claims made by Hitachi-GE, and also therefore with the content of Ref. 80 and the other reports which provide the detailed supporting evidence. Importantly this behaviour is consistent with the assumptions and approach used by Hitachi-GE to define the source terms, as described within my assessment in Ref. 79.
- 351. An important conclusion made in Ref. 80 is that the management principles that can be applied to control and reduce radioactivity within UK ABWR are related to operational chemistry, materials selection, operational practices and quality assurance procedures. I would agree with this conclusion. While the latter of these is outside the scope of GDA, the former three can clearly be influenced by the generic design and hence were assessed. More specifically Ref. 80 identifies a number of important factors for each category of nuclide which is where the claims are made for UK ABWR in terms of controlling and minimising radioactivity. These are given in Table 2.

		Corrosion Products	Activation Products	Fission Products and Actinides
(High)		Minimisation of high-cobalt alloys	Minimisation of impurity ingress	Minimisation of fuel failures
		Minimisation of trace impurities in construction materials in the feedwater and condensate system	Minimisation of ¹³ N and ¹⁶ N transfer to the steam phase	Minimisation of impurities in the zirconium fuel cladding
Importance		Minimisation of the release of activated CPs from the fuel surfaces	Reduction of ³ H in the plant via discharge and make-up processes	Minimisation of tramp uranium contamination on fuel cladding surfaces
		Minimisation of deposition of activated CPs on out-of-core surfaces		Minimisation of FP releases from failed fuel by the use of power suppression
(Low)		Minimisation of the input of metallic impurities into the reactor		
		Minimisation of release of activated CPs formed via neutron activation of components		

Table 2: Summary of Important Factors for Radioactivity Management

- 352. The relative importance of each of these is subjective, but I do agree that these are the most relevant factors to be considered. While I would expect these all to be considered in an ALARP context, I am content that Hitachi-GE has appropriately ranked these in terms of importance. I assess each of these factors in the following sections, including the detailed supporting evidence.

4.3.3.3 Reducing Radioactivity by Material Choices

- 353. From a radioactivity perspective, an important characteristic of the RCS are the materials which are in contact with the coolant. Corrosion causes metallic CPs to be carried by the reactor coolant into the core, where they may become activated in the radiation field. Some of the most important and common CPs in UK ABWR are given in Table 3.

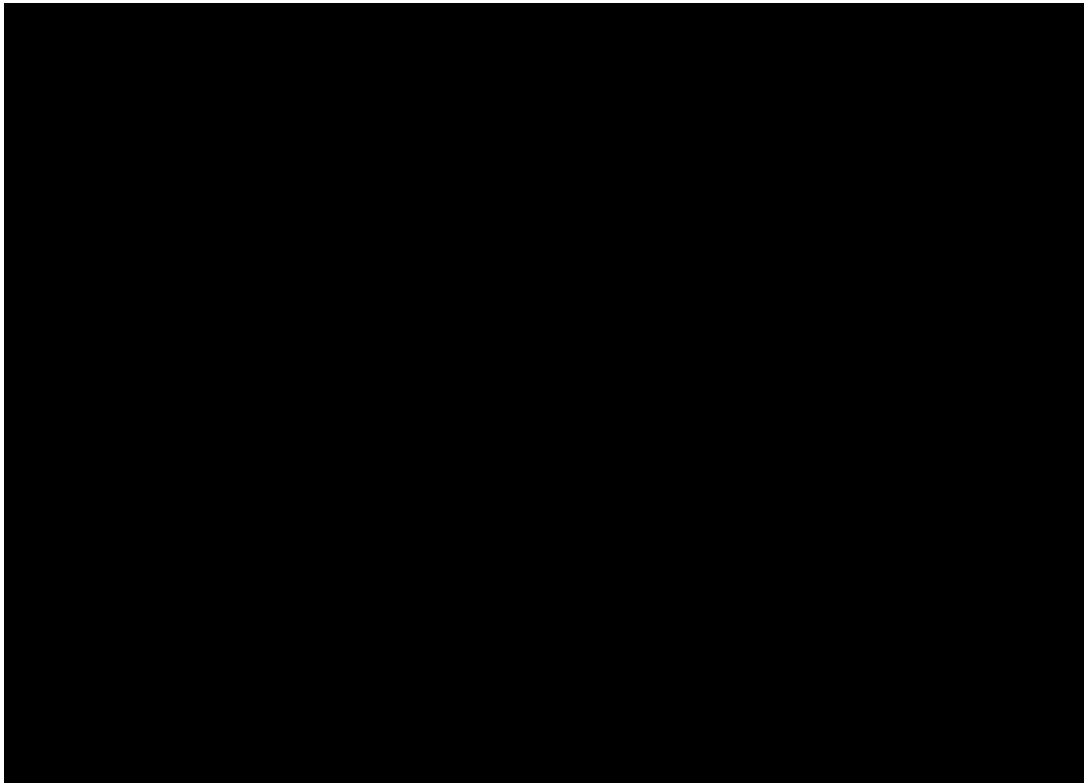
Nuclide	Half-life / days	Production Route	Main Sources
⁶⁰ Co	1925	⁵⁹ Co (n,γ) ⁶⁰ Co	SSs, NBAs, Stellite™
⁵⁸ Co	71	⁵⁸ Ni (n,p) ⁵⁸ Co	NBAs
⁵⁶ Mn	0.1	⁵⁵ Mn (n,γ) ⁵⁶ Mn	Steels
⁵⁴ Mn	313	⁵⁴ Fe (n,p) ⁵⁴ Mn	Steels
⁵⁹ Fe	45	⁵⁸ Fe (n,γ) ⁵⁹ Fe	Steels
⁵¹ Cr	28	⁵⁰ Cr (n,γ) ⁵¹ Cr	SS and LAS

Table 3: Typical Corrosion Products expected in UK ABWR

354. As shown in Table 2, the selection of appropriate materials in contact with the coolant is a primary factor that determines the susceptibility of the reactor to the production of activated corrosion products. In a BWR, this is particularly true for the CFDW system, which is a major source of such material. As described previously (Section 4.3.1.1) Hitachi-GE have put considerable effort to justify the material choices for the various systems and components in UK ABWR (Refs 52 and 55)]. The process applied includes the production or deposition of radioactivity as one of the attributes which influences the final selection. *Topic Report on Source Term Reduction by Material Selection* (Ref. 81) also includes information on the key materials and components which influence radioactivity, historical developments in BWRs and details of how Hitachi-GE consider they have reduced radioactivity SFAIRP by appropriate material choices. In general, I am content with this approach for considering radioactivity related hazards. However, there were a number of specific instances where the application of this process did not generate results which were fully justified, or were different than might be expected given OPEX and other good practices adopted elsewhere. It is in these instances where large parts of my assessment focussed, and are discussed below.

Minimising High-cobalt Alloys

355. The minimisation of high-cobalt alloy use (predominantly Stellite™), as indicated in Table 2, is the factor of highest importance for CP production in UK ABWR. Given that such alloys are known to be a significant source of ⁶⁰Co, and this nuclide is the most important in terms of ORE, it is appropriate that emphasis is placed on minimising the use of such materials. The potential benefits of reducing these materials are illustrated by Refs. 24, 45 and 82 for example. The relative contribution of different SSCs to cobalt release in UK ABWR is shown in Figure 19. The two largest contributors, valves and FMCRD, which make up nearly █████ of all cobalt releases into the coolant, are due to high-cobalt alloy use.



356. These alloys are used because they offer important mechanical properties, particularly in situations where wear or tribology concerns are important and high reliability needs to be assured. Ref. 81 recognises these conflicting requirements and presents a case to attempt to justify that, while some Stellite™ remains in the UK ABWR design, its use is minimised SFAIRP. The case also draws upon the improvements made for the ABWR design, compared to older BWRs. For example, the control rods pins and rollers use alternate nickel-base or iron-base alloys, respectively, eliminating this contribution to the cobalt source term. These are valuable and important improvements, but in themselves do not demonstrate that the UK ABWR generic design is “ALARP”, in this context.
357. During GDA, the UK ABWR generic design initially presented to ONR had around [REDACTED] of high-cobalt alloy. This is almost identical to the J-ABWR reference design. Over half of this is present in various components of the FMCRDs (the remainder is used as valve seats). Hitachi-GE could not provide comparable information from other BWRs; however this was made available to ONR via our TSCs in Ref. 43. This indicates that UK ABWR appears to have more Stellite™ than similar plants; for example similar powered US plants have around 18m², however, the difference is not so large if the FMCRD material isn't included in the comparison, indicating their important contribution.
358. For the FMCRD components, Ref. 81 explains the testing conducted during the development of the ABWR design, when the original motor driven CRD technology from European BWRs was imported. This testing demonstrated that replacement of Stellite™ in a number of FMCRD components was possible. Many others, however, were retained as the replacement materials failed during testing. This allowed over [REDACTED] of the high-cobalt alloys to be removed during the development of the ABWR design. Details are provided of the remaining components and their relative contribution to cobalt release, but the argument is made that data is not available to support further reduction. Hitachi-GE acknowledges that the FMCRDs will contribute almost [REDACTED] of all the cobalt release in UK ABWR (noting that I consider this an

underestimate, given the assumption regarding wear releases), and that other BWRs have removed all the Stellite™ from similar components.

359. Further evidence on this was provided in response to RQ-ABWR-1339 (Ref. 49), which, while not demonstrating this at a component level, did provide further clarification on the importance of the remaining Stellite™ in achieving the safety claims made on the FMCRD components. The important factor is that the mechanical requirements placed on the FMCRD parts in J-ABWR are more severe than in comparable European BWRs. The scram velocity and kinetic energy are considerably higher placing significant demands on the materials. This high-speed scram type CRD was developed and applied to expand the operating margin of BWR plants to improve plant safety. This faster drive speed was retained in J-ABWR, and hence UK ABWR. Given the current uncertainty in the performance of replacement materials under these particular conditions, I am therefore content that this errs on the side of safety and justifies the need to retain the high-cobalt alloys within this part of the design. However, the response to RQ-ABWR-1339 (Ref. 49) also notes that further replacements may be possible in the future, but this would require testing and qualification. I accept this as a reasonable position for this stage in the design process for the reactor. Given the significance of this potential contributor to radioactivity (for BWRs ⁶⁰Co is the most dominant contributor to worker doses accrued during outage maintenance activities), I expect the future licensee to provide a more detailed justification, at a component level, potentially including the outcomes from further testing or more recent experimental studies on this topic. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-09: The UK ABWR generic design includes high-cobalt alloys within the Control Rod Drives and valves, where mechanical properties and reliability requirements are important to safety. However, Hitachi-GE recognise that high-cobalt alloys make a significant contribution to radioactivity within the plant, (cobalt radioisotopes dominate doses received by workers during BWR outage maintenance activities). Licensee choices and further work are needed to fully substantiate the extent of the use of high-cobalt alloys in the UK ABWR design. The licensee shall justify that retaining high-cobalt containing alloys within components that contribute to the generation of radioactivity, reduces relevant risks SFAIRP. This shall include considering relevant operating experience, testing and the mechanical properties required.

360. The other significant usage of Stellite™ in UK ABWR is within valves. Ref. 81 provides information on the different types and locations of valves. The vast majority (>████) is within the CFDW and MS systems. Hitachi-GE estimates that these valves are the greatest source of cobalt release into the coolant of UK ABWR, accounting for around █████ (again, noting that I consider that this is potentially underestimated due to wear releases). Hitachi-GE's approach is that the choice of valve seat hard facing material is based on multiple factors such as size, nuclear safety significance, location, fluid property, fluid temperature, valve type and frequency of use. It is widely acknowledged, including by Hitachi-GE that cobalt-base hard facing materials generally offer superior performance and reliability compared to nickel- or iron-base alternatives. This is a reasonable starting point for reviewing if further reductions are possible.
361. The approach taken by Hitachi-GE is to provide a process, in the form of a flowchart, by which decisions over valve materials can be made, and to demonstrate this by applying it to a number of examples. This does not consider all of the valves within the plant, but the intent is to show that the process is reasonable, and could be further applied by the future license. I am content with this for GDA, given that the choice of materials would not affect the fundamental plant design. I therefore considered the adequacy of the process and the conclusions reached, including whether any changes

could be considered reasonably practicable at this stage of the design. In simple terms five main factors are considered: valve part, valve type, diameter, contact stress and OPEX. The material decision can either be Stellite™ #6, Stellite™ #21, or cobalt-free alloy. In the context of the process itself, I note that:

- The fundamental approach adopted is reasonable. I am content that such a method could be applied successfully by a future licensee. This puts the future licensee into a strong position to further review the use of high cobalt alloys in valves.
- The criteria used include a number of the main factors of relevance. However, I am not convinced that they are as complete as they may need to be for all cases. For example, while Hitachi-GE claim that valve usage and safety significance (i.e. categorisation and classification) are included as part of the existing criteria, this has not been demonstrated in the examples given during GDA. Similarly, the application of some of the criteria may not be adequate in all circumstances, nor their sequencing.
- The overall process is similar to that derived by the EPRI in Ref. 45, although not identical. It is not clear that the same conclusions would be reached by both methods, and the reasons for any differences are not apparent.
- Where specific numeric criteria are used (e.g. [REDACTED] valve diameter or 100 MPa contact stress), although Hitachi-GE explains the basis for these, the evidence for their applicability in all circumstances is not convincing.
- The difference between Stellite™ #6 and #21 is claimed to be significant in terms of corrosion releases, also due in part to the lower cobalt content. This ignores the inferior mechanical properties of Stellite™ #21 (relative to Stellite™ #6). As the criteria used by Hitachi-GE do not consider wear, it is unclear what the net effect of these competing parameters would be. I do not agree therefore that the choice of Stellite™ #6 or #21 is as significant a factor as implied by the process.
- One of the main decision points in the process is the availability of suitable OPEX, particularly in the longer term. Hitachi-GE acknowledges this as a gap in their knowledge. Despite this, and the compelling evidence that Stellite™ replacement is beneficial, great emphasis is put on anecdotal evidence that such replacements have not been successful from a mechanical performance perspective. Given the lack of OPEX I would not agree with this conclusion at this stage. It is clear therefore that a future licensee will need to review and develop their own awareness in these matters to be able to use this process or similar, to further optimise the UK ABWR design. This is an important task, but I am content that a competent licensee should be able to do so as part of normal business.

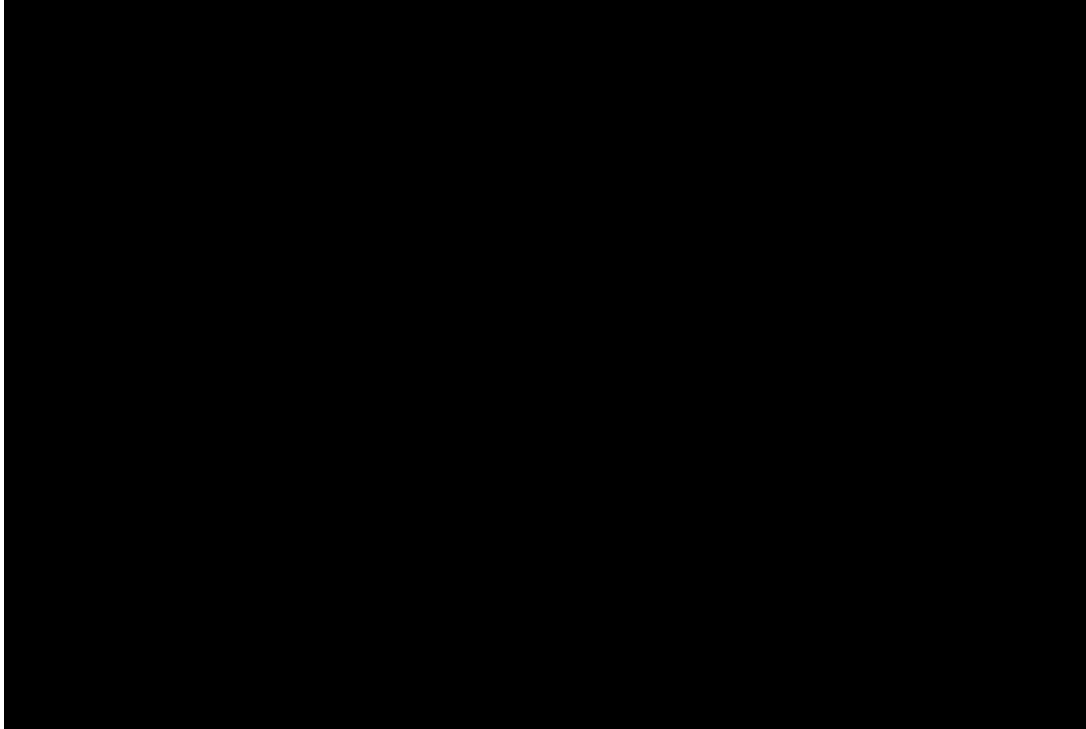
362. Notwithstanding the above, the process is applied to nine specific valves which use Stellite™ #6 in the J-ABWR design. This includes a range of valve types, sizes and operating conditions which cover the likely scope fairly well. The result of the evaluation is to retain Stellite™ #6 in one valve, and to replace it with Stellite™ #21 in two valves and cobalt-free alloy in one valve. For five valves the decision is to use cobalt-free materials, provided additional OPEX can be obtained to support the longer term viability of replacements, otherwise Stellite™ #21 would be specified. The deciding criterion differs for each valve, but these demonstrate their importance in particular the reliance on suitable OPEX. The approach taken by Hitachi-GE at this stage is to retain high-cobalt alloys unless positive confirmation of their long term performance can be obtained. The choice is also influenced by the factors described above regarding the process itself and is partly based upon Hitachi-GE material tests which remain on-going. This is a conservative decision for this stage, but is unlikely to represent the final position.

363. The response also provides information on the potential benefits that could be achieved. Firstly, it is demonstrated that it would be grossly disproportionate to change all the high-cobalt alloy valves. In other words, some valves are much more important than others in terms of cobalt release. This is not surprising, and validates the approach of considering individual valves or groups. Based upon retaining Stellite™ #6 where necessary, replacing it with cobalt-free alloys where practicable and using Stellite™ #21 where further OPEX is necessary, Hitachi-GE estimate that the contribution to the ⁶⁰Co source term by valves can be reduced by [REDACTED] in UK ABWR compared to J-ABWR. Where Stellite™ #21 is replaced by cobalt-free materials; a further [REDACTED] reduction could be achieved. There is some uncertainty in these values, particularly as they are heavily based on the assumptions, but they do demonstrate the scope available to the future licensee.
364. In assessing these examples, and the overall case made by Hitachi-GE during GDA, I have therefore concluded that:
- I am satisfied a process similar to the one derived by Hitachi-GE could be applied by a future licensee to evaluate the impact of valves with a high-cobalt alloy content, on the UK ABWR design.
 - For the reasons given above regarding the process itself, I am not yet convinced that the “ALARP position” has been reached for UK ABWR. I would anticipate further changes may be shown to be reasonably practicable.
 - I am content that it would not be meaningful to specify further changes at this generic stage of the design, despite the fact that I do not agree with some of the decisions made by Hitachi-GE to date.
365. Considering the importance of this source of ⁶⁰Co, my lack of agreement with Hitachi-GE’s position during GDA, and the significance and scale of further work required, I consider this to be a significant matter for the future licensee to resolve. However, I also note that any future changes made in this regard would not alter the fundamental generic design (i.e. the number and types of valves is very unlikely to change). **This has been captured as part of Assessment Finding, AF-ABWR-RC-09.**

Minimising Trace Cobalt Levels

366. Another important factor in reducing radioactivity via material choices relates to minimising the trace levels of cobalt. For the remaining components in Figure 19, the source of cobalt release is not high-cobalt alloys, but rather from general corrosion of structural materials. While the levels of cobalt are often very low anyway, even small changes in the concentration can be significant given the large surface areas of such materials in contact with the coolant. Decisions made here can influence several of the most important factors in controlling and reducing radioactivity as identified by Hitachi-GE in Table 2, including, “*minimisation of trace impurities in construction materials in the feedwater and condensate system*” and “*minimisation of release of activated CPs formed via neutron activation of components*”. OPEX has shown that such changes can have significant effects in reducing radioactivity, particularly of the important ⁶⁰Co isotope (Refs 44 and 45).
367. Hitachi-GE provides a justification for the application of low-cobalt materials in Ref. 81. This report provides full details of the approach and considerations given to the SS and NBA in UK ABWR. This also includes a detailed comparison to the J-ABWR design. In undertaking this review factors such as previous OPEX, calculations of the amounts of radioactivity produced by activation of in-core components, releases of both radioactive ⁶⁰Co and inactive ⁵⁹Co and material availability issues are considered.
368. Based on this evaluation, Hitachi-GE specify low-cobalt levels ($\leq 0.05\%$) for most of the SS and NBA components in the reactor internals and RPV components, as well as the control rods and low pressure feedwater heater tubes. In addition, two further

special cases exist where even lower levels are specified; the High Pressure Feedwater Heater (HPFWH) tubes are specified to be ultra-low cobalt ($\leq 0.02\%$), as are the fuel parts (\leq [redacted] for SS and [redacted] for NBAs). Collectively, these represent improvements on the J-ABWR design. Hitachi-GE estimates that this will reduce the releases of ^{59}Co (by [redacted] and ^{60}Co (by [redacted])). There is obviously uncertainty in the precise benefits this will give, but it is shown that if low-cobalt specification were to be applied for all components considered, the only changes would be a further reduction in ^{59}Co release by [redacted]. I therefore consider that an adequate case has been made that trace cobalt specifications have been reduced SFAIRP for the UK ABWR. This is shown in Figure 20.



369. While the future licensee may need to keep this conclusion under review, for example if lower specifications can be shown to not be grossly disproportionate at the time of procurement (i.e. $< 0.05\%$), I would consider this to be normal business and I would not expect any further changes to have a large impact.

Minimising Other Metallic Impurities by Material Choices

370. While much effort is rightly paid towards the minimisation of cobalt in UK ABWR, Ref. 81 also provides a justification for the reduction in other metallic impurities in the coolant. Claims are made regarding material improvements made elsewhere, and for other reasons, having benefits in reducing corrosion rates. For example, the use of CS with trace chromium ($> 0.10\%$) for the CFDW system pipework and LAS for the CUW system piping at $> 60^\circ\text{C}$ downstream of the CUW system filter-demineraliser (F/D) should also bring benefits. Ref. 81 provides a description of the main sources of the important metallic impurities, such as iron, nickel and chromium. In general, there is little that can be done to minimise the releases of these materials via the material choice itself, for example by specifying a different grade of SS, as they are the main alloying elements present in the materials used within the plant. These choices are dominated by other safety concerns, such as material integrity issues, and are to a large extent fixed.
371. Some materials are however excluded from the design; copper and copper-alloys are eliminated as far as possible in the SSCs in contact with reactor coolant in the UK ABWR, due to known issues with fuel cladding corrosion. Other elements that may be

prone to activation, such as antimony or silver, are similarly restricted. Overall, I am satisfied that Hitachi-GE have considered this aspect appropriately for UK ABWR.

372. Ref. 81 also provides a useful insight into those SSCs which contribute the greatest releases of metallic impurities to the coolant. While there is some variation between the different elements, several SSCs are shown to dominate in most instances. Notably, these include the CFDW system piping, HD system piping and HPFWH tubes. This is summarised in Table 4, taking data from Ref. 81. Note that cobalt is as detailed in Figure 19, and discussed previously.

[Redacted]

373. As described in the previous section of my assessment (4.3.2.3), Hitachi-GE provided materials selection justifications for all of these SSCs (amongst others). In all of these cases radioactivity was considered as a determining factor. The selection of the materials for these systems is dominated by other concerns (primarily integrity) and minimising the release of metallic species is not the determining factor. I am therefore content that an adequate ALARP justification has been provided for these choices, including the impact in the context of radioactivity. The exception to this is for the HD system piping which is discussed further below.

374. In reality, for the important SSCs given in Table 4, the only system where there is a potential to influence a significant reduction on radioactivity through material choices is for the HD system piping, although even in this case, I accept that demonstrating a definite benefit is difficult due to uncertainties and complexity in the underlying behaviour. In this case the material integrity requirements can, in principle, be met whilst simultaneously reducing the release of metallic impurities. This is because:

- The feedwater iron concentration is significant in determining the behaviour of CPs in UK ABWR (see the following section of my assessment for further details of the impact of iron on radioactivity).
- HD system piping accounts for a significant fraction (up to 25%) of all the iron input into the coolant.
- UK ABWR is a forward pumped BWR design.

- Unlike other parts of the plant, the option to influence radioactivity by selecting a different bulk material class is a viable potential option for the HD system piping; and
 - Other BWRs have chosen HD system piping materials with higher corrosion resistance to those specified in UK ABWR (e.g. LAS or SS).
375. As I have described previously (Section 4.3.2.3), the scope of Hitachi-GE's approach to justifying the materials selection for the HD system piping covers both relevant risks I expect to see – loss of integrity due to FAC and doses to workers. To address the impact of the HD system piping material choice on radioactivity, Hitachi-GE devised a two stage method to estimate the impact different materials selections in the HD system may have on worker doses (Ref. 55). Hitachi-GE claim the first part of the method is conservative and adopts a precautionary approach because it assumes a “worst case” worker collective dose and that the worker dose is directly proportional to the feedwater iron concentration (and hence the material choice). In effect this first stage is an initial screening. This identifies four locations where a materials change may be justified on worker dose grounds. This is through a relatively simple calculation, relating dose and monetary costs directly. Although it is not entirely clear in Hitachi-GE's submissions precisely how this is done, it is clear that very conservative assumptions are used. Despite some reservations, and given the use of demonstrably conservative assumptions, I consider it broadly meets the intent of NS-TAST-GD-005, *Guidance on the Demonstration of ALARP* (Ref. 4). I therefore consider Hitachi-GE's conclusions from this part of the method as being broadly justified, from a radioactivity perspective. I am content that the four identified HD system piping sections are those where materials choices to reduce radioactivity merit further consideration.
376. The final materials selection decision for the identified four sections is based on the outcome of the more detailed second stage of Hitachi-GE's evaluation. It is argued that to implement any changes based on the outputs from the initial (first stage) screening alone would be “too conservative”. The basic approach for the second stage is the same, except that instead of assuming the dose reduction from changing a material is directly proportional to the amount of iron released from that section, it is claimed to be more reasonable to assume a maximum reduction of 2.2% is achieved. In Ref. 55, various technical arguments are presented to try and support that 2.2% is a conservative value. I do not discuss these in detail, as at the core of these arguments is the assumption that UK ABWR will behave in exactly the same way as J-ABWR. The evidence shows this is not necessarily a valid assumption to make. Comparing the KK-6 and KK-7 plants is a case in point. Both are J-ABWRs and are fundamentally the same design, but in terms of the generation and transport of radioactivity, each exhibits different behaviour. Furthermore, Hitachi-GE do not perform the corresponding re-evaluation for the detriments side of the ALARP balance i.e. the “cost” of using a more corrosion resistant material, taking account of indirect factors, such as maintenance. Given Hitachi-GE's choice to invoke quantitative methods, I therefore do not agree that a robust outcome has been arrived at. Notwithstanding these reservations, Hitachi-GE's conclusion is that one of the previously identified four sections is still considered reasonably practicable to change.
377. When the overall case presented by Hitachi-GE is considered, I draw the following conclusions based upon on the outcomes as presented in Ref. 55:
- Despite my disagreement over the robustness of the second stage of Hitachi-GE's case, I am content with the outcome from the first stage, which identified four sections of HD system piping where it may be reasonable to make material changes to reduce radioactivity.
 - From a radioactivity perspective, I agree that changing the Location 30 (see Figure 12) piping from CS is appropriate, but I am not convinced by the choice of the replacement material. It is not clear to me why specifying a slightly increased chromium content in the CS is the most appropriate method to also

- mitigate worker doses, given such an approach is primarily used to mitigate FAC, rather than general corrosion. I therefore agree a change is necessary, but I am not convinced the most appropriate material has been proposed.
- For Location 2-1 and Location 2-2 (see Figure 12), which are large sections of CS piping leading from the first and second stages of the MSR to the FWHs, respectively, Hitachi-GE's stage one evaluation demonstrates that changing these materials purely on dose grounds, is grossly disproportionate. Considering the overall conservatism built into this part of Hitachi-GE's methodology, even without a sensitivity analysis, I accept Hitachi-GE's conclusions are valid. This is also consistent with the evaluation of FAC risks performed (i.e. from a chemistry perspective, estimates for the DO concentration) at these locations. Overall, I am satisfied Hitachi-GE has made an adequate case in GDA for the selection of CS for these sections of HD system piping.
 - For Locations 7 and 8 (see Figure 12), I cannot reach the same conclusion because, as the stage one results demonstrate, it is not grossly disproportionate to implement a change. I am therefore not convinced that the "ALARP position" has been reached for these sections, given my reservations over the second part of Hitachi-GE's evaluation methods.
378. Overall, during GDA, this means I consider Hitachi-GE has provided an adequate case to retain the use of CS in Locations 2-1, 2-2 and 5 (see Figure 12) of the UK ABWR HD system main piping. These are the largest sections which use CS and could therefore have the largest impact on worker doses. Despite this conclusion, I also judge that an adequate justification for the materials selected has yet to be provided for several other sections of the HD system main piping. On the basis of the evidence presented to date, I consider there may be merit in changing the material within these sections to reduce radioactivity. **This has been captured as part of Assessment Finding, AF-ABWR-RC-07.** Also, as concluded in Section 4.3.2.3 of my assessment of the integrity aspects of the case, I also consider these are not robust and materials changes could be warranted on integrity grounds alone. From a chemistry viewpoint, the impact of the materials selected for the HD system is significant. Furthermore, the UK ABWR has an increased design life of 60 years (versus 40 years for the J-ABWR reference plant). **Assessment Finding, AF-ABWR-RC-07**, therefore requires the future licensee to only review, but revise their materials selection criteria for the HD system and apply the revised set to further justify the HD system piping choices, including the impact on radioactivity.

Minimising Radioactivity Deposition by Materials Choices

379. As well as releases, the deposition of radioactivity on out of core surfaces can also be influenced by the material choices. In these cases the operating chemistry also has an important effect, and hence needs to be considered in combination with the materials. The main chemistry mitigation is DZO injection, as detailed later in my assessment, but operating under oxidising or reducing conditions (NWC or HWC) is also important. As with other aspects of ORE minimisation, this focuses on the minimisation of ^{60}Co accumulation in oxide films, given the significance of this nuclide to outage doses. Ref. 80 provides details of the consideration given to such effect by Hitachi-GE, which go towards the factor of "*Minimisation of deposition of activated CPs on out-of-core surfaces*" from Table 2.
380. In summary, the deposition of cobalt on SS surfaces is expected to be larger under HWC conditions than NWC. This is due to the changes in the oxides formed on the surfaces. For CS the same behaviour is anticipated, with HWC conditions expected to favour the incorporation of ^{60}Co . For both materials the addition of DZO can reduce the impact of operating under HWC. The behaviour of LAS is less clear. There is a large degree of uncertainty in predicting such behaviour, with OPEX and experiments often producing contradictory results. This is likely due to the complexities of the

processes, and the fact that many other parameters exert an influence. This demonstrates the difficulties in making material selection choices based on deposition behaviour alone.

381. Despite the difficulties highlighted above, and as described in the previous section of my assessment (4.3.1.3), Hitachi-GE have identified a number of material changes for UK ABWR compared with J-ABWR, associated with the change in operating chemistry. While many of these are influenced by integrity concerns, they also impact on radioactivity behaviour. The most notable examples are the changes to the CUW system piping from CS to LAS or SS, and the changes to the BDL materials. In both instances, Hitachi-GE claims that the chosen combination of operating chemistry and materials will produce the lowest amount of deposited radioactivity, when balanced against other safety concerns. From a radioactivity deposition perspective, given the discussion above and the safety significance of maintaining integrity, I am content that these material changes for UK ABWR are reasonable.
382. There is one system where this choice is less straightforward, namely the RHR system. I describe my assessment for this system below, which was a particular focus of my assessment as part of RO-ABWR-0035 Action 6 (Ref. 25). This was considered specifically by Hitachi-GE in their material selection report which covered this system, Ref. 55. The material integrity related aspects of this choice are assessed in Section 4.3.2.3 of my report. As described there, by applying their materials selection methodology, Hitachi-GE concluded CS is the “ALARP option” for RHR system main pipework. This is the same material as J-ABWR. As previously explained in Section 4.3.2.3, from a materials integrity perspective, this conclusion appears reasonable. However, considering the options available and the overall ALARP balance supporting this materials selection, methods to potentially reduce the deposition of radioactivity also need to be adequately considered.
383. A schematic of the RHR system is shown in Figure 14, in Section 4.3.2.3 of my report. During an outage the RHR system provides cooling and two important chemistry-related effects occur:
- HWC and DZO injection are not in operation and the coolant chemistry becomes oxidising, similar to NWC, and;
 - An increase in coolant radioactivity is often observed, brought about by a combination of mechanical removal of deposited oxides (by flow changes) and an increase in their solubility during the shutdown process.
384. The impact of these effects and the overall significance of materials choices and other available options, on the performance of the RHR system with respect to radioactivity deposition are further highlighted by the following:
- For J-ABWR, RHR system maintenance tasks performed during the outage are amongst the highest dose activities performed by workers (Ref. 83).
 - This is consistent with other BWRs and is one of the main reasons why some operational BWRs have performed chemical decontaminations of their RHR systems during outages (Ref. 43).
 - Other BWRs have taken alternative approaches to mitigate radioactivity deposition in the RHR system by selecting different materials (for example, SS) (Ref. 43).
 - Hitachi-GE specifies three specific operational practices: RHR system clean-up, soft shutdown and low temperature RHR system operation. These are assessed later in Section 4.3.3.5; but they minimise radioactivity deposition, by reducing either the temperature, or concentration of radionuclides present (Ref. 84).

385. The central argument made by Hitachi-GE in Ref. 55 is that there is little difference between the performance of CS and SS in the RHR system, with respect to the deposition of radioactivity. Evidence is presented in Ref. 55, but the breadth and depth of this is very narrow in my view, especially when considering the significance of this claim. While the evidence provided does support what Hitachi-GE claim, predicting the expected deposition behaviour is not straight forward. The other added complication is that differences in design, coupled with different materials selections and operating practices means that a reliable dataset on which to base the radioactivity deposition performance of one material, compared to another, does not exist. Despite this, I am satisfied that Hitachi-GE's intent behind their claim is valid, which in effect is to argue that there is little to distinguish the performance of one material against another, under these circumstances.
386. Hitachi-GE's other important argument relates to the operating temperature of the system. This is why low temperature RHR system operation is seen as a dominant factor in mitigating radioactivity deposition. Based on independent information provided to me by my TSCs (Ref. 43), this appears to be a vital and valid argument. Other evidence is available which clearly shows lower temperatures result in reduced deposition. In addition, it is clear that such practices are not widely adopted by other operational BWRs. This demonstrates the importance of Hitachi-GE's specific claim on this practice as part of the UK ABWR generic safety case. This is assessed further, below.
387. The last point to consider is Hitachi-GE's argument regarding RHR system clean-up (flushing). Information provided to me by my TSCs (Ref. 43) indicates there are a variety of practices for operating BWRs in this regard. Some plants do nothing, and others routinely implement these practices. However, there are instances for BWRs where operational and safety impacts have resulted following connection of the RHR system. For example, at least one US BWR has experienced significant problems in the past following the ingress of excessive CPs from the RHR system. During GDA, in response to RQ-ABWR-1513 (Ref. 49), Hitachi-GE confirmed RHR clean-up (flushing) is formally claimed within the UK ABWR generic safety case. I am therefore content with this position.
388. Overall, with respect to the performance of the material for radioactivity deposition, I am also satisfied with the selection of CS for the UK ABWR RHR system main piping. The evidence to support any changes to materials based on the deposition of radioactivity alone is limited, and often contradictory. This therefore makes the identification and justification of adequate surface treatment techniques and/or operational practices, to achieve reductions in radioactivity deposition in the RHR system, very important, in the overall ALARP balance. I have assessed these aspects for the RHR system in the relevant sub-section and Section 4.3.3.5, respectively, below.

Surface Treatments to Minimise the Generation and/or Deposition of Radioactivity

389. For those SSCs which contribute significant fractions of the metallic impurities into the coolant there may be improvements in the control and minimisation of radioactivity that could be made via treatments of the material surfaces. This could be one way to address the factor identified by Hitachi-GE in Table 2, "*minimisation of the input of metallic impurities into the reactor*". In this part of my assessment, by material treatments I am referring to any treatment which is purposefully applied to change the material surfaces (for example, pre-oxidation or electro-polishing) that can impact on radioactivity behaviour. The effects of the operating chemistry, either during commissioning or normal operations is considered later in my assessment.

390. As well as release, such treatments may also impact on the deposition of radioactivity on surfaces, addressing “*minimisation of deposition of activated CPs on out-of-core surfaces*” from Table 2. In this latter case the candidate locations where this could be applied are wider, and instead should be focussed on those areas of plant where ORE is known to be largest. OPEX is therefore an important input in determining the candidate areas. Ref. 82 is the example which is of most relevance to UK ABWR as it was applied to a plant which re-started after many years offline (i.e. similar to what might be considered for a new plant). This demonstrates the types of treatments, potential locations and benefits that may be achievable by using such techniques appropriately, noting that this in itself was completed over a decade ago and was not as fully implemented as originally intended. Even so, the benefits were significant in reducing worker ORE.
391. Rev. 2 of Ref. 81 contains the final position for surface treatments during GDA. The process adopted by Hitachi-GE is to apply their ALARP methodology used throughout GDA (Ref. 21) to aid in the selection of suitable treatment techniques. To do so, and to focus the process on those SSCs of greatest significance, an initial screening is performed to identify a set of target components. This is based upon a review of those components that contribute to the release of metallic impurities (as per Table 4), the release of activated CPs, or worker doses, Ref. 83. The practicability of applying surface treatments is also considered, with some potential target components excluded at this stage due to factors such as complexity in application. The selection of components is adequate, encompassing a large fraction of the contributions to each of the three criteria and a range of different materials. The list of ten components was subjected to a review by Hitachi-GE, with four detailed examples provided in Ref. 81.
392. In addition, it is also important that a suitable range of treatment options is considered. A large part of the response to RQ-ABWR-1455 (Ref. 49) provided an OPEX review. It is acknowledged by Hitachi-GE that further work would be needed from the future licensee in this area, but the basis for informing decisions during GDA is sound. Seven surface treatments are considered as “representative”. I consider this a reasonable basis for the review at this stage of the project, although note that further techniques may prove to be viable at the more detailed site-specific evaluations.
393. In order to determine the “ALARP option” for each component, the surface treatment options are scored relative to the J-ABWR reference design against a set of criteria, including factors such as fabrication, OPEX, operation and maintenance. I have not assessed this process in detail, as there would be limited value in doing so at this stage. However, I have assessed the conclusions and the key rationale cited by Hitachi-GE. On this basis I would consider a similar methodology appropriate for application by a future licensee. I specifically note the consideration given to sensitivity studies in the examples presented by Hitachi-GE, which identify options for the future licensee to review further.
394. The outcome from Ref. 81 is that, of the ten target components evaluated, three have identified different surface treatment techniques to the J-ABWR reference design, while seven remain unchanged. The identified changes are given in Table 5. Significantly, this includes changes to both the HP FWH tubes and RHR system piping. The former is the most significant source of metallic impurities to the coolant, releasing large amounts of chromium, manganese, iron, nickel and cobalt (as shown in Table 4). The RHR system piping is also a major contributor to worker doses during outages, as is the CUW system pump.

[REDACTED]

395. All of these changes do represent improvements to the J-ABWR design and target some of the most significant components in terms of ORE. Ref. 81 presents some evidence for the potential scale of benefits that these could achieve, but this is often limited and incomplete. The effects could potentially be significant, with reductions in deposition of over 80% reported. Importantly, any identified detriments are negligible, and even in the worst case the release or deposition behaviour would not be worse than the J-ABWR reference design. The precise scale of benefits expected in UK ABWR is therefore difficult to predict, as is the longer term behaviour over multiple fuel cycles. This represents a significant weakness in the process applied by Hitachi-GE during GDA and directly influences the conclusion reached. This is exemplified by the four detailed examples provided, where the choice of surface treatment is often subjective and could be challenged mainly as a result of the interpretation of the limited OPEX base. In effect, the future licensee will need to significantly expand on this aspect of the case in order to undertake a full and complete ALARP assessment.
396. To be able to take sufficient confidence in what has been done during GDA, despite these deficiencies, it is important to be able to conclude that the choice of surface treatments does not impact on the material choice. In other words, that the overall “ALARP option” when material and surface treatment options are combined remains the same. This is because not all surface treatments can be applied to all materials. I judge that this is most relevant to the RHR system piping. As described earlier, this system contributes significantly to worker ORE and there is merit in considering if this risk can be further minimised using a different combination of materials and surface treatments. The case provide by Hitachi-GE does not consider this more holistic approach. For example, no consideration or justification is given to whether SS with a surface treatment would be a better option for the RHR system piping. This is a further weakness in the case presented during GDA. However, based on the information contained within other aspects of the generic safety case, as I have already concluded in the previous sections, I am content that the hazard posed by deposition or release of radioactivity is not sufficient to merit a material change, when balanced against other hazards (mainly integrity). The RHR system piping is a case in point, where although a change to SS with electro-polishing would likely reduce deposited radioactivity compared to both the J-ABWR (CS with blast treatment) and UK ABWR cases (CS with oxidation), the increased risk of SCC this would incur would seem to outweigh this.
397. In summary, I therefore conclude that:
- Hitachi-GE has implemented important changes to surface treatments for the UK ABWR design, which should provide tangible improvements in the generation or deposition of radioactivity.
 - On the basis of the evidence provided to date, I do not yet agree that the “ALARP surface treatment techniques” have been demonstrated.

- I am satisfied sufficient information has been provided during GDA to suggest that changes to material choices are unlikely to be justified, for the sole reason of applying alternative surface treatments to improve their performance, with respect to the generation and/or deposition of radioactivity; and
 - Despite identifying several gaps in the case provided by Hitachi-GE during GDA, I am satisfied the future licensee can take this matter forwards. Implementing alternative materials surface treatment techniques (for radioactivity performance) are unlikely to result in fundamental changes to the UK ABWR generic design.
398. On the balance of materials degradation concerns, materials choices for some UK ABWR SSCs result in a much more restricted set of available options to improve their performance, with respect to the generation and deposition of radioactivity. This means in the overall ALARP balance, implementing further measures to improve this aspect of their performance, is fundamental. Furthermore, I have concluded during GDA that in this regard, more could be done in the future. As further information and OPEX becomes available in the future, and because some of these decisions are ultimately for a licensee to make, I expect a future licensee to revisit the generic case made by Hitachi-GE during GDA, to comprehensively justify appropriate, reasonably practicable surface treatment techniques are specified for UK ABWR SSCs, to ensure relevant risks are reduced SFAIRP. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-10: During GDA, Hitachi-GE developed and partially implemented an approach to evaluate if using surface treatment techniques could reduce both the generation, and deposition of radioactivity within certain UK ABWR plant systems. These techniques can result in significant benefits. For GDA, the approach was applied to several example components. The approach was greatly simplified and did not consider a complete review of relevant operating experience (including over multiple operating cycles), nor did it consider the full extent of the associated benefits and detriments associated with each technique. The final selection of surface treatment techniques will also depend on licensee specific choices and decisions. The licensee shall justify the surface treatments to be applied to UK ABWR structures, systems or components, which make significant contributions to the release of metallic impurities, or doses to workers, reduces relevant risks SFAIRP.

Reducing Radioactivity by Material Choices Summary

399. The J-ABWR reference design has already implemented a number of lessons learned from worldwide practice regarding the impact of material choices on radioactivity. The starting point was therefore already well developed. Despite this Hitachi-GE needed to put a large effort into reviewing and justifying the material choices, given the change in operating chemistry, developments since the last J-ABWR plant and the requirement to demonstrate risks are reduced SFAIRP in the UK. This has resulted in a number of further improvements, including reductions in high-cobalt alloy usage, lower specifications for trace cobalt content in some components, changes to materials of construction and refinements in surface treatments.
400. Despite these advances there still remain some aspects of the design and safety case where further improvements may be possible before any UK ABWR is constructed. These relate to the use of high-cobalt alloys in the FMCRDs and valves, and the surface treatments to be applied. I have raised Assessment Findings, **AF-ABWR-RC-09 and -10** on these topics. I am content these can be resolved by a future licensee without undermining the fundamental UK ABWR generic design.

4.3.3.4 Reducing Radioactivity by the Operating Chemistry

401. Hitachi-GE identify two main purposes for chemistry control in the RCS:

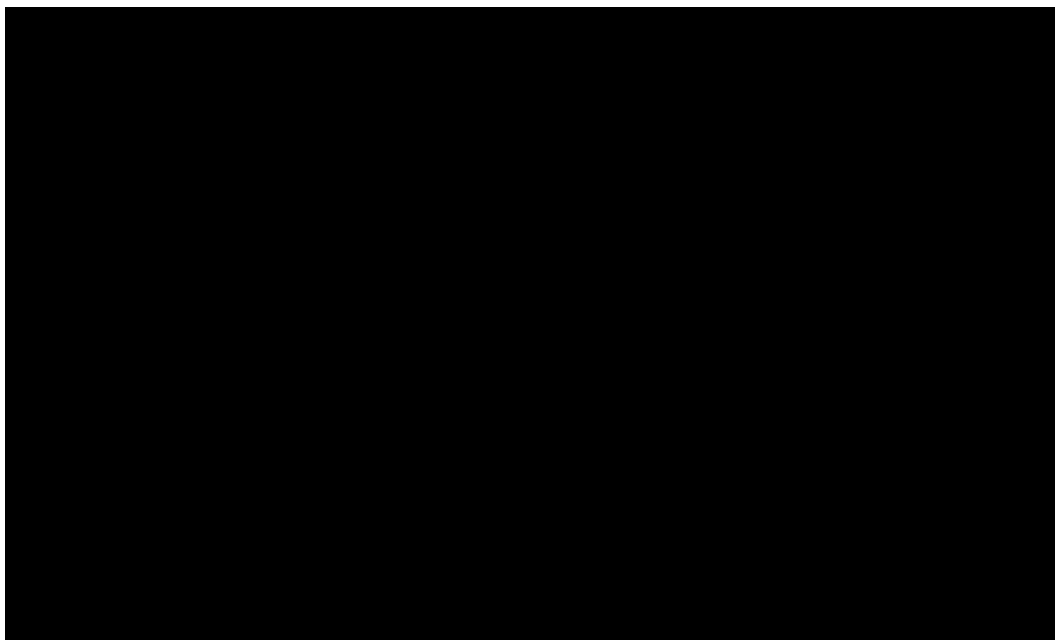
- To maintain the integrity of the fuel, structures in the reactor pressure vessel and the reactor coolant system boundary.
 - To ensure that generation of radioactivity and its impact on radiation protection, environmental discharges, radioactive waste management, decommissioning and in fault conditions is managed.
402. The choice of operating chemistry can therefore have a direct influence on not only the quantity of radioactivity produced within an operating BWR, but also its behaviour including its distribution around the plant under different operating modes. This impacts several important factors defined by Hitachi-GE in Table 2.
403. The operating chemistry regime selected for the UK ABWR is described in, *UK ABWR Reactor Chemistry Safety Case: Demonstration that the Primary Cooling System Operating Chemistry Reduces Risks SFAIRP* (Ref. 26), and, *Topic Report on the Water Chemistry Regime ALARP Assessments on the Iron Control at Power, Commissioning, Start-Up and Shutdown operational modes: Part 1 – Main Justification* (Ref. 27), as assessed in Section 4.3.1 of my report.
404. Although these reports justify that the chemistry regime for the UK ABWR reduces the overall risks SFAIRP, they do not provide the detailed arguments and evidence to support the radioactivity related aspects. During GDA, this was specifically requested as Action 5 of RO-ABWR-0006 (Ref. 25). *Topic Report on Source Term Reduction by Operating Chemistry* (Ref. 85) includes information on the evolution of BWR operating chemistries and how this influences radioactivity, including the effects of various chemistry parameters. The report itself considers all modes of normal operation, including start-up and shutdown conditions and commissioning. Most of the detailed evidence to support the claims and arguments presented in Ref. 85 are often found in other supporting references (Refs 29 – 32, 36 and 38).
405. In this part of my assessment, I assess the impact of the operating chemistry on radioactivity whilst at-power first, before moving on to consider any differences during other operating modes.

Impact of HWC and OLNC on Radioactivity

406. The main decision regarding the operating chemistry in UK ABWR is the adoption of HWC and OLNC. The impact of this choice on radioactivity is considered in Ref. 85, which summarises the detailed evidence provided in *Topic Report on Noble Metal Chemical Addition (NMCA)* (Ref. 32). I consider these choices together, as they are very much related. As described earlier (Section 4.3.1.2) hydrogen is injected into the feedwater in order to reduce the oxidising nature of the coolant and mitigate SCC, mainly for the reactor internal and recirculation piping made of SS. This means that the bulk of the coolant is under reducing conditions, and soluble species can change to more reduced forms. For ^{16}N , which is produced from the ^{16}O naturally present in the water, the chemical form shifts from predominately nitrates, which are non-volatile, to more volatile forms, such as nitrogen oxides and ammonia. The net effect is that more of the ^{16}N produced in the reactor is transferred to the steam phase and transported through the turbine systems. The amount transferred is proportional to the amount of hydrogen added, but does plateau at higher levels. For some plants, ^{16}N can increase dose rates in the main steam by up to five times in the turbine building of BWRs that change from NWC to HWC (Ref. 44), having a direct impact on worker ORE. The same is true for ^{13}N , but the lower concentrations and radiation energy make it less important for worker doses.
407. The noble metals introduced into the reactor act as efficient catalysts for the recombination of hydrogen and oxygen to water, meaning that less hydrogen needs to be added to achieved the required SCC mitigation benefits. HWC and OLNC therefore

directly contribute towards “*minimisation of ^{13}N and ^{16}N transfer to the steam phase*”, identified by Hitachi-GE in Table 2.

408. The general benefits of noble metals in reducing the impact of HWC are well established. Both Refs 85 and 32 provide an adequate basis for these conclusions, alongside OPEX which demonstrates the scale of effects expected. These are consistent with the work undertaken by Hitachi-GE to respond to RI-ABWR-0001, as detailed in my assessment report (Ref. 79). An important factor that is less clear is the prediction of whether the design and operations of UK ABWR would allow hydrogen additions to be minimised to also allow the impact on dose rates to be minimised. There are several factors which determine the amount of hydrogen that needs to be added, many of which are dependent upon the precise core design. An important piece of the evidence provided by Hitachi-GE in this regard is the HWC “mini-test” that was conducted at KK-7 (Ref. 86), which demonstrated ABWR behaves in a similar way to other BWRs, in response to HWC. This response is shown in Figure 21. This shows that for feedwater hydrogen levels up to around 0.3 ppm there is no impact on main steam line dose rates. Above this, the dose rate increases sharply, before plateauing above around 1 ppm.



409. As described earlier in my assessment (Section 4.3.1.2), Hitachi-GE have also undertaken specific analysis for the UK ABWR core design in Ref. 47. The comments and conclusion from my earlier assessment of this report remain relevant here. However, it is also clear that this analysis supports the arguments that the behaviour of KK-7 regarding hydrogen effects on ^{16}N behaviour would be consistent with that expected in UK ABWR. I am therefore content that a sound basis exists on which to control and minimise the impact of HWC and OLNC on ^{16}N behaviour.
410. Operating under reducing conditions may also impact on the behaviour of other radionuclides. Iodine is a case in point, as it can form many volatile species depending upon the conditions. Hitachi-GE consider this as part of Ref. 85, providing plant OPEX which demonstrates that there is not expected to be any significant changes in the behaviour of iodine in UK ABWR, compared to other BWRs, irrespective of the adoption of HWC. This is consistent with the assumptions used by Hitachi-GE in justifying the source terms for UK ABWR (Ref. 79) and in their responses to my questions regarding the behaviour of iodine in BWRs under normal operations (RQ-ABWR-0742 and RQ-ABWR-1340 (Ref. 49)), notably in the report, *Iodine Behaviour During Normal Operations* (Ref. 87). This matter was also considered independently

by my TSC in Ref. 88. The overall conclusion from the case presented by Hitachi-GE for the behaviour of iodine during normal operations is that it is of little safety significance. They argue that compared to how other radionuclides will behave under HWC conditions, the behaviour of iodine in UK ABWR will be much less pronounced and broadly similar to its behaviour under NWC conditions, owing to the adoption of OLNC. In the context of reducing radioactivity, I am content with what has been presented.

411. The change to reducing conditions also affects the solubility of a number of the CP nuclides. For most of these this has no observable effect, as the concentrations are much below the solubility limits. Most CP nuclides will however show an increase in their concentration in the coolant as a result of HWC and OLNC, as a result of changes in their deposition behaviour rather than solubility effects. Such changes are within the typical plant to plant variations observed (factor of two) and are therefore relatively minor. The impact of this on the clean-up system performance is also considered by Hitachi-GE and determined to be bounded by the design. The exception to this behaviour is ^{51}Cr where it is observed that HWC plants exhibit a large reduction in soluble radioactivity, due to the change to an insoluble species under these conditions. This is considered in Ref. 85, and further evidence is provided as part of Ref. 32. For power operations I am therefore content that these consequences are considered, and I assess their impact during shutdown periods later.
412. An important impact of adopting HWC and OLNC is the affinity of surfaces to incorporate cobalt, including ^{59}Co and ^{60}Co . This is heavily influenced by the nature of the oxide films present, a parameter which itself is influenced directly by the operating chemistry environment and whether it is reducing or oxidising. This affects both the fuel and SS surfaces, in particular. The actual behaviour at any given point in the plant is further complicated by the fact that the precise conditions can vary as the oxygen to hydrogen molar ratio changes. In simple terms, under reducing conditions the SS surfaces would be expected to incorporate higher levels of ^{60}Co , all other factors being equal. This could increase deposition by around a factor of two. Evidence to support this behaviour is presented by Hitachi-GE in Refs 32, 81 and 85. This is consistent with both plant OPEX and experimental data (for example, Refs. 44 and 45).
413. The deposition behaviour on the fuel surfaces is similarly affected. The further complication in this case is that the coolant boils and this boiling also removes the soluble gases from the coolant. This means that the different parts of the fuel will see differing conditions, changing from reducing through oxidising through to steam as the coolant rises up the core. This affects both the amounts and nature of the deposits that form on the fuel cladding. There are also many other factors that impact on this, including the feedwater iron input, DZO injection, the presence of noble metal on the deposits, the ratio of iron to other transition metals (e.g. nickel, chromium) plus other minor factors. This is the reason why there are differences in plant OPEX regarding the absolute effects of HWC and noble metals on CP behaviour. What is consistent is that, as with other surfaces, the oxide forms of the deposits have a direct impact on the amount of ^{59}Co and ^{60}Co that is incorporated into the fuel deposits (affecting both deposition and release rates). This is a key factor in determining radioactivity behaviour in the whole plant affecting “*minimisation of the release of activated CPs from the fuel surfaces*”, identified by Hitachi-GE (Table 2).
414. Refs. 80 and 85 provide a detailed explanation of Hitachi-GE’s understanding of the process and their expected behaviour for UK ABWR. The behaviour of CPs in UK ABWR was also modelled by Hitachi-GE in Ref. 89. Given the uncertainties in this behaviour, my TSC also reviewed the Hitachi-GE modelling and performed their own independent analysis in Ref. 60. This latter report also provides an independent examination and view of the explanation, evidence and impact of HWC and noble metals on fuel deposits. In general, these are consistent regarding the overall behaviour, although there are some differences in the underlying details as to why this

may occur. In summary, the oxide on the fuel deposits will consist of haematite, magnetite, nickel oxide and nickel ferrite, plus other minor species. The relative amounts will depend (mainly) on the ratio of nickel and iron in the feedwater as well as the conditions (reducing or oxidising). While a simplification, the reducing conditions will tend to lead to more magnetite, while oxidising conditions favour haematite. The former has a higher affinity for cobalt and hence the net effect is that more ^{60}Co is produced and released to the coolant.

415. The impact on fuel deposits is the area where the precise impact of HWC and OLN (alone) is less certain. This is further complicated by the fact that both DZO injection and iron control in UK ABWR will affect the overall behaviour. Hitachi-GE therefore considers the combination of these effects with the overall aim of trying to tailor the fuel deposits to contain the least amount of cobalt, but also to retain that cobalt as strongly as possible. I consider this to be a sensible aim. In the context of HWC and OLN, the case made by Hitachi-GE is therefore reasonable, given the uncertainty in this topic, and is consistent with the latest understanding.
416. Ref. 85 also summarises the potential impacts of over or under-dosing of hydrogen, or noble metals, on radioactivity. For hydrogen the effects are the same as those given in Figure 21; the effects of overdosing can be significant. For UK ABWR, Hitachi-GE proposes noble metals are applied using OLN technology. This has a number of impacts on radioactivity, but they are transient behaviours only seen during, or shortly after (i.e. for a few days) after platinum injection. These include increases in CP concentrations in the reactor water, increases in ^{16}N transfer to the steam and increases in activation products produced by the chemical additions (mainly ^{24}Na and ^{199}Au). All of these effects were considered in response to RI-ABWR-0001 (Ref. 79). The absolute values for these increases are uncertain, but can be considered bounded within the typical plant to plant, or cycle to cycle variations observed in BWRs and are therefore of minor importance.
417. ORs related to HWC and OLN control are defined in Ref. 20 and also summarised in Annex 5 of my report. From a radioactivity perspective, I am content with these ORs. They clearly relate to the arguments and evidence presented in the UK ABWR generic safety case and represent suitable controls. Only the upper limit on feedwater hydrogen is directly related to radioactivity concerns. The remainder are mainly driven by SCC minimisation, which has more conservative limits.

Impact of DZO on Radioactivity

418. As shown in Figure 5, almost all US BWRs now inject DZO, as do a number of European BWRs, including some which continue to operate under NWC. DZO injection was first implemented in BWRs in 1986 to mitigate the increase in shutdown dose rates that resulted following the adoption of HWC.
419. Hitachi-GE proposes DZO injection will be applied for UK ABWR to address the following factors identified in Table 2: “*minimisation of deposition of activated CPs on out-of-core surfaces*” and “*minimisation of the release of activated CPs from the fuel surfaces*”. The impact of DZO is considered in Ref. 85 and further detailed evidence is presented in *Topic Report on Zinc Injection* (Ref. 31). Collectively, these reports provide the justification for adopting DZO injection and evidence for its impact on radioactivity (amongst other hazards).
420. DZO is composed of zinc which has a much reduced proportion of ^{64}Zn (<1% compared to 48.6% in natural zinc). ^{64}Zn can become activated to ^{65}Zn . Although a shorter lived and weaker gamma emitter than ^{60}Co , it can still result in significant ORE and impact radwaste, when present at elevated concentrations. Hitachi-GE contends that the beneficial effects of using DZO are judged to far outweigh the minor negative

impact resulting from ^{65}Zn production. I agree with this conclusion, and am content that the use of DZO reduces radioactivity SFAIRP, from this source.

421. The technical basis for DZO injection is well understood in the nuclear industry, and is documented and supported in many publications (including Refs. 44 and 45) so is not repeated in detail here. The UK ABWR generic safety case is consistent with this. In summary, DZO is incorporated into oxide films on SS and fuel cladding surfaces in preference to cobalt, thereby minimising ^{60}Co generation and deposition.
422. It is useful to consider the impact of DZO on out-of-core and fuel deposits separately. For out-of-core surfaces, the materials choice and deposition behaviour of DZO are inextricably linked. The benefits of DZO for SS surfaces are much more pronounced, as supported by a wide evidence base, including laboratory and plant data. For CS and LAS there less data are available. Hitachi-GE contends that there will still be benefits for the CS components in UK ABWR, although not to the same extent as for SS. The evidence to support this assertion is not particularly strong, but I am content that the overall net benefits for DZO injection are clear. The magnitude of the impact of DZO can be significant, as exemplified by Figure 22. Note that in this figure, HWC does not include OLCN, hence the amount of hydrogen added is much larger than proposed for UK ABWR (and hence its effects on CPs).

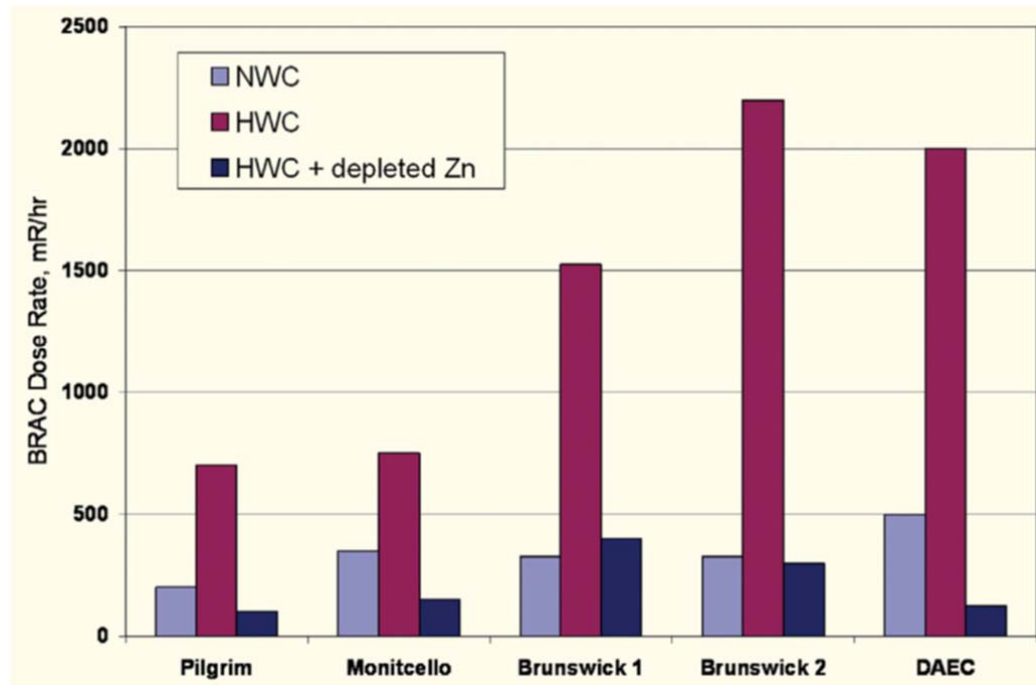


Figure 22: Piping dose rate experience for plants which moved from NWC, to HWC then HWC with zinc injection (Ref. 44)

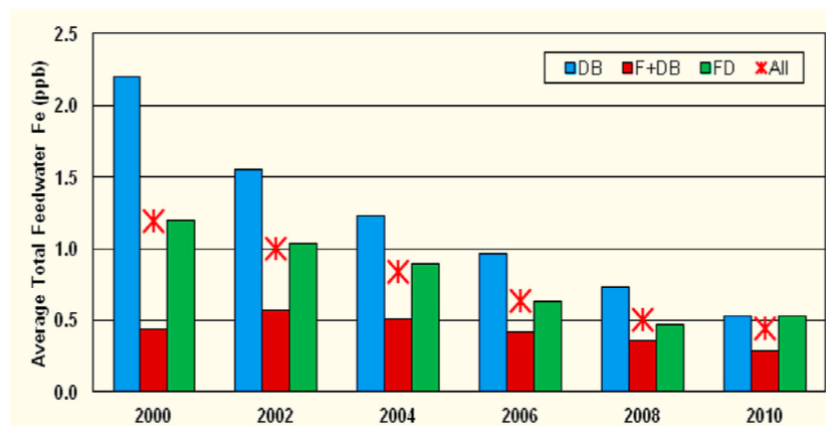
423. The effects of zinc on fuel deposits are similar to those seen in out-of-core surfaces. DZO is incorporated into the oxide structure, restricting uptake of cobalt. In addition, control over the operating chemistry (in particular iron) can favour the formation of zinc ferrite spinels which are less soluble and are beneficial in reducing ^{60}Co activity release from fuel deposits. This achieves the aim of both minimising the cobalt in the fuel deposits, but also minimising that which is produced from being released to the coolant. I am satisfied with the arguments and evidence presented by Hitachi-GE for this aspect.
424. DZO also affects the concentration of ^{60}Co in the reactor water. An equilibrium exists between the cobalt present in deposits and in the reactor coolant. As described above,

DZO reduces the amount of ^{60}Co produced on the fuel, restricts its release to the coolant and also minimises its uptake into surface oxide films. Hitachi-GE argues that the net effect of DZO on the ^{60}Co coolant concentration is to reduce it, by a factor between three and five. This is based on plant OPEX, by comparing plants which operated with HWC before and after DZO injection. I consider it is difficult to use these data in this way, as other effects may be masking the true impact of DZO. In particular, it is not clear if the values compared represent a true equilibrium condition (i.e. the observed effects may also include some impact of oxide restructuring, for example). In common with many aspects of BWR chemistry, plant data needs to be interpreted with care, as many effects interplay and distinguishing single effects is difficult. I am therefore not convinced that the scale of the benefit suggested by Hitachi-GE will be realised. However, I am content that the reactor water ^{60}Co concentration may indeed be lower, but even if it is not, I would not expect it to increase significantly. Significantly, this does not undermine the source terms defined for UK ABWR, as the underlying data used there are for many cycles of operation, hence equilibrium has already been reached. This means the potential equilibrium effects described above are not overly significant (see Ref. 79 for further details).

425. Similarly, it is suggested that DZO injection will lead to a reduced activity in spent CUW system ion exchange media (all other factors being equal). This is based on a reduced ^{60}Co reactor water concentration. As above, I am less certain about the true impact of this claimed effect. In any case, I would not expect the impact to be particularly significant. Hitachi-GE also demonstrates that the additional burden of DZO, which will be removed by the CUW system ion exchangers, is negligible compared to other ions.
426. The effects of adding too much or too little DZO are considered in Refs. 31 and 85. There are no additional impacts on under-dosing DZO identified, aside that the benefits described above would not be achieved. In the extreme, this would be significant in terms of ^{60}Co deposition on piping and hence ORE, as evidenced by early plants which operated under HWC, but without DZO (exemplified by Figure 22). However, these effects would occur on long timescales and would require extended periods without DZO injection in operation. Shorter periods would also be compensated to some extent, by the DZO inventory that would become established within the plant over time. The main detriment of elevated DZO concentrations is related to the impact on fuel deposits. This is assessed in Section 4.3.4.2 of my report.
427. Overall, I am satisfied that the technical case for DZO injection to UK ABWR is compelling, from the perspective of minimising the impact of radioactivity. There are clearly significant net benefits in adopting DZO injection, despite some reservations about the scale of the benefits that may be achieved.
428. To control DZO injection, Ref. 20 defines several relevant ORs. They are presented in Annex 5 of my report. The combination of specifying a minimum value in the reactor water and limiting the feedwater concentration are sufficient for ensuring that DZO injection is controlled. These values are consistent with existing plant experience (for example, Ref. 45 and later revisions thereof). From a radioactivity perspective, I am content with these ORs. However, I do note that other parameters have been controlled in US BWRs which implement HWC, OLN and DZO. This includes the ^{60}Co to soluble DZO ratio. This is discussed in Ref. 31 by Hitachi-GE, but no firm conclusions are reached regarding its applicability to UK ABWR. There may be benefits in specifying controls of this nature, whereby the concentration of DZO is related to other parameters of relevance (for example ^{60}Co or iron). I consider that this does not impact on the suitability of the existing ORs above, but should be reviewed by the future licensee. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

Impact of Iron Control on Radioactivity

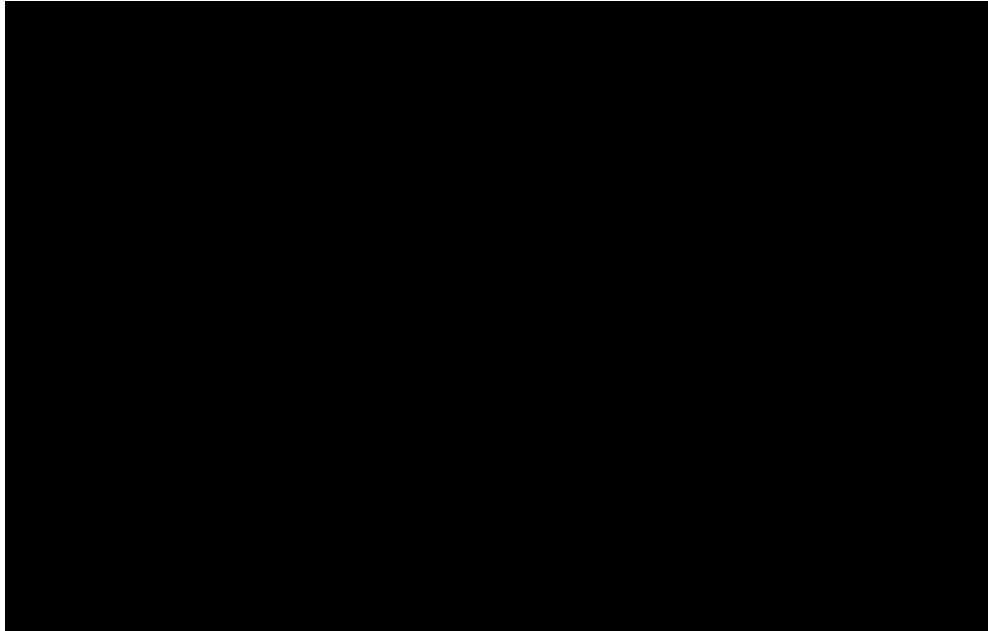
429. Iron is the major component of the deposits that form on the fuel cladding, and as described earlier, the form of this deposit plays a crucial role in determining the tendency of the fuel to retain CPs, or otherwise. This is therefore directly related to “*minimisation of the release of activated CPs from the fuel surfaces*”, but also impact on the “*minimisation of the input of metallic impurities into the reactor*”, from Table 2.
430. The impact of iron is considered in Ref. 85 and further detailed evidence is presented in the “*Topic Report on Fe Concentration Control in Feedwater*” (Ref. 29). Ref. 85 provides a concise summary of the case made by Hitachi-GE to justify that their approach to iron control is adequate. The approach to be adopted was first considered as part of the overall operating chemistry choice in Ref. 26, which reviewed a number of potential options and identified the preferred approach. This is discussed in Section 4.3.1.1. The supporting topic report is therefore somewhat different to a number of the other chemistry related topic reports, in that its main focus is more related to justifying this chosen option. I would therefore consider it less objective and evidential than other aspects of the safety case.
431. Intuitively, the ideal situation would be to have zero iron input into the reactor, but practically this isn’t achievable. As most of the iron that enters the reactor ends up on the fuel cladding, as deposits, minimising this would bring obvious benefits, primarily to fuel cladding integrity but also to radioactivity control and minimisation. The main source of iron to the reactor water is via the feedwater. CS and LAS components and piping in the CFDW system release iron-based CPs which are fed forward to the reactor, unless it is removed by the CPS. In UK ABWR the high temperature heater drains are forward pumped and can also be a source of significant iron input. Minimising iron input via material choices was assessed earlier in my report (Section 4.3.1.1). The reality is therefore that there will always be some level of iron input to the reactor, but the precise concentration of this is extremely difficult to predict before operating the plant. It is also very likely to change over the course of operations, certainly within the early cycles when components and surfaces are passivating and corrosion rates stabilise.
432. Ref. 32 provides information on plants which operate with “ultra-low” iron levels. This refers to plants which have less than 0.1 ppb in the feedwater and in effect are trying to achieve zero iron input. Many US and European plants have adopted this approach, and certainly they strive towards minimisation of iron to the extent that is practicable. This is exemplified in Figure 23, which shows that most plants now operate with well below 1 ppb of iron.



Key: DB – Deep Bed demineraliser; F+DB – Filter and Deep Bed demineraliser; F/D – Filter Demineraliser; All – average for all types

Figure 23: Feed water iron levels in the US BWR fleet from 2000 to 2010 for different CPS designs (Ref. 44)

433. Hitachi-GE also provides OPEX from Japanese BWRs which attempted this strategy. The overall conclusion is that operating with very low concentrations of feedwater iron carries the risk of increasing the coolant concentrations of ^{60}Co , potentially by up to an order of magnitude. This conclusion is mainly based on the Japanese experience. The US data is less conclusive, as only some plants exhibit this behaviour. It is acknowledged that further work would be needed to fully support this position. The technical rationale for this behaviour relates to the nature of the oxide films formed on the fuel; if too little iron is present in the feedwater, the films may become deficient in iron, meaning that the oxide phases that form may be more soluble and therefore will release more ^{60}Co to the reactor coolant. This suggests that it is not the absolute value of iron that is important, rather how it relates to other metallic impurities that enter the reactor. This may be particularly relevant for UK ABWR where the early cycles of operation may be different than the eventual steady-state reached. All of the OPEX considered by Hitachi-GE is for mature plants.
434. The case made by Hitachi-GE for UK ABWR is therefore to operate with “optimised” iron control. This means operating within a defined control band, towards the lower range of iron concentrations. Hitachi-GE expect UK ABWR to be able to operate with very low levels of iron, and should this become too low, they would expect the plant would operate to by-pass the CF, whereby some portion of the condensate does not pass through the CPS, resulting in feedwater with a higher iron concentration.
435. The rationale for this approach is that Hitachi-GE is attempting to “oxide tailor” the fuel deposit, with the aim of minimising ^{60}Co production whilst also maximising its retention within the oxide layer. More specifically this requires the production of a thin, tenacious inner oxide layer which contains minimum cobalt, but holds that cobalt tightly (aided by DZO injection) and the minimum quantity of loose outer oxide. This approach is based upon the review of OPEX provided in Ref. 32. Theoretically, what is proposed appears reasonable. However, the problem is that the evidence to support it is not convincing as it is based on multiple correlations and extrapolations from plant data. In this data several effects often occur simultaneously, so ascribing a particular change to iron control alone can be difficult. It is also evident from Ref. 32, and elsewhere (e.g. Ref. 44) that it is not solely iron that is of importance. It is clear from the TR that controlling iron within a certain range is only useful if the other metallic impurities are controlled (or at least understood), especially nickel which is produced from SSs. Ref. 32 describes a number of approaches adopted in Japanese BWRs to minimise radioactivity through iron control. These include iron to nickel ratio control (greater than two) and also operation with excess nickel in the feedwater. The success or otherwise of these approaches is very much plant dependant.
436. Ref. 32 does not make reference to the experience at KK-6 and -7 regarding iron control. This is reported in Ref. 86 and describes OPEX from both plants, which coincidentally operated with different iron control strategies. KK-6 operated under ultra-low iron control, whereas KK-7 operated with iron to nickel ratio control. Applying this directly to UK ABWR is difficult given the operating chemistry and design differences; however it is a very useful comparison given that KK-6 and -7 are identical plants. The impact on reactor water ^{60}Co concentrations is shown in Figure 24, from which it is clear that KK-7 is notably lower. It is not possible to undertake further comparisons, such as RHR system dose rates, as other differences between the plants make these impossible (e.g. alkaline pre-filming was applied in KK-7, but not KK-6).



437. Ref. 32 also considers the possible impact of other operating chemistry parameters on iron control, namely HWC, noble metals and DZO injection. Arguments are made that while the chemistry choice does not affect the feedwater, it will impact on the generation of nickel and other metallic species from the RPV and internals, due to reduced corrosion rates. Further arguments are made regarding the fuel deposits, stating them to be *“basically the same for NWC and HWC”*. I consider this an oversimplification, and inconsistent with other aspects of the safety case. Hitachi-GE also acknowledge that DZO may impact on iron control, specifically because it will react with some of the iron (thereby altering the iron to nickel ratio). The report concludes that the optimum iron concentration needs to consider both the nickel and DZO concentrations in the feedwater, as well as considering feedback from the ^{58}Co and ^{60}Co concentrations in the reactor water. It is acknowledged that the absolute value will need to change during the early cycles of operation. To account for these Hitachi-GE propose a flowchart to determine the “optimum” iron control value, which would be used by the operator to continuously optimise operations.
438. Overall, I am satisfied Hitachi-GE have progressed iron control for UK ABWR sufficiently for GDA. The importance of this parameter to radioactivity is clear in the generic safety case and supporting documents. Some uncertainty remains over the precise controls that may need to be applied to reduce radioactivity SFAIRP, using this parameter, but the fundamental principles the future licensee will need to apply are transparent. However, this should be an area of further work by the future licensee, to provide a more consistent, structured, evidentiary and objective UK ABWR safety case, in this regard. Similarly, an important assumption in the GDA case is that should feedwater iron become too low, CF by-pass operations would be implemented. This is clearly a viable option, having been implemented on other BWRs, including several ABWRs. However, no justification is provided by Hitachi-GE as to whether this is the “ALARP option”. Other BWR plants have used specific iron injection systems. For GDA, I am content with this assumption given other means would be straightforward to implement, but consider that this will need to be justified by the future licensee considering the significance implementing adequate control of feedwater has over radioactivity and fuel integrity related risks. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-11: The UK ABWR generic safety case places key claims on the importance of adequate feedwater iron control for minimising radioactivity and risks to fuel integrity. During GDA, ONR’s chemistry assessment established these aspects of

the generic safety case, as presented, are complex, inconsistent, lack evidence, and are not objective. The licensee shall review, update and supplement the UK ABWR safety case and supporting evidence on this topic. This shall include specifically justifying the impact feedwater oxygen injection has on radioactivity, and provide a proportionate justification for the decision on which approach will be implemented during UK ABWR operations; to practically control the feedwater iron concentration and ensure relevant risks are reduced SFAIRP.

439. Several ORs relating to iron control are specified in Ref. 20. They are presented in Annex 5 of my report. Collectively, I am not content they represent an adequate set of ORs to be able to implement adequate iron control. As described in my preceding assessment, there are several important aspects of the generic safety case presented by Hitachi-GE which are not reflected in the output as limits and conditions. These include:

- there is no lower limit on feedwater iron;
- the upper limit on feedwater iron may be too high (although it is acknowledged that this relates to fuel integrity concerns);
- there is no reflection of the importance of the relationship between iron and nickel; and
- there is no link to the concentration of DZO injected.

440. Considering the overall importance of iron in maintaining adequate control over radioactivity in the plant, I expect the future licensee to review and revise the iron-related ORs defined in the UK ABWR safety case, to address the gaps identified during my assessment of the generic case. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

Impact of Feedwater Oxygen Injection on Radioactivity

441. As previously explained, UK ABWR is designed to operate using oxygen injection into the CFDW system to mitigate the risk of FAC. The impact of feedwater oxygen injection on radioactivity is considered by Hitachi-GE in Refs 80 and 85.

442. Mitigating FAC in the CFDW system will also impact the generation of iron-based CPs. However, the concentration of oxygen required to mitigate FAC is too small to have an effect on reactor water oxygen concentrations, because the radiolysis of water in the core is the major source of the oxidants present in the RPV. Therefore, oxygen injection does not influence the fuel oxide structure directly, but does play an important role in the transport of particulate iron-based CPs, to the RPV. This therefore impacts directly on "*Minimisation of the input of metallic impurities into the reactor*", from Table 2. Ref. 85 provides arguments to support these claims, but does not include any evidence. While I am content with the claims, and I agree they are logical, I expect the safety case documentation to include the necessary evidence, in this regard. **This has been captured as part of Assessment Finding, AF-ABWR-RC-11, above.**

443. The ORs set by Hitachi-GE to control feedwater oxygen injection are detailed in Ref. 20. They are presented in Annex 5 of my report. I am content with Hitachi-GE's suggestion of implementing feedwater oxygen as a control parameter, because this does not require direct control in the reactor water. With respect to FAC mitigation, as I have previously concluded in Section 4.3.2.1 of my report, I am not content with the lower "limit" suggested. The upper feedwater oxygen limit is set on the basis of increases in corrosion rates (pitting) for CS. No consideration is given to the impact the selected value may have on radioactivity. As above, I expect the future licensee to review these aspects of the feedwater oxygen ORs defined in the UK ABWR safety case. **This has been captured as part of Assessment Finding, AF-ABWR-RC-11, above.**

Impact of Impurity Ingress on Radioactivity

444. From a radioactivity perspective, the net effect of impurities is an increased concentration of CPs in the reactor. Considering Table 2 again, in this section I therefore focus my assessment on “*minimisation of impurity ingress*”. A number of the APs in UK ABWR are produced as a result of impurities entering the RCS. In addition, a number of important APs are produced by activation of the coolant itself. Some of the important APs in UK ABWR are given in Table 6.

Nuclide	Half-life	Production Route	Main Sources
^3H	12.3 years	$^2\text{H} (n, \gamma) ^3\text{H}$ $^{10}\text{B} (n, 2\alpha) ^3\text{H}$ $^{11}\text{B} (n, n2, \alpha) ^3\text{H}$ $^{235}\text{U} (n, f) ^3\text{H}$ $^{238}\text{U} (n, f) ^3\text{H}$	Coolant (main source) Control rods Control rods Impurity Impurity
^{13}N	10 minutes	$^{16}\text{O} (n, \alpha) ^{13}\text{N}$	Coolant
^{16}N	7.1 seconds	$^{16}\text{O} (n, p) ^{16}\text{N}$	Coolant
^{14}C	5,730 years	$^{13}\text{C} (n, \gamma) ^{14}\text{C}$ $^{14}\text{N} (n, p) ^{14}\text{C}$ $^{17}\text{O} (n, \alpha) ^{14}\text{C}$	Impurity Impurity Coolant (main source)
^{15}C	2.5 seconds	$^{18}\text{O} (p, \alpha) ^{15}\text{C}$	Coolant
^{18}F	1.8 hours	$^{18}\text{O} (p, n) ^{18}\text{F}$	Coolant
^{24}Na	14.9 hours	$^{23}\text{Na} (n, \gamma) ^{24}\text{Na}$	Impurity
^{36}Cl	301,000 years	$^{35}\text{Cl} (n, \gamma) ^{36}\text{Cl}$	Impurity
^{41}Ar	1.8 hours	$^{40}\text{Ar} (n, \gamma) ^{41}\text{Ar}$	Impurity

Table 6: Typical Activation Products expected in UK ABWR

445. Ref. 80 describes the consideration given to APs in the generic safety case for UK ABWR, including their expected behaviour. Ref. 80 is also supported by *Topic Report on Impurity Ingress* (Ref. 30), which presents information on potential impurity sources, risks to plant components, mitigation measures and OPEX of impurity ingress events in BWRs. While the latter TR is much broader in scope than just considering radioactivity. The impact of impurities is considered, and I am satisfied this is done in an appropriate manner. The responses to RI-ABWR-0001 (Ref. 66) also contained information on the generation, behaviour and sources of APs in UK ABWR. I considered this as part of my assessment reported in Ref. 79, so I do not repeat the detail here.
446. What is clear is that many of the important APs (^3H , ^{13}N , ^{16}N , ^{14}C , ^{15}C and ^{18}F) are either entirely or predominately generated by activation of the coolant. There is therefore nothing the operating chemistry can do directly to minimise them, nor impact their behaviour, which is mainly influenced by other factors. As assessed earlier, the exceptions to this are ^{13}N and ^{16}N . Of the remaining APs (including ^{14}C , ^{24}Na , ^{36}Cl and ^{41}Ar), they are minimised by a combination of minimisation at source and efficient removal from the RCS. Both of these factors are considered as part of my assessment of the chemistry control system for the RCS (Section 4.3.1.4). My conclusions presented later in the report remain valid here. In summary, I am content that it should be possible for a future licensee to operate UK ABWR in a manner which reduces radioactivity caused by impurities to levels considered “ALARP”.

447. Ref. 20 also defines a number of ORs which are related to impurities. They are presented in Annex 5 of my report. The limiting values for these parameters are determined by materials degradation (especially SCC) concerns, not radioactivity. It is not realistic to set ORs for all the potential impurities, hence direct conductivity is used as a general measure of (ionic) impurity concentrations, and the limit is lower than the corresponding limit for chloride, assuming it is present as hydrochloric acid. I consider this reasonable. However, I would have expected a similar general control over the level of radioactivity, for example, total radioactivity as a diagnostic parameter. It is also not clear to me how the existing ORs enable the future licensee to understand the concentration of other, more specific radiochemical parameters. In general, therefore, I am not content that these ORs give sufficient consideration to radiochemistry related parameters. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

Impact of Commissioning Chemistry on Radioactivity

448. I have already assessed certain aspects of the commissioning chemistry topic earlier in my report, including the overall approach and strategy for commissioning chemistry in Section 4.3.1, and its impact on materials integrity in Section 4.3.1.1. This was also looked at independently by my TSC in Ref. 37. My assessment here concentrates on aspects of the topic which are directly relevant to demonstrating that radioactivity is reduced SFAIRP.
449. From a radioactivity perspective, one of the most important aspects of commissioning is the final stages when the RCS is commissioned. This is when the reactor is operated at (near) full system temperatures and pressures, and is commonly referred to as Hot Functional Testing (HFT).
450. As already explained in Section 4.3.1, Chapter 23 of the PCSR recognises the importance of commissioning chemistry, with respect to minimising radioactivity, making the following claim:
- “[RC SC5.4] *Commissioning activities will be optimised to ensure that the longer-term radiation exposure will be minimised to ALARP*”.
451. Claim [RC SC5.4] is very clearly targeted towards the importance of aiming commissioning activities at reducing longer-term ORE. Commissioning chemistry is also considered in multiple other reports within the UK ABWR generic safety case, including Refs. 27, 80 and 85, and *Topic Report on Commissioning Chemistry* (Ref. 38). Ref. 38 contains the detailed evidence used by Hitachi-GE to support their overall arguments for the HFT period of commissioning. Applying adequate chemistry control during this period can influence many factors identified by Hitachi-GE in Table 2. Those most readily influenced are associated with the control and minimisation of CPs, by minimising the loading of CPs and influencing the tendency of the surfaces to incorporate ⁶⁰Co in later stages, when fuel is loaded to the core and nuclear heating commences.
452. As defined in Ref. 38, the latter two stages of commissioning (≥ 100 °C without fuel, and ≥ 100 °C with fuel) most closely align with HFT. Ref. 38 provides an overview of several options considered to reduce radioactivity for each of these phases, but also states that the last of these phases is the most important, overall, for future plant radioactivity performance. I consider the range of options identified for all of these phases by Hitachi-GE, to be suitable for GDA. Ref. 38 provides a description of the evidence and rationale considered in forming a judgement on their potential applicability. The net effect is that Hitachi-GE propose the following key aspects of commissioning chemistry related to the mitigation of radioactivity:

- Phase one (≤ 100 °C without fuel) – minimising impurities (including metallic CPs) is prioritised during this phase. In addition to using high purity water, this means that dry storage, if periods are long, and the ability to completely clean the CFDW system are recognised as areas where further consideration is needed. The addition of hydrazine, to scavenge oxygen, is also identified as a potential “ALARP improvement”.
 - Phase two (≥ 100 °C without fuel) – during this phase the development of the oxide film on surfaces becomes an important consideration, in addition to reducing impurities. This is targeted towards those areas of the plant which are known to impact ORE, such as the CUW system and BDL. Given the eventual at-power operating chemistry, and the effects this has on oxide films, an important consideration is when noble metals or DZO injection should be applied, or whether its eventual implementation would restructure the oxides, rendering alternative approaches nugatory. Hitachi-GE’s conclusion is that, at present, the preferred approach would be to use high purity water without noble metals. DZO injection or nickel controls are recognised as potential options, subject to further development. Previously applied techniques, such as alkali pre-filming, are ruled out due to changes to materials (from CS) and the “new” proposed RCS operating chemistry making any benefits negligible.
 - Phase three (≥ 100 °C with fuel) – this is the first time radioactivity is generated; therefore the approach is largely consistent with at-power operations. Hitachi-GE’s currently preferred option is to use high quality demineralised water up until the first start-up, with hydrogen injection started at lower power (~5-10 % of power) and DZO injection deferred until HWC is underway (on the first start-up), due to concerns regarding the formation of zinc chromite in the presence of the high chromium concentrations expected, as new surfaces passivate. Iron control measures, such as condensate filter by-pass, or iron to nickel ratio control may also be required. Oxygen injection will be started when the feedwater flow is established and noble metal injection (via OLNLC) injection will be carried out after operating not less than 60 days at power (owing to fuel integrity concerns).
453. The philosophy and approach adopted in Ref. 38 is reasonable. I am content that suitable consideration has been given to radioactivity minimisation, in comparison to other hazards, during this important period. Hitachi-GE’s rationale for eliminating many of the options considered is reasonable, although clearly, this needs to be kept under review by the future licensee as technology and good practices develop in this area.
454. The fundamental objective of Ref. 38 is to address the fact that while there is much experience of commissioning BWRs to operate under NWC, it is much less clear whether this would be appropriate for UK ABWR, given the eventual adoption of a different at-power operating chemistry, and therefore whether changes could be made which would offer further benefits. This means the key consideration is deciding the most appropriate point to begin implementing HWC, noble metals and DZO injection. Controlling other chemistry-related parameters i.e. impurities and iron, would be the same regardless of whether NWC or HWC were adopted. Considering the strong influence DZO has on oxide films, and therefore the deposition behaviour of ^{60}Co , the intuitive answer would be to begin DZO injection as soon as practically possible. The ideal situation would be to have fully formed oxides before radioactivity is produced, perhaps which already contain DZO to inhibit ^{60}Co uptake. The problem in making this decision is that because there is no plant OPEX on which to directly draw, there is some considerable uncertainty over whether this is practicable.
455. As explained above, the conclusion reached by Hitachi-GE is therefore that the earliest this can be achieved is likely to be some point towards the first plant start-up, or early within it. In some instances this is down to the specifics of how a BWR can be commissioned (e.g. the capabilities to generate significant coolant temperatures

without nuclear heating). In other instances, Hitachi-GE try to draw conclusion based on other OPEX, or by using first principles scientific arguments. I consider this approach is satisfactory at this stage. However, it does make it very difficult to be able to conclude if the solution proposed for GDA is the option which reduces radioactivity SFAIRP. For example, some options are conclusively dismissed by Hitachi-GE on the basis of this limited OPEX (e.g. early injection of DZO due to risks of zinc chromite formation), while others are not adopted, even though the balance of evidence is comparable (e.g. application of noble metals earlier in the process, despite it having obvious benefits in oxide layer formation (and SCC mitigation)). However, there is little benefit to be gained by progressing these matters further at this stage of the project.

456. Overall, I am satisfied that suitable and sufficient consideration has been given to this topic for GDA. However, I am not content that the specific claim made by Hitachi-GE, [RC SC5.4], is achieved at this stage. Owing to the uncertainties which remain and potential significant benefits that could be gained by implementing some of the options proposed during GDA, it is not clear whether the conclusion reached at this stage will eventually turn out to be the "ALARP option". Important, well informed decisions which build on the work performed during GDA will be required of a future licensee, to address some of the uncertainties. Nevertheless, I am broadly satisfied the UK ABWR generic does not foreclose the option of other chemistry choices, aimed at minimising radioactivity, being made (for example, the ability to implement DZO injection earlier). **This has been captured as part of Assessment Finding, AF-ABWR-RC-03.**
457. Ref. 38 also highlights that the operating chemistry during commissioning is likely to offer little benefit to some systems, notably the RHR system. This therefore places further emphasis on other means to reduce the deposition of radioactivity, such as surface treatments described earlier in my assessment, or operating practices considered later, below.

Impact of Start-up and Shutdown Chemistry on Radioactivity

458. A number of significant chemistry changes take place during start-up and shutdown operations as the primary coolant is taken from hot reducing conditions to cold oxidising conditions, and back again. These changes cause a number of effects. During shutdown the most important of these is an increase in reactor coolant concentrations of both soluble and particulate radionuclides (from fuel deposits and soluble CPs, and from fuel depressurisation). This is the so-called shutdown radioactivity "spike". This has a pronounced effect not only on the speed and safety of the outage activities but also on future operation of the reactor during the subsequent fuel cycles. A similar (but generally much smaller) event occurs during start-up, where the main concern is related to impurity ingress. Proper chemistry controls during these periods can minimise these effects, thereby directly influencing a number of factors in Table 2.
459. My assessment here is focussed on the principal chemistry changes which occur during start-up and shutdown operations, and the controls placed over them, as opposed to a system-by-system assessment. Start-up and shutdown chemistry was also considered by my TSC in Ref. 37. I have drawn on their work in various places and I have also considered relevant industry approaches, such as Refs. 44, 45 and 53, in support of my assessment of this sub-topic.
460. Start-up and shutdown chemistry is considered in, *Topic Report on the Water Chemistry Regime ALARP Assessments on the Iron Control at Power, Commissioning, Start-Up and Shutdown operational modes* (Ref. 27). Further detailed evidence is also provided in, *Topic Report on Start-up and Shutdown Chemistry* (Ref. 36). Both of these reports consider the impact of these operations on radioactivity. Start-up and shutdown is therefore only considered briefly within Refs. 80 and 85, which both refer out to Ref. 36 to provide the detailed arguments and evidence.

461. The basic approach adopted by Hitachi-GE for shutdown chemistry control (in Ref. 36) is to maintain the at-power operating chemistry conditions for as long as possible, whilst ensuring chemistry measures which may contribute to ORE during the outage are managed appropriately. Impurities will be minimised throughout the transient, but this means that HWC, DZO and feedwater oxygen injection remain in operation until the low feedwater flow trip. Ref. 36 provides the evidence to support this approach, which is based on US and Japanese OPEX. I am content with this evidence.
462. As well as discussing the effects of the various chemistry changes during shutdowns on radioactivity, Ref. 36 also considers the shutdown process more generally. This is important because most of the shutdown release occurs due to mechanisms which are not chemistry related, but rather due to physical processes. The main of these are changes to the boiling regime in the core, which removes insoluble deposits from the fuel cladding as particulates, the temperature decrease, which can increase the solubility of the various species, and the pressure reduction, which can allow FPs to be released from within the fuel, particularly if any fuel cladding is damaged. In practice what is observed is a peak in the reactor coolant concentration of both soluble and insoluble radioactivity, perhaps by several orders of magnitude. This peak tends to occur at a similar time to when the RHR system is placed in service to provide shutdown cooling. An example of this behaviour is given in Figure 25 for ^{60}Co .

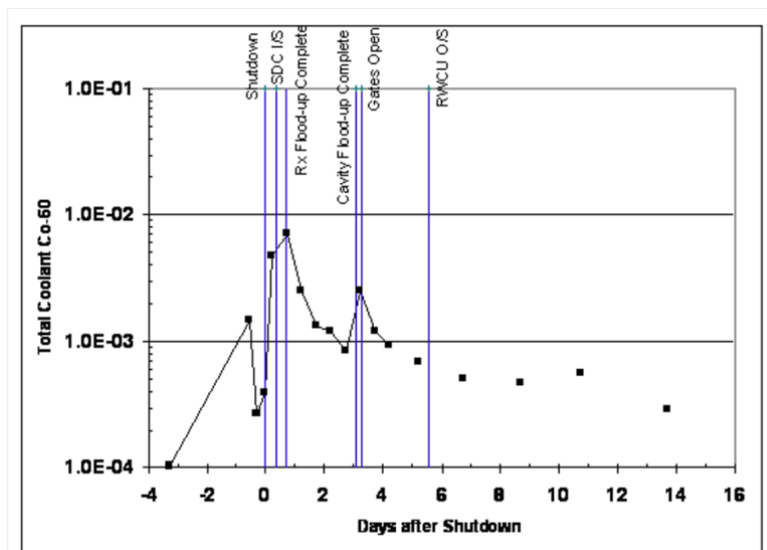


Figure 25: Typical BWR shutdown behaviour for ^{60}Co (Ref. 53)

463. I note that Ref. 36 concentrates on CPs, providing no information on the FP releases from fuel. The magnitude of these FP releases is entirely due to physical processes, which the chemistry cannot influence. However, as with at-power operations, the chemistry conditions can influence the behaviour of these nuclides in particular the volatile species. Such effects are not considered by Hitachi-GE. Removal of the noble gases is straightforward, but can be time-consuming; considering they are entirely volatile they can be removed by venting through the OG system. Purging systems for iodine removal can be rather difficult, as the iodine is continuously delivered by drying surfaces or is volatilising from the coolant. This is only likely to be of a concern when significant quantities of iodine are present in the coolant, due to either fuel failures or tramp uranium contamination. While the minimisation of both of these causes has been considered by Hitachi-GE, I consider that the generic safety case should address what impact this would have on the shutdown, if they were to occur. This is especially important as some of the limits associated with the Generic Technical Specification (Ref. 90) allow UK ABWR to run with a proportion of failed fuel (and hence associated

FPs in the coolant) present in the core. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-12: During GDA, ONR's chemistry assessment established there is a specific gap in the UK ABWR generic safety case. There is no consideration of how shutting the reactor down with elevated levels of radioactivity contained within the reactor coolant, may impact on maintaining adequate levels of safety. Chemistry controls during shutdown assume "normal" levels of radioactivity. Other plants have specific controls and measures in place when radioactivity is elevated, to minimise effects which include the potential transfer of radioactivity to the turbine, and therefore the impact on worker doses. The licensee shall review and update the UK ABWR safety case to justify that the chemistry controls in place during a shutdown are adequate to account for such circumstances, and whether additional measures need to be adopted, to ensure relevant risks are reduced SFAIRP.

464. The impact the operating chemistry has on this spike release was an important part of my assessment of Hitachi-GE's responses to RI-ABWR-0001 (Ref. 79). I was content with the evidence provided there and my assessment concluded that the generic safety case produced by Hitachi-GE adequately accounts for the potential impacts of the operating chemistry on the magnitude of the release.
465. Given the potential variability in the timings and quantities of radioactivity released to the reactor water during shutdown, Hitachi-GE also consider if additional controls or constraints are necessary to ensure that workers are not exposed to unnecessary doses. On this basis, controls are suggested for both the ^{131}I and ^{60}Co concentrations before permitting RPV head removal and the refuelling cavity to be filled with water. I consider the adequacy of these ORs later, in Section 4.8.2, but judge that these are valuable additions to the safety case.
466. During the outage itself, the coolant is maintained as clean as possible, both from a chemistry and radiochemistry perspective. There are no purposeful chemical additions during this phase. No radioactivity is generated during this period, but measures to maintain coolant purity will be beneficial in terms of redistribution of existing deposits. I am satisfied with this approach. Further assessment of the chemistry aspects of refuelling activities, and how these affect the SFP is provided in Section 4.4.
467. The fundamental objective for chemistry control during the start-up phases is to establish the at-power operating chemistry regime as soon as possible. Ref. 85 summarises the approach adopted by Hitachi-GE in the context of reducing radioactivity, along with the supporting evidence.
468. The technical basis for this approach is very similar to the one outlined for at-power conditions, and the reasons why this approach is adopted here is the same as why the at-power operating chemistry regime is maintained into the shutdown phase for as long as possible. I am therefore content this is a suitable strategy.
469. In order to ensure that these important periods are appropriately controlled, several chemistry-based ORs are defined in Ref. 20. They are presented in Annex 5 of my report. Notwithstanding the ambiguity in the definition of these ORs considered in **AF-ABWR-RC-05**, there are a number of chemistry-based ORs which are related to radioactivity during start-up, shutdown and outage periods. The general approach is very similar to the at-power phase; the limiting values are the same or similar, but the expected values are less stringent. In addition to chemistry-related ORs, Ref. 20 also identifies a number of radiochemistry and other ORs which are of relevance to start-up and shutdown periods. Confusingly, these are not formally captured in the chemistry parameters presented the *Water Quality Specification* (Ref. 20), and are presented separately. The basis for these is discussed within Ref. 36. They include limits on ^{131}I and ^{60}Co . These are important ORs from an ORE perspective, as they are primarily

aimed at limiting the potential impact of radioactivity on workers. It is beyond the scope of my assessment to consider the specific values suggested. Why these are not included within the specifications for the coolant quality is unclear, nevertheless, I am content that such controls have actually been specified. What is less clear is whether these, in themselves, would be sufficient. During GDA, I have seen no justification for whether additional limits on radioactivity tied to other important stages during the outage provide benefits. For example, before stopping the RIPs or isolating the CUW system. Similar measures are suggested for consideration in Refs. 44 and 54.

470. Finally, Ref. 20 also appears to recognise that some aspects of the supporting evidence, mainly from the TRs, are not fully reflected within the reactor water or feedwater specifications at present. A number of “*additional reactor chemistry related ORs from other safety case documentation*” are presented (Annex 2 of Ref. 20). This includes (of relevance to radioactivity);

- Oxygen within the reactor water as ≤ 200 ppb prior to start-up;
- Insoluble material (iron) within the feedwater and condensate water as ≤ 1 ppb prior to start-up; and
- Insoluble material (iron) within the RHR system as ≤ 500 ppb following RHR system flushing.

471. Overall, I am not content that collectively, the UK ABWR generic safety case provides an adequate set of limits and conditions in the form of ORs, which reflect the key safety aim defined in the generic safety case to minimise radioactivity during start-up and shutdown periods, SFAIRP. As described in my preceding assessment, there are several important aspects of the generic safety case made by Hitachi-GE that do not appear to be reflected in the output, in terms of the identified limits and conditions (or ORs). This includes:

- Several control parameters do not have associated limits.
- Several parameters will not be applied for the full period (notably hydrogen, DZO and oxygen), but the ORs do not appear to reflect this.
- Several important parameters may require more explicit controls, such as iron (and other metallic impurities), or oxygen, depending upon the temperature or other factors.
- The relationship between radiochemistry ORs and key steps in the start-up and shutdown processes needs to be further justified.
- Some conditions identified do not appear to be reflected in ORs, for example the 60 day fuel conditioning period before OLNLC is applied.
- No consideration appears to have been given to aligning the requirements identified in all aspects of the safety case, and;
- No ORs are given for the CFDW system during outages.

472. Considering the overall importance of these parameters to be able to control and minimise radioactivity within the plant, I expect the future licensee to demonstrate the above gaps identified in the UK ABWR generic safety case, have been adequately addressed. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

Reduction of Radioactivity by the Operating Chemistry Summary

473. Hitachi-GE has clearly put significant effort into understanding and justifying the impact of the operating chemistry choice on radioactivity generation, transport and behaviour during all modes of operation. This is reflected within the large quantity and generally good quality of safety case documents that relate to this topic. It is clear what factors are important and how the operating chemistry influences these. As with all aspects of chemistry, there are sometimes compromises that need to be made in relation to other hazards and risks, but I am content overall, Hitachi-GE has demonstrated the UK

ABWR operating chemistry has been suitably optimised to include the impact it has on the generation and transport of radioactivity.

474. An important aspect of ensuring that radioactivity is reduced SFAIRP within the overall operating envelope for the plant, is to define a suitable set of ORs. It is this aspect of the generic safety case that will require significant attention moving forward. I am not satisfied that the current ORs are sufficient in this regard, both in their breadth of coverage, but also in terms of the limits specified, in some instances. This has been captured as part of Assessment Finding, **AF-ABWR-RC-02**, but I have also raised Assessment Findings, **AF-ABWR-RC-11 and -12**, during my overall assessment of the impact of the operating chemistry on radioactivity. These will be very important areas of work for a future licensee to address.
475. I particularly want to stress the impact of commissioning chemistry. While I am content that sufficient information has been provided during GDA, I also expect this to be an area of significant, important work conducted by a future licensee. Substantial and tangible nuclear safety benefits can be achieved by addressing this important period of operations, but this will be more challenging given the UK ABWR will be a first of a kind, in this regard. This has been captured as part of Assessment Finding, **AF-ABWR-RC-03**.

4.3.3.5 Reducing Radioactivity by Operating Practices

476. The operator can also influence the production, distribution and behaviour of radioactivity via decisions over how the plant is operated. Many of these are related to chemistry-related systems or processes. During GDA, this is the reason why RO-ABWR-0006 Action 6 (Ref. 25) was raised.
477. Hitachi-GE's principal document covering this subject is, *Topic Report on Reduction of Source Terms by Operating Practices* (Ref. 84). This is closely linked to other submissions and TRs on this theme, including Refs. 30, 36 and 80. Collectively, these reports provide the justification and supporting evidence for the adoption of a number of operating practices. Ref. 84 also identifies those practices which may not be directly relevant for UK ABWR, given the material and operating chemistry choices - even if they have been previously applied on J-ABWRs. A summary of the conclusion from Ref. 84, which is based on worldwide experience, is given in Table 7 below. Table 7 also identifies the main benefit of each practice; either minimising the quantity of radioactivity in reactor water, or deposited on piping and surfaces.

Operating mode	Operating Practice	Main Benefit
Power Operation	Impurity minimisation	Reduction in reactor water radioactivity
	Power suppression	Reduction in reactor water radioactivity
	CUW system and CPS operations	Reduction in reactor water radioactivity
Shutdown	RHR system clean-up	Reduction in reactor water radioactivity
	Soft shutdown	Reduction in reactor water radioactivity
	Low temperature RHR system operation	Reduction in deposited radioactivity
Outage	Storage conditions	Reduction in reactor water radioactivity
	Reactor draining route	Reduction in reactor water radioactivity
	Chemical decontamination	Reduction in deposited radioactivity
Start-up	CFDW system clean-up	Reduction in reactor water radioactivity

Table 7: Identified operating practices to be applied for UK ABWR to reduce radioactivity (Ref. 84)

478. All of these identified practices have been previously applied in operating J-ABWRs. I consider the suitability of each of these to UK ABWR, below.

Operating Practices during Power Operation

479. During power operation impurity minimisation is a key practice. This has been described previously in my assessment, when considering the impact of the operating chemistry. Ref. 84 reflects the importance of this, referring to the impurity ingress topic report (Ref. 30) as providing the detailed evidence.
480. Correctly operating the CUW and CPS clean-up systems is also important for minimising radioactivity, not only during power operations, but also during start-up, shutdown and outage periods. Ref. 84 provides the links to the detailed engineering for these systems, but more relevantly describes how they can be managed and optimised to maximise their efficiency for radioactivity removal. This includes factors such as management of filters and ion exchange resins, timings for maintenance activities and flow rates. It is clear that the design of these systems allow sufficient flexibility for the operator to determine the most efficient Operating Procedures. I am content that a competent licensee will be able to develop this post-GDA, with Ref. 84 providing a useful input. This directly contributes towards both "*minimisation of the input of metallic impurities into the reactor*" and "*minimisation of impurity ingress*" from Table 2.
481. The final practice identified by Hitachi-GE used during power operation periods is "*power suppression*" of damaged fuel. Damage to fuel cladding will result in the release of some fraction of the radioactivity normally contained with the fuel rods into the reactor water. BWRs use a process known as Power Suppression Testing, to locate the damaged fuel. This involves reducing power, inserting control rods and monitoring the noble gas releases until the damaged fuel is identified. Once located, and provided the radioactivity remains below the limits with the Generic Technical Specifications (Ref. 90), the fuel rod can be isolated by keeping the control rods around it inserted. This stops further radioactivity release, provided the damage does not degrade further to secondary failures. Identifying this practice directly responds to "*minimisation of FP releases from failed fuel by the use of power suppression*", identified in Table 2. I consider the benefits and detriments for power suppression more comprehensively in Section 4.3.1.4. I consider if it can be claimed as an operating practice for minimising radioactivity, here.
482. Supporting evidence for the practice is contained in the report, *Management of Damaged Fuel* (Ref. 91). This shows typical data for radioactivity releases from damaged fuel following power suppression (for noble gas radioisotopes and ⁹²Sr). These data are shown in Figure 26. As Figure 26 shows, the noble gas activity rises sharply around 50 days into the fuel cycle. A period of power suppression follows, during which the fuel is identified and adjacent control rods inserted. The radioactivity levels then return to lower levels, for the next 21 months of operation. From a theoretical and practical point of view it is clear that power suppression is therefore a suitable practice for minimising radioactivity.

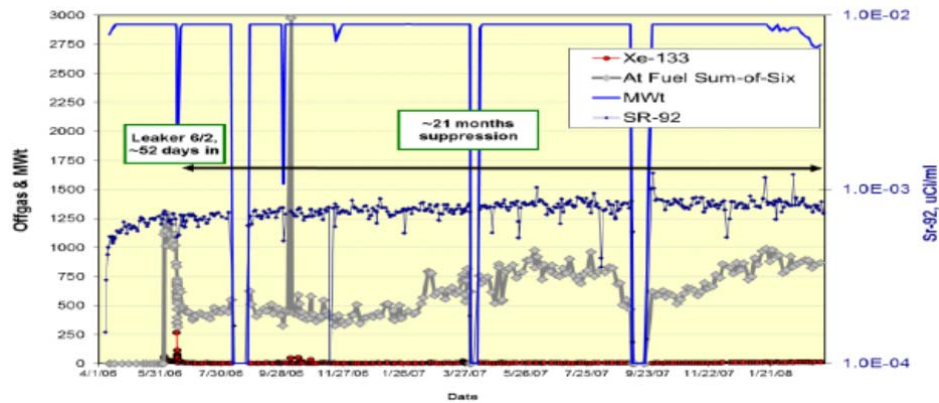
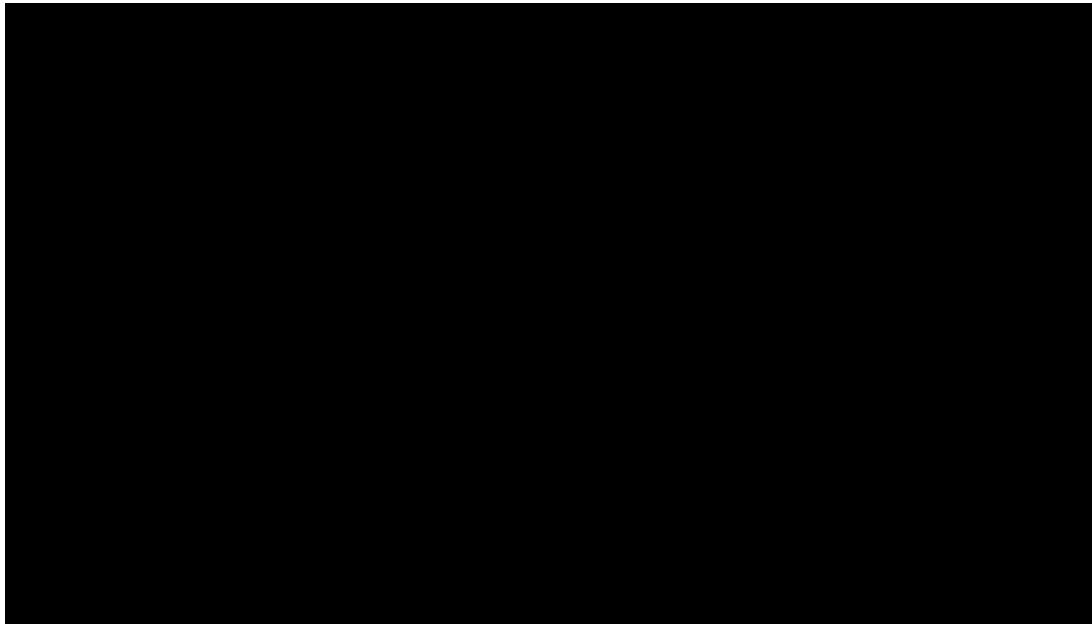


Figure 26: Radioactivity releases from damaged fuel during power suppression (Ref. 91)

Operating Practices during Shutdown

483. During a shutdown Hitachi-GE identify three beneficial operating practices. Firstly the RHR system is flushed with water, typically from the suppression pool or water make-up system. This removes corrosion products which have accumulated within the RHR system while it has been in storage, minimising the burden that is transferred to the reactor water when shutdown cooling is engaged. Hitachi-GE estimates this can reduce this by a factor of five or more. Ref. 20 identifies an OR which states that Insoluble material (iron) within the RHR system as ≤ 500 ppb following RHR system flushing. As noted earlier in my assessment, I am not convinced by the value stated for this parameter, but am encouraged that such a control is specified for UK ABWR.
484. The second process is the so called “soft shutdown” which is a process whereby the cool down rate during the shutdown is minimised, from the $50\text{ }^{\circ}\text{C hr}^{-1}$ allowed under the Generic Technical Specifications (Ref. 90) to a much lower rate of less than $30\text{ }^{\circ}\text{C hr}^{-1}$. This latter value is noted as an OR within Ref. 20. This is based on practices originated in German BWRs and has been applied successfully to Japanese BWRs. This can reduce the release of deposits from the fuel, minimising the reactor water radioactivity.
485. Finally, OPEX has demonstrated that reducing the temperature at which the RHR system is brought in to service for shutdown cooling has the potential to reduce radioactivity deposition within the RHR system by up to 80%. This is known as Low Temperature – Shutdown Cooling (LT-SHC). This latter practice is the one which has demonstrated the greatest benefit, as shown in Figure 27. The start temperature of the RHR system is stated as $\leq 120\text{ }^{\circ}\text{C}$ within the ORs defined in Ref. 20.



486. The evidence for the adoption of these practices within UK ABWR is compelling. I am content that both the generic safety case specifies them, and that the UK ABWR generic design is capable of delivering them.

Operating Practices during Outages

487. Following a shutdown, there are also practices which can be applied during the outage. While no radioactivity will be generated during this period, these measures reduce the burden on the plant at start-up, mainly in relation to minimising impurities. While it will be for the future operator to put such measure into practice, I see no reason why they cannot be done and the evidence provided in Ref. 84 demonstrates their potential benefits. These include:
- Adopting appropriate storage conditions for the CFDW system, including dry storage for extended periods. This will greatly minimise corrosion and subsequent corrosion product generation.
 - An important consideration is the storage conditions for the CD ion exchange resins, which degrade faster, producing organic contaminants if stored in aerated conditions. Hitachi-GE proposes to use deaerated conditions for UK ABWR and also to drain the storage water to waste before returning the system to service. These have proved beneficial in Japanese BWRs to minimise resin degradation products for contaminating the reactor water.
 - Draining excess water from the reactor during start-up or shutdown periods via the condenser hotwells has been shown to contaminate the CFDW system components and Condensate Storage Tank (CST). Hitachi-GE propose to adopt the common Japanese practice of routing this water directly to the radioactive waste treatment systems.
488. Chemical decontamination is a process that has been applied to existing BWRs (including ABWR) to reduce deposited radioactivity, by up to 95%. It is typically conducted during an outage and uses a series of chemical stages to remove the contaminated oxide films from system surfaces. A number of techniques have been developed and used commercially, including by Hitachi-GE (details in Ref. 44). Importantly, while Ref. 84 notes this as a potential operating practice, it is acknowledged that it is not expected to be routinely applied in UK ABWR given the changes to the materials and operating chemistry to reduce radioactivity SFAIRP. It is argued that such processes are compatible with and could be applied to UK ABWR, if

necessary, with secondary wastes compatible with the existing plant radwaste systems. The operating chemistry choice does not impact on the applicability of such techniques, but may influence how the plant is re-commissioned following any decontamination. I am satisfied with these arguments, and consider it would be disproportionate to provide further details or evidence at this stage of the project, given these are best developed if and when chemical decontamination is to be applied.

Operating Practices during Start-up

489. Despite the careful storage of the CFDW system some corrosion will inevitably occur, particularly in the heater drains, feedwater and condensate piping, which during start-up could enter the reactor water. This could be a significant fraction of the overall iron input to the reactor over a full fuel cycle (10s of %). To prevent this UK ABWR will be operated such that the feedwater is recirculated via the CFs to remove insoluble species prior to start-up. The CDs are isolated to prevent carbonate absorption which would reduce their future efficiency. Overall, I consider this to be an important step in the start-up process for UK ABWR and it is valuable that it is recognised as such by Hitachi-GE.

Other Operating Practices

490. Ref. 84 also identifies other operating practices, which are not considered suitable for UK ABWR. They include: Hi-F coat, alkali pre-filming and nickel injection, and are considered for application during outage periods. They were also considered by Hitachi-GE under commissioning, in Ref. 38. Low Temperature NobleChem™ (LTNC), Enhanced Low Temperature NobleChem™ (ELTNC) and titanium dioxide addition were also identified as other practices which could bring benefits, but further development is required.
491. For GDA, I am satisfied with Hitachi-GE's conclusion that these operating practices are currently not deemed applicable to UK ABWR. These, plus others which develop with the passage of time, would need to be considered in more detail by the future licensee in the interim period, up to the point when UK ABWR may become operational. I therefore consider this to be a matter for normal business.
492. Other parts of the generic safety case also identify important operating practices for controlling radioactivity, which are not explicitly detailed in Ref. 84. They include:
- Moisture carryover is discussed in the topic report on start-up and shutdown chemistry (Ref. 36). Some BWRs can change the core flow rates towards the end of cycle to ensure maximum fuel utilisation. This increases the moisture carryover in the steam, which directly affects the transport of radioactivity to the turbine systems. Hitachi-GE have assumed that UK ABWR operates on an 18 month fuel cycle (17 operation and one outage), and therefore such flow effects are not expected to be significant. In UK ABWR this may raise the moisture carry-over by 10%, but this is considered bounded by the design assumptions. This is clearly an operator led decision and would need to be reviewed again should this assumption change.
 - Production of ^3H is an intrinsic part of operating a BWR as it is produced mainly from ^2H within the water coolant. As such it also stays with the water, and due to there being no abatement system to remove it, all process water and steam volumes have the same concentration. There are three loss mechanisms for ^3H ; evaporation, leakage and discharge of water. From these mechanisms, discharge of water is the only one that is able to be managed. By actively discharging water via the liquid radwaste system and replenishing the balance from the CST, the concentration of ^3H will be reduced. This procedure is used to manage the concentration in the various plant systems, in accordance with the factor "*reduction of ^3H in the plant via discharge and make-up processes*" in

Table 2. This process was considered in my assessment for RI-ABWR-0001 (Ref. 79), and supporting calculations are given in Ref. 73. These show that the concentration reaches equilibrium after four or five cycles of operation, but can be managed by the future licensee if necessary.

493. I am satisfied that these important operating practices are considered appropriately within the safety case at this stage, but they need future licensee input and site specific decisions to be properly implemented and to be able to demonstrate they have reduced radioactivity SFAIRP. I consider this to be a matter of normal business.

Reducing Radioactivity by Operating Practices Summary

494. I consider that the UK ABWR generic safety case produced by Hitachi-GE to demonstrate that all reasonably practicable operating practices can be adopted by the future licensee to reduced radioactivity is adequate. A number of valuable processes have been identified and included within the identified Operating Rules and it clear that the existing plant design is sufficient to allow these to be done. A number of other potential options have been highlighted, which could be considered for future adoption, depending on additional OPEX or research as it becomes available.

4.3.3.6 Other Controls on Radioactivity

Fuel Quality Control

495. As described earlier the FPs produced within the fuel will tend to remain contained within the fuel cladding unless damage occurs during operations. The impact the operating chemistry has on the integrity of the fuel cladding is a topic in its own right and my assessment is presented in Section 4.3.1.5 that follows. An indication of some important FPs likely to be present within the coolant is given in Table 8, along with their behaviour. Noble gases are transferred 100% to the steam phase, soluble species remain within the reactor water while volatile species will partition between both the water and steam phase. This behaviour was assessed as part of my assessment for RI-ABWR-0001, presented in Ref. 79. I remain content that the consideration given to these by Hitachi-GE is reasonable. In summary, the operating chemistry only has very limited impact on this behaviour, except for some specific nuclides such as Iodine (as discussed earlier in my assessment).

Nuclide	Half-life	Behaviour
^{85m} Kr	4.4 hours	Noble gas
⁸⁷ Kr	76 minutes	Noble gas
⁸⁸ Kr	2.8 hours	Noble gas
¹³³ Xe	5.3 days	Noble gas
^{135m} Xe	9.2 hours	Noble gas
¹³⁸ Xe	14 minutes	Noble gas
⁹¹ Sr	9.7 hours	Soluble
⁹⁹ Mo	67 hours	Soluble
¹³¹ I	8.0 days	Volatile
¹³⁸ Cs	32 minutes	Soluble
²³⁹ Np	2.4 days	Soluble

Table 8: Typical Fission Products and Actinides expected in UK ABWR

496. There are a number of other factors identified by Hitachi-GE in Table 2 which are related to the fuel and fission product concentrations. Even for a plant operating without fuel damage, FPs will be generated directly from “tramp” uranium. The different sources of tramp uranium are discussed in detail in the Primary Source Term Supporting Report (Ref. 73) and are:
- Residual uranium dioxide dust on the outside of fuel pins, present from the fuel fabrication process.
 - Uranium impurity within the Zircaloy cladding material.
 - Uranium that has been previously washed out of damaged fuel, has plated-out on internal reactor surfaces and becomes redeposited on fresh fuel in subsequent reactor cycles.
497. The minimisation of the sources for the first two of these is achieved by quality controls during fuel manufacturing for UK ABWR. These therefore impact on the factors of “*minimisation of tramp uranium contamination on fuel cladding surfaces*” and “*minimisation of impurities in the zirconium fuel cladding*” in Table 2. These are discussed in Ref. 80, with calculations on the impact of these sources given in Ref. 73. Hitachi-GE claims that modern fuel manufacturing practices are sufficient to limit these sources to negligible levels. I am content with these arguments and consider it to be normal business for a future licensee to ensure such measures are implemented.
498. For the third source above, Hitachi-GE claim that power suppression also means that defected fuel is less likely to suffer further deterioration, secondary failure and release of bulk uranium dioxide material into the coolant. I consider this further in Section 4.3.1.4 of my assessment.

Other Radioactivity-related Operating Rules

499. Hitachi-GE has also identified a number of other ORs which are relevant to radioactivity. The Generic Technical Specifications for UK ABWR (Ref. 90) identifies two specific Limits and Conditions of Operation (LCO):
- LCO 3.4.5 limits the specific activity of the reactor coolant to ^{131}I specific activity of $\leq 7.7\text{E}+01 \text{ Bq g}^{-1}$; and gross specific activity of $\leq 1.0\text{E}+04 \text{ Bq g}^{-1}$ (as the weighted average of beta and gamma energy levels).
 - LCO 3.7.5 limits the gross gamma activity rate of the noble gases measured at outlet of the OG Cooler Condenser to $\leq 1.0\text{E}+8 \text{ Bq s}^{-1}$ after decay of 30 minutes.
500. Both of these values are set to ensure that the radiological consequences of potential faults are acceptable and are consistent with the assumptions made in the design basis faults analysis. I am familiar with these values from other parts of the safety case. I have not assessed the Generic Technical Specification actions associated with the LCOs, but I am satisfied that such limits have been included. I queried the basis for these values, in particular the reactor coolant radioactivity, which were clarified in Ref. 92. This clarified the relationship to the source term values, which although peculiar were demonstrably consistent. I am content for the future licensee to consider this as part of normal business.

4.3.3.7 Radioactivity Summary

501. During GDA I have assessed two main aspects of radioactivity in UK ABWR. These are, the definition and justification for the amount of radioactivity expected, and the justification for how this has been reduced SFAIRP. While Hitachi-GE found these topics challenging at times, resulting in a number of significant ROs and one RI, eventually the generic safety case and design has been judged to be acceptable for GDA.

502. The definition and justification of the UK ABWR radiological source terms is detailed in a comprehensive suite of documentation produced specifically for UK ABWR. This considers the sources, behaviour and changes in radioactivity throughout all plant systems and operating modes. It is based primarily on relevant plant OPEX, giving confidence in the defined values. A range of values are defined for use in the generic safety case, which are fit-for-purpose.
503. The control of coolant chemistry, and the selection of appropriate materials and operating practices are all important to radiation field control, and hence dose rates, and ultimately to radioactive waste management and environmental discharges. Hitachi-GE has provided a suitable case for how these factors have been appropriately considered. This builds upon the J-ABWR reference design, and incorporates a number of significant improvements in all of the areas above, which should further reduce the radioactivity generated and deposited in UK ABWR.
504. I have identified a number of areas where the future licensee will need to give further consideration to, and justify specific aspects of the UK ABWR design and safety case related to radioactivity. These have been captured in four Assessment Findings, **AF-ABWR-RC-09 to -12** and one Minor Shortfall, **MS-ABWR-RC-05**. I am content that a future licensee will be able to resolve these matters satisfactorily. I judge that these matters do not undermine the UK ABWR generic design.
505. Overall, I am satisfied the intent of the relevant SAPs relating to the adequacy of the safety case (FP.4, SC.4, SC.5 and SC.6), consideration given to chemistry effects (ECH.1 and ECH.2) and those relating to minimisation of operator doses (EHT.5, RP.1) and radioactive waste (RW.2), have been broadly satisfied.

4.3.4 Chemistry and Fuel Integrity and Performance

506. By definition, because boiling occurs at fuel surfaces in BWRs, the majority (typically 90%) of the chemical species contained in the reactor water deposit on the fuel. BWRs mitigate this by maintaining the reactor water as pure as possible, typically close to the theoretical conductivity of pure water. Historically, poor chemistry control in the RCS led to several reported fuel damage and failure events in BWRs. For chemistry, this can occur by several mechanisms:
- corrosion;
 - embrittlement; and/or
 - heat transfer impairment caused by the build-up of excessive deposits.
507. These mechanisms can lead to fuel cladding failures and the subsequent release of radioactivity (notably FPs) into the RCS, which could ultimately lead to increases in radiation doses to workers and/or the public.
508. This section of the report presents my assessment of the impact of the UK ABWR operating chemistry on fuel reliability and performance. This is an important part of my overall assessment of the operating chemistry choice because, as described earlier, if not properly controlled, one of the main detriments of implementing HWC, DZO and OLNLC could be on fuel.
509. Hitachi-GE's overall case to justify the impact of the UK ABWR operating chemistry on fuel is presented in several reports, at various levels within their hierarchy of generic safety case documentation. Chapter 11 of the PCSR (Ref. 14) presents information on the fuel, fuel assembly design and system functional claims, but there are also links to other PCSR chapters, including Chapter 23 (Ref. 11). Ref. 14 recognises the importance of adequate chemistry control to maintain fuel integrity and performance. The following "top claim" is made:

- “[RC SC3] *The UK ABWR reactor chemistry regime will contribute to the maintenance of the fuel integrity by control within limits and conditions.*”

510. Under this “top claim” several additional claims are presented:

- “[RC SC3.1] *The specified levels and timing of noble metal injection will have no detrimental effect on fuel integrity.*”
- “[RC SC3.2] *Suitable zinc injection within the upper LCO will have no adverse effect on fuel integrity.*”
- “[RC SC3.3] *Iron concentration in feedwater within the LCO will have no detrimental effect on the fuel.*”
- “[RC SC3.4] *The application of HWC and management of the H₂ injection concentration in the feedwater to LCO will ensure there is no detrimental effect on the fuel as a result of hydriding.*”
- “[RC SC3.5] *Adequate control of metal impurities especially copper in feedwater will have no detrimental effect on the fuel.*”
- “[RC SC3.6] *Foreign materials which might cause fretting will be minimised to ALARP before the first fuel loading.*”

511. As an overall set of claims, it is clear Hitachi-GE recognise the importance of adequately controlling the UK ABWR operating chemistry to prevent and/or mitigate possible detrimental effects on fuel integrity and performance. Although the precise wording of the claims could be improved, the intent is clear. I am broadly satisfied this is an adequate set of claims to present for GDA. The underlying theme of the claims is the importance of setting appropriate limit values for various chemistry parameters and ensuring they controlled within those limits.

512. Beneath the PCSR there are a number of more detailed documents containing information pertinent to the topic of chemistry and fuel, including:

- *Fuel Cladding Integrity and Control Limits of Water Chemistry* (Ref. 93);
- *Management of Damaged Fuel Report* (Ref. 91);
- *Topic Report on Noble Metal Chemical Addition (NMCA)* (Ref. 32);
- *Topic Report on Fe Concentration Control in Feedwater* (Ref. 29);
- *Topic Report on Zinc Injection* (Ref. 31);
- *Topic Report on Impurity Ingress* (Ref. 30), and;
- *The Water Quality Specification* (Ref. 20).

513. Considering Hitachi-GE’s generic safety case claims and my own knowledge of the chemistry and fuel topic, I have presented my assessment in this section of my report around three main themes:

- Controlling chemical additions and excluding impurities to minimise deleterious effects on fuel cladding and corrosion.
- Minimising the build-up of fuel deposits, SFAIRP, to ensure safety margins are not eroded.
- Chemistry and radiochemistry related to the detection and management of failed fuel.

514. The main SAPs of relevance include ECH.1, ECH.2, ERC.1 and ERC.3. To support my assessment of this topic I also commissioned a TSC to undertake an independent review of BWR fuel failure mechanisms and reported events, most relevant to chemistry. I also asked my TSC to undertake a review of a selected number of Hitachi-GE’s submissions, listed above. I have drawn on the outcome of my TSC’s work (Ref. 94) in various places throughout my assessment below.

515. The principal document in the generic safety case outlining Hitachi-GE’s understanding of how coolant chemistry is related to fuel clad integrity is, *Fuel Cladding Integrity and*

Control Limits of Water Chemistry (Ref. 93). This report includes a worldwide review of BWR chemistry-related fuel failures, including crud-induced corrosion experienced at the US BWRs, River Bend and Browns Ferry, during the late 1990s and early 2000s, respectively. The report divides the review of fuel failures into corrosion-initiated failures and crud-initiated failures. The impact of specific metallic and non-metallic impurities is also considered, as well as the potential impact of purposefully added chemicals (i.e. hydrogen, platinum, DZO). Brief descriptions of relevant OPEX and justifications for proposed limits for each of the impurities covered are also provided.

516. Fuel degradation mechanisms and the management of damaged fuel are also described separately in, *Management of Damaged Fuel* (Ref. 91). While not an exhaustive list, the mechanisms covered include: debris fretting, Pellet-clad Interaction (PCI), Crud Induced Localised Corrosion (CILC), hydriding, dryout and corrosion (oxidation). These are the main chemistry-related mechanisms I would expect such a report to cover. They are also consistent with the degradation mechanisms identified independently by my TSC. The key conclusion in Ref. 91 is that mechanical wear from debris fretting is the biggest contributor to BWR fuel failures in modern times (90 %). Based on this conclusion, Hitachi-GE's overall case is that adopting the limits proposed by their fuel failure review, as well as large parts of the EPRI's guidelines (Ref. 45), into the generic safety case for UK ABWR, in combination with using a modern fuel clad material (Zircaloy-2, P9), means the risk of fuel clad failure in UK ABWR owing to the operating chemistry, is small. This is an important overall claim, which I have assessed the relevant parts of below.
517. The overall trend for worldwide BWR fuel failure rates between 1994 and 2006 is shown in Figure 28, below:

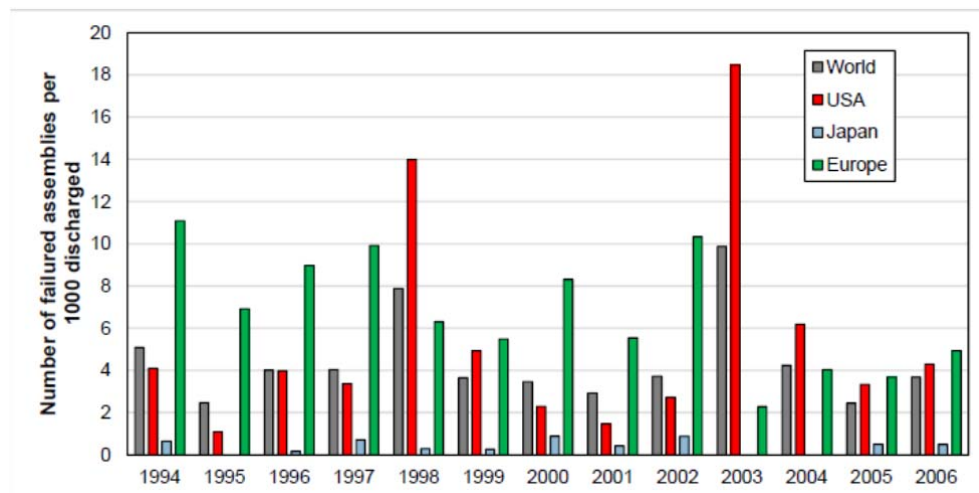


Figure 28: Worldwide BWR fuel failure rates (Ref. 94).

518. Data presented in Figure 28 does not distinguish between fuel failure mechanisms, but the trend clearly points towards the fact that BWR fuel reliability has increased significantly in recent times. Ref. 95 where Figure 28 is reproduced from does however provide a further breakdown of the fuel failure data, by mechanism. From 1990 and later, this information shows debris failures were consistently one of the top two leading causes of BWR fuel failures (Ref. 94). I am therefore satisfied with Hitachi-GE's overarching claim that chemistry is largely no longer implicated in modern day BWR fuel failure events. This conclusion is also supported by the independent work performed by my TSC (Ref. 94).
519. As I mentioned earlier, the most recent and infamous chemistry-related BWR fuel failure events occurred at the US BWR plants, River Bend and Browns Ferry, during

the late 1990s and early 2000s, respectively. The overall impact of these events on BWR fuel failure rates is clear from Figure 28. Although I agree with Hitachi-GE's overarching conclusion that chemistry is no longer implicated in significant numbers of BWR fuel failures, the River Bend and Browns Ferry events serve as important reminders that adequate control over the operating chemistry must be maintained to avoid fuel reliability and/or failure events.

520. Although the cause(s) of the River Bend fuel clad failures were not definitively identified, following a number of investigations and extensive fuel examination campaigns, the evidence points towards high copper, DZO, iron and silica levels in the reactor coolant. The presence of these particular elements in higher concentrations than normal was implicated in the formation of tenacious fuel deposits, which ultimately led to the fuel cladding failures. The formation of tenacious fuel deposits was also implicated in the fuel clad failures experienced at Browns Ferry. Here, the largest constituents of the fuel deposits were iron and zinc. In response, a number of measures have been successfully adopted by operating BWRs. They include tighter controls over the Zircaloy-2 cladding material, and better control of feedwater DZO and iron concentrations. My detailed assessment of the impact of, and control over impurities in the UK ABWR reactor coolant, presented below, therefore gives special attention to these parameters. What is clear is that chemistry is not implicated in BWR fuel failure events, as long as adequate water chemistry controls are in place.

4.3.4.1 Chemistry Control to Minimise Fuel Cladding Corrosion

Impact of HWC on Fuel

521. When purposefully injecting hydrogen, the propensity for the fuel degradation mechanism known as hydriding, needs to be considered. This was one of the potential degradation mechanisms identified independently by my TSC (Ref. 94). The UK ABWR generic safety case acknowledges the mechanism, but argues that based on many decades of OPEX for BWR plants running HWC, some with feedwater hydrogen concentrations as high as 2 ppm, no issues with fuel degradation caused by hydriding have been reported (Ref. 93).
522. As shown in Annex 5 of my report, the upper limit value for feedwater hydrogen defined in the UK ABWR generic safety case is 0.5 ppm (i.e. substantially lower than 2 ppm). Furthermore, limit values for the UK ABWR feedwater hydrogen ORs are not set on the basis of fuel integrity concerns. Based on my own knowledge of the topic, and the independent review undertaken by my TSC (Ref. 94), I am satisfied with the arguments presented by Hitachi-GE regarding the impact of this fuel failure mechanism for UK ABWR. For a generic safety case, I am content adequate information has been provided to support claim [RC SC3.4]

Impact of DZO on Fuel

523. In Ref. 93) and Ref. 31, Hitachi-GE state the main considerations for the impact of DZO on fuel are primarily to avoid fuel deposit spallation and auto-catalytic corrosion, including avoiding thick deposits of zinc ferrite and zinc silicate. Based on my own knowledge of BWR fuel failures and the independent work undertaken by my TSC (Ref. 94), I agree these are the principal aspects of DZO chemistry and its impact on fuel, which need to be considered.
524. Based on these considerations, Hitachi-GE propose several limits for DZO in UK ABWR. The ORs presented in the *Water Quality Specification* (Ref. 20) and Chapter 23 of the PCSR (Ref. 11) (re-produced in Annex 5 of my report), include a minimum "limit" on DZO reactor water concentrations of ≥ 2 ppb, and a maximum limit for feedwater DZO of ≤ 3 ppb. "Expected values" are also provided. The minimum "limit"

for DZO in reactor water is based on radioactivity considerations. The focus of my assessment here is therefore the adequacy of the proposed DZO limit for feedwater.

525. During GDA, Hitachi-GE provided OPEX to demonstrate plants which have experienced short-term excursions of DZO up to this level, have not experienced fuel failure or reported any associated fuel integrity concerns. On this basis, Hitachi-GE argue the proposed limit is conservative, because beyond this value there may be unknown consequences. Furthermore, the overall basis for these instantaneous limit values for reactor water and feedwater DZO, include several quarterly and cycle average values for DZO limits, below which BWRs have not experienced fuel integrity concerns.
526. The values provided are familiar to me and are consistent with authoritative industry guidelines on BWR fuel integrity matters (Ref. 44 and 45). I also asked my TSC to consider the appropriateness of these limits. Based on their independent review they broadly concur with Hitachi-GE's conclusion (Ref. 94). On the basis of my own knowledge of this topic and the independent advice from my TSC, I am satisfied with the limit values proposed for DZO, with respect to minimising fuel failure risks. The one aspect I am not content with is how the quarterly average values are reflected throughout the relevant UK ABWR generic safety case documentation. These values are clearly very important, yet they are inconsistently presented throughout the case. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**
527. Another positive part of Hitachi-GE approach to deriving and justifying the controls for DZO is to consider the impact of DZO, in combination with other chemistry parameters; namely iron (Ref. 93). They draw on information presented by the EPRI, which uses the CORAL model. This model is used by some parts of the BWR industry to model the impact of CPs on fuel deposits. The method uses OPEX from the River Bend BWR, obtained during a fuel cycle when crud-induced fuel failures occurred. These data and modelling are used to determine the likely onset of crud-induced auto-catalytic cladding corrosion, as a function of plant feedwater iron and DZO levels. Based on the output of this work, the EPRI provide recommended limit values for BWR feedwater iron and DZO concentrations, to avoid fuel degradation. These have been adopted by Hitachi-GE in the relevant limits proposed for DZO in the generic safety case.
528. Considering the OPEX-led nature and the pedigree associated with this aspect of Hitachi-GE's approach to underpin the DZO limits, I am satisfied this is suitable for GDA. Overall, I am content Hitachi-GE has provided suitable and sufficient evidence to support claim [RC SC3.2]. I have assessed Hitachi-GE's specific approach for iron in Section 4.3.4.2, below.

Impact of OLNC on Fuel

529. As explained earlier in Section 4.3.2.1 of my assessment report, the basis for the limits and conditions specified by Hitachi-GE for OLNC are dominated by the potential impact of platinum on fuel.
530. Claim [RC SC3.1] reflects the importance of safely controlling this aspect of the UK ABWR operating chemistry; I consider making this claim is appropriate and necessary. Hitachi-GE's safety case evidence to support claim [RC SC3.1] is largely contained in Ref. 93 and Ref. 32. These reports present information from a number of sources to justify several limits for platinum deposition on fuel, which are based on platinum loadings per unit surface area of fuel cladding. Based on these maximum allowable platinum loadings, Hitachi-GE perform a simple calculation based on the surface area of the UK ABWR fuel and core, to work out an equivalent yearly limit for the maximum amount of platinum permitted to be added to UK ABWR. The platinum loading limits are presented in the *Water Quality Specification* (Ref. 20) and the corresponding

- annual limit of 920 g of platinum is also presented in Chapter 23 of the PCSR (Ref. 11).
531. As well as restrictions on platinum mass, as already explained in Section 4.3.2.1, Hitachi-GE also propose several conditions associated with conducting OLNLC. The one which is most pertinent to my assessment here is the restriction of not performing OLNLC within the first 60 days of the fuel cycle. The basis for this requirement is essentially to allow “fresh” fuel added to the core during the previous outage to form a protective oxide layer and avoid direct contact of platinum on “fresh”, un-oxidised fuel surfaces.
532. Hitachi-GE argue the limits and conditions derived for platinum (OLNLC) for GDA are adequate because:
- They are based on many cycles of BWR plant OPEX which shows operating below these limits results in no adverse impact to fuel.
 - They are supported by results from the Duane Arnold Experiment Centre, NMCA demonstration programme.
533. The quantitative limits and associated conditions proposed by Hitachi-GE are familiar to me and are consistent with the work on this topic presented in other sources of independent information available to me (Ref. 44). Furthermore, I also asked my TSC to undertake an independent review of Hitachi-GE’s proposed limits. Their work also supports Hitachi-GE’s assertion that the limits proposed for platinum have not resulted in BWR fuel failures (Ref. 93). Overall, on this basis, I am therefore broadly satisfied Hitachi-GE has provided suitable and sufficient evidence during GDA to support claim [RC SC3.1].
534. Despite this conclusion, as I have already noted in Section 4.3.2.1, there are some gaps in the generic safety case in Hitachi-GE’s overall approach to defining an adequate set of ORs to control the HWC and OLNLC aspects of the UK ABWR operating chemistry. This has already been captured in Assessment Finding, **AF-ABWR-RC-02**. However, for the OLNLC aspects of the operating chemistry, considering these ORs are derived based on fuel integrity concerns, clearly **AF-ABWR-RC-02** is also directly relevant here too.

Impact of Copper on Fuel

535. As already explained during my assessment of radioactivity, earlier BWR designs with main condensers tubed in brass, experienced elevated levels of copper in the reactor coolant. For some of these plants which also experienced fuel failures; some failures attributed to the CILC mechanism were associated with elevated levels of copper.
536. Positively, the UK ABWR generic design and safety case recognise the detrimental impact elevated levels of copper can have on fuel. The main condensers are tubed in titanium, which eliminates the main source of copper ingress present in older BWRs. Furthermore, in the context of chemistry and fuel, claim [RC SC3.5] specifically mentions the importance of adequately controlling copper. In terms of specific controls on impurities, it is also notable that copper is the only specific impurity formally reflected in the feedwater ORs presented in Chapter 23 of the PCSR (Ref. 11). (See Annex 5 of my report). On this basis, I am satisfied UK ABWR should be able to operate to achieve low levels of copper in the reactor coolant, and I am also content the generic safety case provides the necessary visibility of the importance of the impact of copper on fuel.
537. During GDA, I therefore focussed my assessment (RQ-ABWR-1040 (Ref. 49)) on the adequacy and basis for, the quantitative limits proposed by Hitachi-GE for copper in

the *Water Quality Specification* (Ref. 20). Hitachi-GE assert the values specified are adequate because:

- They are derived based on an evaluation of relevant BWR OPEX for plants which have not experienced fuel failures when operating below the levels of copper, specified.
- UK ABWR will use Zircaloy-2 P9 fuel cladding material, which has been specifically selected for its very high corrosion resistance and resistance to nodular-corrosion-related fuel failures. Hitachi-GE argue there have been no reported P9 fuel cladding corrosion-related fuel failures in BWRs, even for plants with main condensers still tubed in brass.

538. On this basis, I am satisfied with the discrete limit(s) proposed by Hitachi-GE for copper. This is also supported by the independent work undertaken by my TSC on this topic (Ref. 94).

539. As described earlier, analysis of the River Bend fuel failures and the tenacious fuel deposits formed there, showed not only copper was implicated, but also other metallic species, including iron and DZO. As part of their BWR plant OPEX review, Hitachi-GE do recognise the importance of considering the controls specified for copper, in combination with the controls specified for both iron and DZO. They state limit values on a quarterly cycle average basis for iron and relate this value back to quarterly cycle averages for a combination of both DZO and copper. This is positive. However, the outcome of this work does not appear to be reflected in the *Water Quality Specification* (Ref. 20). This is related to my earlier finding, raised during my assessment of the controls specified for feedwater iron and its impact on radioactivity. Controls which related iron to other chemistry parameters of importance in the outputs from the generic safety case were not provided. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

Impact of Other Impurities on Fuel

540. During GDA, I also assessed the impact of other chemistry parameters which may be important for maintaining fuel integrity, including: silica, sodium, lithium and “aggressive” anions (chloride and sulphate). My assessment for these parameters is presented in the following paragraphs.

Other Impurities – Silica

541. Hitachi-GE claim no BWR fuel failures have been directly attributed to the presence of silica alone. Nevertheless, for UK ABWR, they recommend reactor water silica is controlled to maintain a quarterly average concentration of ≤ 300 ppb. I agree it is necessary to specify specific controls over this parameter because as previously explained; zinc silicate has been previously implicated in past BWR fuel failures. However, the 300 ppb value suggested by Hitachi-GE does not appear to be reflected in the expected and limit values for reactor water silica given in the *Water Quality Specification* (Ref. 20); which are significantly higher, at [REDACTED] and ≤ 9000 ppb, respectively.

542. During GDA, I therefore asked (RQ-ABWR-1042 (Ref. 49)) Hitachi-GE several queries about the controls suggested for silica and their presentation in the generic safety case. In response, Hitachi-GE assert the 300 ppb quarterly average value for silica is based on the results of fuel inspections for BWR plants which have routinely operated below this value. On this basis, I am satisfied with this aspect of the silica control measures specified during GDA. This is also supported by the independent work performed by my TSC (Ref. 94).

543. The aspect of the silica controls suggested during GDA which I am not satisfied with, is how the different “limits” are presented in the generic safety case. The limiting factor for silica control appears to fuel failure concerns. However, the silica control values provided in the *Water Quality Specification* (Ref. 20) are based on materials integrity concerns, which appear to have less stringent requirements. As silica is not directly implicated in fuel failures alone, and given that I am satisfied with the basis of the 300 ppb control suggested by Hitachi-GE, I am content with this position for GDA. However, considering the importance of having suitable ORs as outputs which adequately reflect the requirements of the underlying safety case, I consider this should be addressed by a future licensee. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

Other Impurities – Sodium

544. Hitachi-GE claim (Ref.93) short-term exposure of fuel to elevated levels of sodium is not particularly significant. There are therefore no specific water chemistry controls proposed for UK ABWR, which are directly related to the impact of sodium on fuel.
545. Based on my knowledge of the topic, I do not necessarily disagree with this assertion. However, there are some limited examples from OPEX for Japanese BWRs where sodium has been implicated as a possible factor in corrosion-related fuel failures. During GDA, I therefore asked (RQ-ABWR-1040 (Ref. 49)) Hitachi-GE to justify why their position for the impact of sodium on fuel, is reasonable. Hitachi-GE argue:
- Authoritative industry guidelines on BWR fuel reliability and performance (Ref. 45) issues provide evidence which shows short-term exposure of fuel to elevated levels of sodium hydroxide, does not result in any significant failure concerns.
 - Non-regenerative operation of the UK ABWR chemical clean-up systems is proposed, and implementing appropriate controls over conductivity, will provide the necessary control to be able to detect and manage sodium ingress events.
546. On this basis, I am satisfied with the arguments presented by Hitachi-GE.

Other Impurities – Lithium

547. Compared to sodium, lithium is an impurity with the potential to be more significant in terms of its potential impact on fuel, if present in elevated quantities in the reactor coolant. This is recognised by Hitachi-GE in the generic safety case, which states the primary source of lithium in UK ABWR would be as the result of leaking boron carbide control rod blades. Ref. 93 explains lithium can significantly corrode Zircaloy fuel cladding if it becomes concentrated on fuel surfaces and that even at very low levels, lithium could start to concentrate in fuel clad deposits and become a corrosion concern. Despite these concerns, there are no specific control parameters for lithium identified by Hitachi-GE in the *Water Quality Specification* (Ref. 20). During GDA, I asked (RQ-ABWR-1040 (Ref. 49)) Hitachi-GE why this is appropriate. They argue:
- The technical basis for investigating the impact of lithium on BWR Zircaloy fuel cladding is largely based on laboratory experiments. Hitachi-GE say this has shown to be not representative of BWR plant OPEX; which demonstrate the effect of lithium on Zircaloy corrosion is not as significant.
 - With respect to the main potential source of lithium, the control rod blades, boron and tritium trending are more appropriate controls for detecting and monitoring control rod blade degradation.
548. On this basis, and given the fact that Hitachi-GE’s approach for lithium is consistent with authoritative industry guidelines on BWR fuel reliability and performance (Ref. 45), I accept their argument for lithium control, with respect to its impact on fuel. However,

for control rod blades, I note Hitachi-GE's *Management of Damaged Fuel Report* (Ref. 91) states, "Following [power suppression] and determination of the defect location, the lifetime of the control rods being considered as suppression rods should be ascertained. In most cases, sufficient lifetime is remaining for the selected control rod(s)". Despite this requirement, the *Water Quality Specification* (Ref. 20) does not capture any chemistry-based control parameters, which could potentially include lithium. I expect a future licensee to give further consideration to this aspect of the UK ABWR generic safety case, as it is developed further. **This has been captured as part of Assessment Finding, AF-ABWR-RC02.**

Other Impurities – Chloride and Sulphate

549. I have already assessed the impact of chloride and sulphate on the integrity of other UK ABWR SSCs, in Section 4.3.2.1 of my report. These "aggressive ions" can also impact the integrity of fuel, therefore suitable and sufficient controls need to be in place. The basis for the controls on chloride and sulphate in the generic safety case is their impact on RINs, not fuel. During GDA I asked (RQ-ABWR-1040 (Ref. 49)) Hitachi-GE why this approach is adequate. Hitachi-GE argue the more limiting concern is the impact of chloride and sulphate on SSC of RINs, therefore the controls specified for these parameters are justified on this basis. As a philosophy, I am satisfied with Hitachi-GE's approach. As I have concluded in Section 4.3.2.1 though, I am not yet satisfied Hitachi-GE has provided an adequate justification for the limit values defined for chloride and sulphate in UK ABWR. For RINs and fuel, as these parameters are inextricably linked, this means my earlier conclusion is also relevant here. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

Impact of Commissioning Chemistry on Fuel

550. At the top level of the generic safety case, Chapter 23 of the PCSR (Ref. 11) captures the importance of commissioning chemistry on minimising fuel failure risks in claim [RC SC3.6], which I consider is a particularly positive aspect of Ref. 11.
551. As well as the PCSR (Ref. 11), Hitachi-GE also recognise the importance of chemistry control during commissioning in the lower tiers of the generic safety case. In Ref. 38, Hitachi-GE acknowledge the importance of removing foreign materials, debris and swarf etc., to minimise the risk of fuel failures once the first core of fuel is loaded. Ref. 38 recommends implementing the largest number of measures (five off) during phase one of commissioning, including: system flushing, wiping, filling and draining, temporary strainers and water jet cleaning.
552. Ref. 38 also explains that during phase three of commissioning, the risk of fuel failures increases significantly owing to the potential for wash-out of impurities from steam line piping and turbine sections of the plant, which have not previously experienced water flows during the earlier phases of commissioning. Compared to phase one, only three measures aimed at minimising the risk of fuel failures during phase three of commissioning are suggested by Hitachi-GE: using high purity water, clean-up using the CUW system and temporary strainers. Considering phase three represents the point in time when fuel is first loaded to the core, from the information presented during GDA, it isn't immediately obvious why additional measures cannot be taken during earlier phases of the commissioning cycle. **This has been captured as part of Assessment Finding, AF-ABWR-RC-03.**
553. Nevertheless, considering the root cause of the vast majority of BWR fuel failures is mechanical wear from debris fretting, I am satisfied the measures suggested by Hitachi-GE during GDA for commissioning, represent a suitable starting point to begin to address the risk of fuel failure by this particular mechanism.

554. Positively, based on US and Japanese OPEX, the generic safety case also presents typical water quality specifications for BWR commissioning activities. This is a welcome addition to the *Water Quality Specification* (Ref. 20) and I am satisfied the information presented during GDA represents an adequate starting point for a future licensee.
555. Overall, I am broadly satisfied for GDA, Hitachi-GE has provided suitable and sufficient evidence to support claim [RC SC3.6]. It is clear though there is still significant and important work for a future licensee to conduct beyond GDA, to ensure this claim can be met in practice. This has been captured as part of Assessment Finding, **AF-ABWR-RC-03**.

4.3.4.2 Chemistry Control and Fuel Deposits

556. As previously explained, because, by definition, boiling occurs in the core of a BWR, up to 90% of the impurities present in the reactor coolant deposit on fuel surfaces. This part of my assessment of chemistry and fuel therefore focusses on the nature and quantity of fuel deposits expected for UK ABWR. In this section I have focussed mainly on the impact of iron, because it makes the largest contribution to fuel deposits in BWRs.
557. Considering the overall importance of adequate iron control to maintaining fuel integrity, it is particularly positive to see Chapter 23 of the PCSR (Ref. 11) makes a specific claim, [RC SC3.3], on this parameter. Hitachi-GE also provide further information to evaluate the impact of iron on fuel deposits in Ref. 93. In addition, during GDA, they also performed a series of calculations to model the expected behaviour of CPs in UK ABWR. The model and results are presented in the report, *Corrosion Products Behaviour Model and Calculation Results for UK ABWR* (Ref. 89). With respect to fuel deposits, Hitachi-GE performed several studies to evaluate the impact of feedwater iron. Their analysis demonstrates how the total mass of fuel deposits expected for UK ABWR is particularly sensitive to feedwater iron. For example, at 0.1 ppb feedwater iron, fuel deposits are estimated to be [REDACTED]. At 0.5 ppb, they are estimated to reach [REDACTED].
558. As part of my TSC's work, based on measured plant chemistry data, I asked them to provide independent information on estimated fuel deposit loadings for several operating BWRs. These data are also broadly consistent with the outputs from Hitachi-GE's work, in that the same trend in fuel deposit total mass loading is very sensitivity to feedwater iron.
559. In addition to the potential impact of elevated levels of feedwater iron on fuel, Hitachi-GE also consider Ref. 93 the impact of low iron, and a particularly unusual fuel corrosion mechanism known as Enhanced Shadow Spacer Corrosion ESSC. Hitachi-GE argue this particular mechanism will not be a concern for UK ABWR, because its prevalence has been restricted to a single Swiss BWR (Kernkraftwerk Leibstadt), which operated with low iron concentrations. Furthermore, they argue the use of modern Zircaloy-2 P-9 fuel cladding material should further reduce the risk, on the basis that it has only been observed in older cladding materials. While there is some considerable uncertainty over the precise mechanism, I am satisfied Hitachi-GE's arguments are reasonable in this regard, mainly because they are consistent with the information available in the open literature on this topic (Ref. 45). Furthermore, my own conclusion is supported by the independent work performed by my TSC (Ref. 94).
560. Based in some part on the work above, Hitachi-GE's proposed limit for feedwater iron (of ≤ 50 ppb) is set on the basis of fuel integrity concerns being the limiting factor. I have already explained in Section 4.3.3.5 that I consider this limit may be too high. This has been captured as part of Assessment Finding, **AF-ABWR-RC-02**. Nevertheless, the limit proposed by Hitachi-GE during GDA has been based on BWR

OPEX for plants where spikes in iron have occurred (although with no resulting fuel failures). In this regard, I consider this aspect of Hitachi-GE's justification for the feedwater iron limit is acceptable for GDA. However, it will be important for a future licensee to re-visit and further justify this limit as part of responding to **AF-ABWR-RC-02**.

561. In the context of claim [RC RC3.3], based on my overall assessment above, I am satisfied Hitachi-GE has provided suitable and sufficient information for a generic safety case. Hitachi-GE's approach to estimating the likely fuel deposits for UK ABWR appear reasonable. An important point to note is the continued relevance of Assessment Finding, **AF-ABWR-RC-02** to this topic and also **AF-ABWR-RC-11**. In responding to these, I also expect the future licensee to give adequate consideration to the inter-relation of the fuel topic.
562. In *Topic Report on Noble Metal Chemical Addition (NMCA)* (Ref. 32) Hitachi-GE also acknowledge other potential detriments associated with platinum injection. This includes some instances where some BWRs have experienced fuel deposit spalling. I have already assessed the basis for the limits proposed for platinum and concluded they are reasonable. However, Ref. 32 also notes another specific impact of platinum injection is that the timing can influence the distribution of fuel deposits across the core. Considering OLNK will be applied from the very first fuel cycle, Hitachi-GE argue they expect the fuel deposits for UK ABWR to be bottom peaked^{††}. Considering the inherent uncertainties with applying OLNK technology for the first time to an ABWR, this argument is not particularly robust. Furthermore, it is not supported by the independent work performed by my TSC (Ref. 94), who suggest that fuel deposits may be bottom peaked, but for a different reason (heat flux profile). Nevertheless, despite these differences, the predicted trends for UK ABWR fuel deposits are in broad agreement (i.e. bottom peaked). On this basis, I am satisfied with the information provided during GDA.
563. Overall, considering the inherent uncertainty in radioactivity generation and transport modelling for BWRs; and the overall status of the technical knowledge, internationally, in this field, I am satisfied with the information provided by Hitachi-GE during GDA to justify the likely fuel deposits for UK ABWR.

4.3.4.3 Chemistry and the Detection and Mitigation of Fuel Failures

564. *Management of Damaged Fuel* (Ref. 91) is Hitachi-GE's principal document which discusses the detection and physical mitigation measures that would be implemented, if fuel failures were to occur in UK ABWR. Ref. 91 demonstrates that US, Japanese and some EU BWRs, use the same method to detect and manage fuel failures in the core. This process, known as Power Suppression (Testing), was previously described in Section 4.3.3.5 of my report, where I assessed the impact on radioactivity. My assessment in this section is focussed on the wider impact implementing Power Suppression may have on fuel risks and any associated requirements which may be placed on the operating chemistry to implement the practice.
565. ONR's overall decision regarding the adequacy of Power Suppression to manage failed fuel in UK ABWR is led by the fuel and core specialist. Chemistry has a much narrower interest in the practice, namely, to assess the evolution of the potential fission product source term(s) and the associated requirements to undertake (and frequency of) undertaking grab sampling. This includes determining whether any claims are made on the requirement to implement additional controls over particular chemistry and/or radiochemistry parameters. From a regulatory perspective, the important questions to

^{††} The term bottom-peaked refers to the distribution of fuel deposits along the length of the fuel rod. In this case, more deposits are expected on the bottom parts of the fuel cladding surface.

answer during GDA therefore concern the overall acceptability of implementing Power Suppression for UK ABWR, and the extent to which the generic safety case has adequately captured the necessary requirements, to enable a future operator to safely implement the practice.

566. Bearing the above in mind, I note ONR's fuel and core inspector has accepted Hitachi-GE's overall case to implement Power Suppression (Ref. 96). In terms of implementing the practice, Ref. 91 identifies several "limits" on key parameters, including power distribution and thermal requirements. Furthermore, from a chemistry perspective, Ref. 91 also presents several (three) criterion which a future plant operator should apply to detect and manage failed fuel in the core.
567. Potential fuel failures are detected by increases in the release rate of several FP noble gases at the OG system⁵⁵ and increases in the concentrations of certain radionuclides in the reactor coolant. For chemistry, the most relevant aspects of Hitachi-GEs Power Suppression criteria relate to the requirements to determine the concentration of ¹³¹I in the coolant and ¹³³Xe and ¹³³Xe/¹³⁸Xe activity levels at the OG system. The criteria provide clear applicable "limits" for these parameters and importantly, also link back and are consistent with, the relevant LCO determined for permitted radioactivity levels in the coolant. I am satisfied the information presented represents a suitable starting point for a future licensee to further develop their own arrangements for managing failed fuel in the core. The requirements and radiochemical parameters identified are what I would expect to see.
568. The other very important aspect regarding the implementation of Power Suppression is providing the necessary evidence to demonstrate the practice does not lead to significant secondary fuel failures. During GDA, Hitachi-GE provided evidence which demonstrates no further fuel pin degradation has been reported at BWR plants which use the limits and criteria presented by Hitachi-GE in the generic safety case, to implement power suppression. I also asked Hitachi-GE (RQ-ABWR-1039, -1044 (Ref. 49)) how chemistry control may impact the potential for secondary fuel failures and/or whether any additional, or more stringent chemistry controls are necessary, when power suppression is being implemented. Hitachi-GE's response was that no additional controls are required. They also asserted the operating chemistry and the propensity for secondary fuel failures are independent from one another. Based on my own assessment and the independent work undertaken by my TSC (Ref. 94), I am satisfied with Hitachi-GE's response.
569. Overall, from a chemistry perspective, I am satisfied with the case presented by Hitachi-GE to justify the impact of chemistry and/or radiochemistry on managing the potential for failed fuel in UK ABWR. The generic safety case provides a suitable starting point for a future operator to develop their plans to implement the practice of power suppression, should they choose to. However, it is clear that key operational decisions will be required regarding the practical implementation of power suppression and the associated continued demonstration that relevant risks are reduced SFAIRP. Based on my assessment above, I am satisfied this is a matter of normal business.

4.3.4.4 Chemistry and Fuel Integrity and Performance Summary

570. During GDA I have assessed three main aspects of the operating chemistry and its impact on fuel integrity and performance for UK ABWR:
- Controlling chemical additions and excluding impurities to minimise deleterious effects on fuel cladding and corrosion.

⁵⁵ Also known as the "sum of six". This calculation is commonly used in BWRs to detect fuel failures by determining measured radionuclide release rates for six fission product noble gases at the OG system.

- Minimising the build-up of fuel deposits, SFAIRP, to ensure safety margins are not eroded.
 - Chemistry and radiochemistry related to the detection and management of failed fuel.
571. These topics are considered in detail by Hitachi-GE in several submissions in the generic safety case. Furthermore, Chapter 23 of the PCSR (Ref. 11) makes several key explicit claims on this topic. I am broadly satisfied Hitachi-GE has presented suitable and sufficient evidence, for a generic safety case, to meet the intent of these claims.
572. Hitachi-GE have considered the impact of implementing the “new” proposed operating chemistry for UK ABWR and have undertaken a comprehensive review of BWR fuel failure mechanisms. This shows chemistry is not implicated in modern BWR fuel failures, as long as key chemistry parameters are adequately controlled. Based on my assessment, supported by the work of an independent TSC with extensive knowledge in the field of BWR chemistry and fuel reliability, I agree with this conclusion.
573. Hitachi-GE’s case also provides relevant supporting evidence, largely informed by OPEX from operating BWR plants and authoritative industry guidelines (Ref. 44, 45 and 94), as the basis to justify both the most important chemistry control parameters to control to maintain fuel reliability, but also suitable operating limits for them. This approach is adequate and fit-for-purpose for a generic safety case.
574. I have identified a number of areas where the future licensee will need to give further consideration to, and justify specific aspects of the UK ABWR design and safety case related to chemistry and fuel. These are discrete Assessment Findings, but have been captured, as appropriate, as part of Assessment Finding, **AF-ABWR-RC-02, -03** and **-11**. Considering a large proportion of the generic safety case claims for chemistry and fuel specifically mention limits and conditions, **AF-ABWR-RC-02** is particularly important. Nevertheless, I am content that a future licensee will be able to resolve the fuel related matters of these satisfactorily and I judge that they do not undermine the UK ABWR generic design.
575. Overall, I am satisfied the intent of the relevant SAPs, ECH.1, ECH.2, ERC.1 and ERC.3, have been broadly satisfied.

4.3.5 Chemistry Control Systems

576. Having assessed Hitachi-GE’s choices and justifications for the chemistry to be applied for UK ABWR across the different operating modes in Sections 4.3.1.1, 4.3.1.2 and 4.3.1.3 of my report, the next section presents my assessment of the adequacy of relevant UK ABWR systems which control the chemistry of the RCS within the limits and conditions (or ORs) defined in the generic safety case. There are broadly three categories of UK ABWR systems which fall into this area of my assessment:
- Systems which are purposefully designed to inject the chemical additives necessary to implement the chosen UK ABWR operating chemistry.
 - Systems which provide a clean-up function to remove chemical impurities and/or radioactivity from UK ABWR, and;
 - Systems used to sample and monitor chemistry parameters during normal operations.
577. My assessment below is therefore divided into three separate sub-sections to represent this division. Most of my assessment considers normal operations, namely start-up, at-power, shutdown and outage periods. However, I also specifically consider commissioning as a separate sub-topic for each of the above, given the requirements and capabilities for chemistry control are very different at this time.

578. There is a broad range of relevant safety case documents including topic reports, BoSCs and SDDs which provide pertinent information on these systems. These are identified in my assessment that follows, but the *Topic Report on Design Justification in Chemistry Aspect for Primary Water Systems* (Ref. 18) is of particular note. Ref. 18 is a fundamental part of the generic safety case. The aim of Ref. 18 is to provide relevant information which justifies the chemical performance of numerous UK ABWR systems which perform a chemistry-based safety function.
579. The main SAPs (Ref. 2) of relevance to this part of my assessment are therefore ECH.1, ECH.3 and ECH.4, and the guidance provided in TAG-088 (Ref. 4).

4.3.5.1 Chemical Injection Systems

580. The scope of the Chemical Injection System in UK ABWR covers the majority of the systems that are required to implement (inject the necessary chemicals) the operating chemistry of HWC with OLN and DZO injection. The sub-systems covered by the Chemical Injection System include the:

- Hydrogen and Oxygen Injection System (HOIS);
- Noble Metal Injection System (NMIS);
- Zinc Injection System (ZNIS); and
- Oxygen Injection (OI) system.

581. All of these systems are described to some extent in Chapter 23 of the PCSR (Ref. 11). Further detail is provided in lower tier safety case documentation, including: *Topic Report on Design Justification in Chemistry Aspect for Primary Water Systems* (Ref. 18), *Topic Report on Noble Metal Chemical Addition (NMCA)* (Ref. 32), *Topic Report on Zinc Injection* (Ref. 31) and the *Chemical Injection System SDD* (Ref. 97).
582. As I would expect to see, Chapter 23 of the PCSR (Ref. 11) has a number of specific formally identified claims concerning the need to adequately control the injection of hydrogen, oxygen, platinum and DZO into UK ABWR. A number of the claims are also themed around the main argument that purposefully injecting these chemicals has no detrimental effects and that on balance any detriments are controllable and mitigated. In the context of the Chemical Injection System, I consider these claims to be reasonable.

Hydrogen and Oxygen Injection System

583. The HOIS is the system which injects hydrogen into the CFDW system to enable the UK ABWR to implement HWC. Only very high-level information on the HOIS was provided during GDA, a design specific to the requirements of UK ABWR was not provided. Nevertheless, the HOIS presented during GDA is based on existing technology already used for other BWR plants. For UK ABWR, the HOIS provides two main functions:
- Injection of hydrogen into the main feedwater line at the inlet of the High Pressure Condensate Pump (HPCP).
 - Injection of oxygen at the inlet of the OG system heater, before the OG recombiner.
584. This latter function is because the oxygen concentration in the main steam decreases when HWC is implemented. Additional oxygen is therefore necessary to ensure effective hydrogen and oxygen recombination in the OG system. The HOIS therefore includes cascaded control, where these two functions are linked. The HOIS mainly comprises a “hydrogen supply device”, “oxygen supply device” and associated piping, valves and instrumentation etc., which are necessary to inject hydrogen into the CFDW system and oxygen into the OG system. Hitachi-GE’s categorisation and classification

of the HOIS presented in Chapter 23 of the PCSR (Ref. 11), is C3. Considering the claims made in the generic safety case and safety functions provided by the HOIS, from a chemistry perspective, I am satisfied C3 is an appropriate categorisation and classification.

585. Considering the change in the “new” proposed operating chemistry for UK ABWR, Hitachi-GE identified the requirement to install hydrogen meter(s) in the main feedwater line. Considering feedwater hydrogen is identified as a control parameter in the UK ABWR generic safety case, I consider this modification to be both necessary and beneficial. Despite the introduction of hydrogen meter(s), control over hydrogen injection for UK ABWR is still planned to be performed using the HOIS. In addition, during GDA, Hitachi-GE asserted (response to RQ-ABWR-0783) it is not necessary to install oxygen meters in the OG system because control over oxygen injection is achieved by the cascaded arrangement with feedwater hydrogen (Ref. 49). For GDA, I am satisfied with position, but I would expect a future licensee to give further consideration to these conclusions as the detailed design for the UK ABWR Chemical Injection System is developed. **This has been captured as part of Assessment Finding, AF-ABWR-RC-13, below.**
586. In summary, the type of information provided by Hitachi-GE during GDA on the HOIS was of insufficient detail to undertake a meaningful assessment of the adequacy of the HOIS against the expectations of SAP ECH.3, amongst others. I do not consider *Topic Report on Design Justification in Chemistry Aspect for Primary Water Systems* (Ref. 18) provides what I would regard as an adequate justification for the design of a HOIS for UK ABWR. There are very important aspects of any such justification missing. While I do not expect a detailed design for GDA, I would have expected to see more detail than Hitachi-GE have provided. **This has been captured as part of Assessment Finding, AF-ABWR-RC-13, below.** Despite this, because the information provided on the HOIS for GDA is based on existing BWR technology, this suggests that if a similar system were made specific to the requirements of UK ABWR, it may deliver the required volumes and control over feedwater hydrogen (and oxygen at the OG system).

Noble Metal Injection System

587. The design of the NMIS provided by Hitachi-GE during GDA is based on a proprietary system used by many US BWRs to inject platinum. The main function of the NMIS is to inject a 1% solution of platinum as the complex $\text{Na}_2\text{Pt}(\text{OH})_6$, directly into the feedwater line. The system comprises injection, analysis and data acquisition functions, mounted on a portable skid. Hitachi-GE’s categorisation and classification of the NMIS presented in Chapter 23 of the PCSR (Ref. 11), is C3. Considering the claims made in the generic safety case and safety functions provided by the HOIS, from a chemistry perspective, I am satisfied C3 is an appropriate categorisation and classification.
588. Unlike Ref. 18, *Topic Report on Noble Metal Chemical Addition* (Ref. 32) provides a lot more information on the NMIS, including descriptions of the individual pieces of equipment, how such systems are generally operated and which chemistry parameters are monitored and/or trended during platinum injection. In this regard, it is actually Ref. 32 and not *Topic Report on Design Justification in Chemistry Aspect for Primary Water Systems* (Ref. 18), that provides information which most closely resembles a justification for the design of the NMIS presented during GDA.
589. In Ref. 32, the NMIS is identified as “*permanently installed equipment*”, but the report explains for multi-unit reactor sites it may be categorised as “*temporary plant equipment*”, and therefore moved between reactor units to conduct platinum injection, as and when required. Given the NMIS is based on existing BWR technology, which did not commence OLNLC from the very first fuel cycle, Hitachi-GE did not justify why a portable arrangement is most suitable for a “new” BWR design like UK ABWR. Hitachi-

GE's overall conclusion in Ref. 32 is that because the NMIS presented in GDA has been used at lots of BWRs to successfully inject platinum, the design is adequate for UK ABWR.

590. Overall though, for GDA, I cannot yet conclude whether or not the NMIS will provide the necessary level of control over platinum injection for UK ABWR. This is because the information provided is not specific to the UK ABWR design, nor is it of sufficient detail to undertake a meaningful assessment of the adequacy of the system against the expectations of SAP ECH.3 and other relevant SAPs, amongst others. While I do not expect a detailed design for GDA, I would have expected to see more detail than Hitachi-GE have provided. **This has been captured as part of Assessment Finding, AF-ABWR-RC-13, below.** Despite this, because the information provided on the NMIS for GDA is based on existing BWR technology, this suggests that if a similar system were made specific to the requirements of UK ABWR, it may deliver the safety functions required.

Zinc Injection System

591. The design of the Zinc Injection System (ZNIS) provided by Hitachi-GE during GDA is based on a design installed at a number of operating BWRs which inject DZO (Ref. 31). Hitachi-GE's categorisation and classification of the ZNIS presented in Chapter 23 of the PCSR (Ref. 11), is C3. Considering the claims made in the generic safety case and safety functions provided by the ZNIS, from a chemistry perspective, I am satisfied C3 is an appropriate categorisation and classification.
592. The ZNIS is designed to passively "inject" DZO into the feedwater to achieve the specified concentrations of DZO required in the reactor water. The system features a "zinc supply device", which is essentially a vessel containing a removal basket capable of holding 90 kg of zinc in the form of DZO pellets. The "zinc supply device" is connected to the suction and feed sides of the feedwater pumps located in the main feedwater line. Controlling the flow rate of hot feedwater through the vessel causes the DZO pellets to slowly dissolve over time. In this sense, the concentration of DZO is "controlled" by adjusting the flow rate through the dissolution vessel, which, in turn, controls the rate of DZO pellet dissolution. The required flow rate is manually controlled and is set by obtaining grab samples to determine the concentrations of zinc in the reactor water and feedwater, downstream of the "zinc supply device". In Rev. 3 of Ref. 18, Hitachi-GE performs a calculation for an existing BWR to roughly determine the DZO demand of the plant. This is used to estimate the equivalent feedwater DZO concentration the ZNIS would provide for UK ABWR. This calculation does demonstrate the ZNIS design presented during GDA may be capable of delivering the required control of DZO within the limits and conditions defined in the generic safety case. This is the only information Rev. 3 of Ref. 18 provides to justify ZNIS is adequate for UK ABWR.

Like the NMIS, the corresponding *Topic Report on Zinc Injection* (Ref. 31), which also covers aspects of the ZNIS, actually contains much more information which most closely resembles a justification for the design presented during GDA. In summary, Ref. 31 explains existing BWRs have moved away from "active" DZO injection systems (this is the approach used for PWRs which inject depleted zinc) for several reasons:

- Operational issues, including pipe slugging and valve failures occurred routinely, which led to problems being able to inject DZO and the system achieving the reliability required.
- The passive system is much simpler to maintain, easier to operate and leads to reductions in worker doses.

593. On this basis, whatever the final site-specific design for the UK ABWR ZNIS looks like, I accept that basing it on passive "injection" is a reasonable approach to take. Overall

though, my position for GDA on the assessment of the adequacy of the ZNIS is the same as my position on the other sub-systems which comprise the UK ABWR Chemical Injection System. Ref. 31 states, *“The specific DZO system to be incorporated into the UK ABWR will be detailed during the site specific phase of the project. The following sections provide an overview of the functional and equipment description of a DZO injection system for UK ABWR, based on systems already in use in BWRs”*. This means I cannot yet conclude whether the ZNIS will provide the necessary level of control over DZO injection for UK ABWR. The information provided is not specific to the UK ABWR design, nor is it of sufficient detail to undertake a meaningful assessment of the adequacy of the system against the expectations of SAP ECH.3 and other relevant SAPs. **This has been captured as part of Assessment Finding, AF-ABWR-RC-13, below.**

Oxygen Injection System

594. The OI system maintains the desired concentration of DO in the condensate and feedwater. The system is very simple. It depressurises oxygen from an oxygen gas cylinder and injects it directly into the condensate piping at the LPCP discharge or CF outlet, through a series of pressure and flow control valves.
595. As described in much more detail in Sections 4.3.2.1 of my report, the OI system plays a very important role in mitigating FAC in the CFDW system main piping and controlling corrosion products and their impact on the plant’s radiological source term. Despite the functions the OI system provides and the safety case claims made by Hitachi-GE, Chapter 23 of the PCSR (Ref. 11) identifies the system as being non-safety classified. This does not undermine the generic design, but, from a chemistry perspective, I am not satisfied with this aspect of the UK ABWR generic safety case. I consider the OI system has been under classified by Hitachi-GE. **I consider this to be a Minor Shortfall:**

MS-ABWR-RC-06: The UK ABWR generic safety case does not assign a safety classification to the feedwater oxygen injection system, but makes a number of important claims on the safety functions it provides. These include minimising the plant’s radiological source term and mitigating the risk of flow accelerated corrosion in Condensate and Feedwater system carbon steel main piping. The licensee should review and update the UK ABWR safety case to assign and justify a suitable safety categorisation for the oxygen injection system.

Chemical Injection Capabilities during Commissioning

596. Although Hitachi-GE has left the options of early adoption of HWC, OLNC and DZO injection during commissioning open for the future licensee to decide, the UK ABWR generic design does not currently have the engineered capability to implement them. In Ref. 38 Hitachi-GE therefore provides further information to justify these options are not foreclosed. For platinum and DZO Hitachi-GE states:
- Earlier application of platinum can be achieved relatively easily by making a modification to a drain line position in the CUW system and for even lower temperature platinum application techniques, as part of temporary equipment designed for chemical decontamination.
 - For DZO injection an “active” system can be deployed at the outlet side of the CUW filter demineraliser, again by making a relatively simple modification to a drain line position in the CUW system.
597. On this basis I am satisfied with the justification provided by Hitachi-GE for DZO injection and platinum. I am content the UK ABWR generic design does not foreclose these options for a future licensee.

598. For hydrogen injection the position is slightly more complicated. As described earlier in my assessment, EHWC is not proposed for UK ABWR. **This has been captured as part of Assessment Finding, AF-ABWR-RC-04.** In Ref. 38 Hitachi-GE have also justified the UK ABWR generic design does not foreclose EHWC implementation (and therefore “early” hydrogen injection during commissioning) on the following basis:
- Temporary hydrogen injection equipment can be installed at the outlet side of the CUW system filter demineraliser, again by making a relatively simple modification to a drain line position in the system.
 - The SSCs required to implement EHWC are largely based on the equipment that will already be incorporated into the UK ABWR design to implement HWC.
 - Japanese BWRs that have practised “early” hydrogen injection have done so successfully based on the modifications and equipment described above.
599. I am satisfied with the position for GDA, but I note here, as the generic safety case has left these decisions open for a future licensee to make, further consideration of these aspects will be required as part of **Assessment Finding, AF-ABWR-RC-03.**

Chemical Injection Systems Summary

600. The common theme for all of the Chemical Injection System sub-systems assessed above is that for GDA, Hitachi-GE has presented very high-level information on systems which are not specific to the UK ABWR design. I therefore do not consider what has been provided during GDA is an adequate justification which shows that these have been properly optimised to be able to implement the UK ABWR operating chemistry. Only limited evidence has been provided towards achieving this objective.
601. Throughout GDA, I therefore raised several RQs (RQ-ABWR-1145 and 1463 (Ref. 49)) requesting further information to enable me to undertake a more meaningful assessment of the design of these systems against the expectations of SAP ECH.3 and other relevant SAPs. No such information was provided. What was provided in response to RQ-ABWR-1463 (Ref. 49) is a comparison between the UK ABWR and existing BWR designs, of relevant technical information including: flows, pressures, temperatures, volumes etc. for each of the HOIS, NMIS and ZNIS. This information shows the operating envelope for UK ABWR is very similar to the corresponding operating envelopes for existing BWRs, where the HOIS, NMIS and ZNIS have all been successfully deployed. In the absence of an adequate justification for a UK ABWR specific design for these systems, this provides some confidence that technology exists to be able to control these aspects of the operating chemistry.
602. I consider this is an acceptable position to reach for GDA. However, considering the change proposed for the UK ABWR operating chemistry, providing an adequate justification for the design of these systems is fundamental to be able to show the UK ABWR chemical injection systems are able to control “new” chemistry parameters important to safety, within the limits and conditions (Operating Rules) defined in the safety case. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-13: During GDA, Hitachi-GE changed the operating chemistry proposed for UK ABWR. Considering this change, it is important to develop adequate chemical injection systems to control “new” chemistry parameters important to safety, within the limits and conditions (Operating Rules) defined in the safety case. The licensee shall justify that the specific design of the UK ABWR chemical injection system (including the hydrogen oxygen injection system, noble metal injection system and zinc injection system), enables the UK ABWR operating chemistry to be controlled within the limits and conditions defined in the safety case.

4.3.5.2 Chemical Clean-up Systems

603. In this part of my assessment I consider the systems claimed to perform a chemistry-based safety function relating to the clean-up of chemical impurities and/or the removal of radioactivity from the RCS. This means two principal systems are covered, the CUW system and the CPS.
604. Of particular relevance is *Topic Report on Impurity Ingress* (Ref. 30). This TR provides information on the main sources of impurities for UK ABWR, a more targeted risk assessment of impurity ingress, the measures in place to mitigate impurity ingress events and selected OPEX looking at the levels of impurities typically found in BWRs. Ref. 30 is essentially Hitachi-GE's main source of information to justify the levels of chemical impurities in UK ABWR are reduced SFAIRP. The document is well structured and considers the main sources of impurities I would expect to see. Hitachi-GE has clearly recognised the importance of implementing adequate foreign materials exclusion programmes. I consider this is a matter of normal business for a future licensee to develop such programmes to be able to implement the requirements provided in Ref. 30. Furthermore, Hitachi-GE has also introduced a reasonably practicable design enhancement for UK ABWR, in the form of a Chloride Ingress Protection System (CIPS). The CIPS provides protection during events where significant quantities of seawater may ingress into the RCS if the main condenser suffers tube ruptures. I consider adding the CIPS is a very positive step and have assessed the system in more detail in Section 4.7.5.1 of my report.

Reactor Water Clean-up System

605. An outline of the UK ABWR CUW system is given in Figure 15 in Section 4.3.1.1. As described in Chapter 23 of the PCSR (Ref. 11), the CUW's chemistry related safety functions are to:
- provide continuous clean-up of soluble species and particulate matter present in reactor water; and
 - maintain "low levels" of radioactivity in the reactor water.
606. As well as at-power operations, the CUW system is also used during start-up, shutdown and hot standby. In refuelling mode the CUW system is used in conjunction with the FPC to achieve the required clean-up capacity. Hitachi-GE's categorisation and classification of the CUW system presented in Chapter 23 of the PCSR (Ref. 11), is C3. Considering the claims made in the generic safety case and safety functions provided by the system, from a chemistry perspective, I am satisfied C3 is an appropriate categorisation and classification.
607. The CUW system supports a number of chemistry related claims in Chapter 23 of the PCSR (Ref. 11), including [RC SC1], [RC SC2] and [RC SC7], which are about assuring proper chemistry control. The detailed claims are summarised in Annex 4 of my report. Although not explicitly presented as such, Chapter 23 of the PCSR (Ref. 11) also claims, "*the chloride, sulphate and other ionic impurities are maintained ALARP by the clean-up systems in the Reactor Coolant System, namely the CUW and CPS*". Ref. 18 is Hitachi-GE's principal safety case document to justify the design of the CUW system is adequate to enable these explicit and implicit claims to be met and to demonstrate UK ABWR can be operated within the limit values identified for the relevant chemistry-based ORs for the CUW system (listed in Annex 5 of my report).
608. The design of the CUW system for UK ABWR is based on the J-ABWR reference plant. The part of the system which performs the clean-up function consists of two 50% capacity F/Ds. The type of F/D technology used in the UK ABWR CUW system is based on a pre-coat system, whereby powered ion exchange media is pre-coated onto filter holders. Ref. 18 provides information to justify the appropriateness of this technology. Hitachi-GE argues excluding a deep-bed demineraliser enables the system to be simplified. Ref. 18 also contains a comparison of the clean-up

performance of CUW systems which use different F/D technologies and information which shows the F/D technology selected by Hitachi-GE is commonly used in the CUW systems of many other BWRs. On the basis of this information and relevant independent information available to me (Ref. 44), I am satisfied Hitachi-GE has provided an adequate justification for the use of pre-coat F/Ds in the CUW system for UK ABWR.

609. As well as the F/D technology selected, the design basis flow rate of the CUW system is also a key parameter governing the clean-up performance delivered by the system. The UK ABWR CUW system design basis flow rate is 2%^{***}. Hitachi-GE acknowledges that increasing the flow rate may improve the clean-up performance of the system, but argues a 2% is appropriate for UK ABWR because:
- Based on OPEX and theoretical calculations presented in Ref. 18, there is a significant reduction in reactor water conductivity (and therefore the clean-up performance of the CUW system) moving from 1 to 2% and the increase in clean-up performance past 2% is much more marginal.
 - OPEX shows a 2% CUW system capacity maintains the reactor coolant conductivity at $\leq 10 \mu\text{S/m}$ during at power operations (the expected value for UK ABWR is $\leq 15 \mu\text{S/m}$ (Ref. 18)).
 - A large number of US BWRs and several Japanese BWRs have 1% capacity CUW systems.
 - Other improvements have been made to the ABWR design, including the adoption of separate CFs and CDs in the CPS.
 - Higher flow rates increase equipment and operational costs, volumes of radioactive waste and reduce the thermal efficiency of the plant.
610. On this basis I am satisfied that a flow rate of 2% is adequate for the CUW system.
611. Ref. 18 also provides (limited) information to justify the CUW system is able to achieve the F/D outlet conductivity values ($\leq 10 \mu\text{S/m}$ (at 25°C)) specified by the *Water Quality Specification* (Ref. 20). The evidence is measured plant data from several Japanese BWRs, including data which shows F/D outlet conductivities of $\sim 6 \mu\text{S/m}$ (at 25°C) are achieved. On this basis, for GDA, I am satisfied with this particular aspect of the evidence provided by Hitachi-GE to justify the chemical performance of the CUW system.
612. With respect to radioactivity removal, Ref. 18 itself provides very little evidence to justify the performance of the CUW system. Only one piece of information is provided - a comparison between different BWR plants looking at their ^{60}Co removal efficiencies during shutdown. This is used by Hitachi-GE to support the justification of the adequacy of a 2% flow capacity for the UK ABWR CUW system. Further information regarding the system's ability to remove radioactivity was provided in response to RI-ABWR-0001 (Ref. 66). This information was considered as part of my closure report (Ref. 79) for RI-ABWR-0001. I would have expected Ref. 18 to make better use of this information. However, taking all of this information together, I am content the performance of the CUW system's ability to remove radioactivity has been adequately justified during GDA. This is really a presentational matter, which a future licensee could choose to address in responding to **Assessment Finding, AF-ABWR-RC-14, below**.
613. On the basis of tests performed when commissioning J-ABWR plants in Japan, Hitachi-GE argues that a Decontamination Factor (DF) of 10 for soluble and insoluble species has been "confirmed" for the UK ABWR CUW system. Despite this, Ref. 18 does not provide any evidence to support the arguments made. Furthermore, whilst a

^{***} 2% corresponds to 2% of feedwater flow

DF of 10 does seem achievable for the clean-up technology selected by Hitachi-GE, different chemical species in different physical forms have very specific DFs. There will be a wide range of DFs which the CUW system achieves. Ref. 18 does not provide any information to identify what the limiting species might be, nor does it justify why even for the most limiting species, the CUW system design to give a DF of 10 is adequate. **This has been captured as part of Assessment Finding, AF-ABWR-RC-14, below.**

614. In terms of the CUW system's ability to remove specific chemical impurities, the principal and explicit claim ([RC SC2.2]) made in Chapter 23 of the PCSR (Ref. 11) is its ability to remove silica. Ref. 18 does not provide any specific evidence to support claim [RC SC2.2]. Silica is an important impurity to control because high reactor water silica levels have been implicated in past BWR fuel failures, but also because it is notoriously difficult to remove.
615. By making claim [RC SC2.2], Chapter 23 of the PCSR (Ref. 11) formally acknowledges silica as being an important impurity to control. Ref. 11 also states, "*Silicate chemistry can be very complex and the concentrations can be very high* [in UK ABWR]". The approach taken by Hitachi-GE to claim the CUW system provides "control" over silica levels is to optimise the back-washing frequency of the F/Ds to manage silica breakthrough. This means the CUW system is probably not "controlling" (i.e. removing it) silica in the same way the system functions to remove soluble chemical impurities, such as chloride and sulphate. There is some ambiguity here, which is directly related to the topic discussed immediately below regarding Hitachi-GE's criteria for performing CUW system F/D backwashing and pre-coating. If the limiting factor is performing this operation to "control" reactor water silica levels within "limit" values, it isn't immediately clear why the silica levels which UK ABWR operates to is, "ALARP". **This has been captured as part of Assessment Finding, AF-ABWR-RC-14, below.**
616. In Ref. 18 Hitachi-GE argues there are several criteria which determine when the CUW system F/Ds are backwashed and pre-coat re-applied: high F/D outlet conductivity, differential pressure and/or silica breakthrough. Hitachi-GE argues OPEX from Japanese BWRs shows backwashing and pre-coating of the UK ABWR CUW system F/Ds may only be required every 70 days, based on the first criterion of high F/D outlet conductivity. They therefore claim the CUW system design basis assumption of 40 is conservative. This may be true, but in Ref. 18, Hitachi-GE does not justify whether this is the most limiting of the three criterion. **This has been captured as part of Assessment Finding, AF-ABWR-RC-14, below.**
617. In summary, in terms of my judgement against the relevant SAPs (ECH.3 and ECH.4), there are a number of gaps in Hitachi-GE's justification for the chemical performance of the CUW system:
- Information to support the claimed DFs, in particular relating to filtration performance.
 - Direct evidence to support the claim on the system's ability to remove silica.
 - Evidence wider than just normal operations to substantiate the chemical performance of the system under more limiting conditions for example, start-up and/or shutdown when impurity and radioactivity levels, respectively, will be higher.
 - Information regarding level of margin the two 50% capacity F/Ds design provides, in being able to meet the requirements in the *Water Quality Specification* (Ref. 20).
618. I am therefore not satisfied the expectations of SAPs ECH.3 and ECH.4 have been fully satisfied. Considering the safety case claims made by Hitachi-GE regarding the chemical clean-up performance of the CUW system, and the radioactivity removal

function the system provides to ensure worker doses are reduced SFAIRP, **I consider the above gaps taken together to be an Assessment Finding:**

AF-ABWR-RC-14: The chemical clean-up performance delivered by the reactor water clean-up and fuel pool cooling clean-up systems, is vital for removing radioactivity from the plant during normal operations, to ensure worker doses are reduced SFAIRP. To adequately substantiate the clean-up performance of these systems and address the requirement to demonstrate worker doses are reduced SFAIRP, the licensee shall review and update the UK ABWR safety case to address limitations in the level of detail and justifications provided on these matters during GDA. This shall include all operating modes and cover the removal of soluble and insoluble forms of radioactivity and impurities.

619. Despite raising this Assessment Finding, in the context of a generic safety case, I am satisfied Hitachi-GE has provided the minimum amount of information I expect to see to justify the adequacy of some of the key aspects of the design of the CUW system, from a chemistry perspective. For example, the flow rate and clean-up technology selected.

Condensate Purification System

620. The CPS is part of the CFDW system. One of the main functions the CFDW system fulfils is to provide reactor coolant water which is of sufficient quality; this objective is achieved by the CPS. An outline of the UK ABWR balance of plant, including the CPS, is provided Figure 4 in Section 4.3.1 of my report. The CPS mainly comprises a CF and CD and associated piping, vales etc. Hitachi-GE's categorisation and classification of the CPS presented in Chapter 23 of the PCSR (Ref. 11), is C3. Considering the claims made in the generic safety case and safety functions provided by the system, from a chemistry perspective, I am satisfied C3 is an appropriate categorisation and classification.
621. Unlike the CUW system, the formally numbered claims presented in Chapter 23 of the PCSR (Ref. 11) do not mention the CPS directly. I have not presented the relevant claim here again, but my presumption is that claim [RC SC1.2] also includes the CPS when refereeing to "*the clean-up system*". Although the CPS isn't explicitly mentioned in the formally numbered claims in (Ref. 11), it is clear from the main body of the text that Hitachi-GE recognises the important role the CPS plays. Ref. 11 states:
- *"Impurity control and monitoring in the feedwater is therefore important to maintain the water quality in the reactor water."*
 - *"Impurity removal in the CPS is fundamental in achieving the feedwater quality, and the protection by the Chloride Ingress Protection System (CIPS)."*
 - *"The CD is installed at the condenser outlet and is designed to protect the plant against chloride ingress in the case of a condenser tube leak, maintaining the water quality for a required period, which is specified by assuming one condenser tube leakage."*
622. As explained in Section 4.3.1.2 of my report, UK ABWR is a forward pumped plant. A large number of BWR plants are forward pumped. During GDA I asked (RQ-ABWR-0079 (Ref. 49)) Hitachi-GE to explain the consequences of forward pumping, on the design of the CPS. In summary, forward pumping enables the capacity of the CPS and feedwater heaters to be reduced, which significantly enhances the thermal efficiency of the plant. Based on Hitachi-GE's explanation, it is clear the costs of making any significant alterations to the forward pumped arrangement of the HD system would be grossly disproportionate to any benefits gained, and would likely not be technically feasible given the rated power of the reactor. On this basis, I accept Hitachi-GE's justification for the UK ABWR being forward pumped.

623. *Topic Report on Design Justification in Chemistry Aspect for Primary Water Systems* (Ref. 18) contains Hitachi-GE's justification, from a chemistry perspective, for the design of the UK ABWR CPS. The CPS features three separate CF units arranged in parallel. Ref. 18 states the CF is designed to remove insoluble particulates (mainly iron-based insoluble corrosion products). Immediately downstream of the CFs, condensate is treated by a total of six separate CD vessels, also arranged in parallel. The CDs in the UK ABWR CPS are the deep-bed type and are principally designed to target the removal of soluble ionic impurities. The vessels contain a mixture of strong acid cationic resins and strong base anionic resins. In Ref. 18 Hitachi-GE also claim OPEX shows the CDs have a small capacity to remove insoluble species (DF of two), but re-iterate the purpose of the CDs is not remove insoluble species because this is the role of the CF. The design basis of the CD system as a whole (all six beds) is primarily based on the chemistry parameters measured at the outlet of the CD.
624. In Ref. 18 Hitachi-GE argues separate CFs and CDs is the optimum approach which provides the "best" CPS chemistry performance. Figure 23 in Section 4.3.3.4 of my report plots the average feedwater iron concentrations for BWRs as a function of CPS design. Based on these data, it is clear that plants with separate CFs and deep-bed CDs, like UK ABWR, exhibit the "best" performance. In addition, the general trend in recent years has been for BWRs to move towards this particular design of CPS. This is one of the changes, amongst others, which has led to BWR plants now exhibiting much lower levels of impurities in main feedwater than they did during their early years operation. I therefore consider Hitachi-GE are implementing what appears to be the modern, good practice technology for removing impurities from BWR main feedwater.
625. Ref. 18 also presents information to justify the performance of two particular types of filter technology that could be selected for the UK ABWR CF vessels – pleated and hollow fibre filters. Hitachi-GE presents selected data from Japanese BWRs which utilise both types of filters. Based on the data provided for GDA, it shows that either technology is able to reduce insoluble iron to levels ≤ 1 ppb, as measured at the outlet of the CF for the Japanese BWRs. Hitachi-GE argues that either technology could be selected and that the decision should be taken and justified by the future licensee. Based on the information provided, I am satisfied with this position for GDA and I consider the decision on CF filter technology choice is a matter of normal business.
626. The supporting evidence provided to justify the chemical performance of the CDs is based on a selected range of measured plant data taken from Japanese BWRs; more specifically, conductivity measurements taken at the outlet of the CDs. The evidence shows during at-power operations the CDs for these plants are able to achieve direct conductivity values of $\sim 5.5 \mu\text{S/m}$ (25°C). This is lower than the design basis value of $\leq 10 \mu\text{S/m}$ (25°C) and lower than the expected value of $\leq 6 \mu\text{S/m}$ (25°C) given in the *Water Quality Specification* (Ref. 20). Based on these data Hitachi-GE concludes the clean-up capacity of the CDs in UK ABWR is adequate. On this basis, I accept the UK ABWR CDs should be able to clean-up the feedwater to meet the required specification. However, I do have several observations regarding the nature and scope of the information provided by Hitachi-GE during GDA:
- It isn't possible to determine how many and which type of BWRs these data are for.
 - Only data for at-power operations are provided. Start-up and/or shutdown will probably be much more limiting in terms of the ability of the installed capacity to meet the clean-up requirements.
 - No data are provided to justify the chemical performance of the CDs in removing specific impurities i.e. chloride or sulphate. According to the design basis, these are impurities the CD specifically targets the removal of.
627. Unlike the gaps identified in the generic safety case to justify the chemical clean-up performance of the CUW system, I do not consider the above gaps for the CPS are

significant enough to raise an Assessment Finding. They do not undermine the generic design, however, **I consider they are gaps in the generic safety case which constitute a Minor Shortfall:**

MS-ABWR-RC-06: The UK ABWR generic safety case contains several gaps in the information provided to justify the claimed chemical clean-up performance of the Condensate Purification System. The system provides the primary means for producing main feedwater at the required flow and water quality, to secure continued, normal, at-power reactor operations. The licensee should review and update the UK ABWR safety case to address the gaps identified during GDA.

628. Ref. 18 also presents a justification for the number of CD vessels and the volume of resin they contain. For the number of vessels, Hitachi-GE argues that installing eight or nine CD vessels (as opposed to six which the design already includes), would be difficult to achieve mainly due to space constraints. This aspect of Hitachi-GE's case isn't particularly robust. It doesn't examine the other side of the balance – more vessels would increase the clean-up capacity of the system and the level of margin it provides. Despite this, for GDA I consider it is more important to determine the overall acceptability of the generic design in terms of the clean-up technology selected (i.e. separate CF and deep bed CDs) and the evidence provided to show it will work. I have already accepted Hitachi-GE has satisfied both of these requirements, above.
629. To justify the volume of resin present in each of the six CD vessels, Hitachi-GE stipulates the feedwater quality specified in the generic safety case needs to be met based on several bounding conditions. For example: CD vessels in-service for five calendar years of rated power operation before resin change-out and sea water has ingress into the RCS for [REDACTED] minutes at a volumetric flow rate of 25 m³/h, amongst others. On this basis, I am satisfied Hitachi-GE has provided an adequate justification for the feedwater clean-up capacity which the selected volume of ion exchange resin for the CDs, will provide for UK ABWR.
630. Ref. 18 also provides further information on the maximum condensate flow rates which the entire CF and CD installed capacity is designed to process, and each individual CF and CD vessel. Chapter 23 of the PCSR (Ref. 11) summarises some of this information in order to place limits on the maximum flow rates which each CD and CF vessel can process. Two resultant ORs are identified in Ref. 18 for CF and CD maximum flow rates. These are listed in Annex 5 of my report. From my assessment of the generic safety case, it isn't immediately clear why it is necessary to identify these particular parameters as chemistry-based ORs. **This has been captured as part of Assessment Finding, AF-ABWR-RC02.**

CPS Capability during Commissioning

631. Ref. 38 states that the CUW system is the main system available throughout commissioning^{†††}. The CPS only becomes available once reactor cooling by the CFDW system is required, during the final stages of commissioning. At this later time both systems essentially function as they do during normal operations, hence my assessment of the chemical performance of the CPS presented above remains relevant. Ref. 38 also identifies several more options available to a future licensee to increase the available clean-up capacity during the earlier stages of commissioning:
- installing portable clean-up devices in the reactor cavity, and;
 - flooding the Spent Fuel Pool (SFP) earlier to bring the Fuel Pool Cooling and Clean-up System (FPC) into service.

^{†††} The CUW system becomes available after functional testing is completed during phase one of commissioning.

632. On this basis I am satisfied that the chemistry clean-up systems should not unnecessarily restrict the future plant operator when deciding on the chemistry control options to apply during commissioning.
633. Overall, my view is that Hitachi-GE has provided a justification for the chemical performance of the CPS which is fit-for-purpose for a generic safety case and I am satisfied that the expectations of SAPs ECH. 3 and ECH.4 have been broadly met. I expect a future licensee to give consideration to **Minor Shortfall, MS-ABWR-RC-06**, I have raised above.

Chemical Clean-up Systems Summary

634. As part of the generic safety case, Hitachi-GE has produced documentation solely aimed at justifying the chemical clean-up performance of the CUW system and CPS. This justification is largely based on selected Japanese OPEX. I am satisfied the overall approach adopted by Hitachi-GE is reasonable. However, I have noted some specific gaps and limitations in the information provided to justify the chemical performance of these systems.
635. For the CPS, I am broadly satisfied the nature and scope of these gaps is consistent with a Minor Shortfall, **MS-ABWR-RC-06**, which the licensee should address. However, for the CUW system, considering the claims made by Hitachi-GE on this system and the significant role it plays in removing radioactivity to ensure worker doses are reduced SFAIRP, I consider the gaps are more significant. I have raised an Assessment Finding, **AF-ABWR-RC-14** to address them.
636. Despite raising **AF-ABWR-RC-14** and **MS-ABWR-RC-06**, overall, I am satisfied that for GDA, Hitachi-GE has provided the minimum amount of information necessary to justify the design and performance of the CUW system and CPS, from a chemistry perspective. I am content that the nature of the gaps identified are not significant enough to undermine the UK ABWR generic design and are therefore matters which are suitable for a future licensee to address.

4.3.5.3 Sampling and Monitoring Systems

637. This part of my assessment considers the system(s) Hitachi-GE claim regarding their ability to perform functions to sample and monitor for chemistry parameters. For UK ABWR, the principal system is the Sampling and Monitoring (SAM) system. The SAM system is required to function both during normal operations and during and following fault and/or accident conditions. In this part of my assessment I consider normal operations only; accident sampling is assessed in Section 4.7.6 of my report. In addition, I also give specific consideration to the ability of the generic design to sample for/measure DZO, ECP and platinum in the RCS. These are important changes proposed for UK ABWR, which based on the generic safety case claims made on these parameters; need to be reliably and accurately determined.

Chapter 23 of the PCSR (Ref. 11) presents the following “top claim” and claim for the SAM system:

- “[RC SC13] *The UK ABWR sampling and monitoring systems will ensure that the plant is operated within the limits and conditions*”.
- “[RC SC13.1] *Chemistry and radiochemistry control parameters will be sampled and monitored in accordance with the sampling philosophy*”.

638. As well as these formally identified claims, Chapter 23 of the PCSR (Ref. 11) also presents high-level summary information outlining Hitachi-GE’s justification for the design of the UK ABWR SAM system. This includes a description of some of the changes that have been considered, and made to the design of the UK ABWR SAM

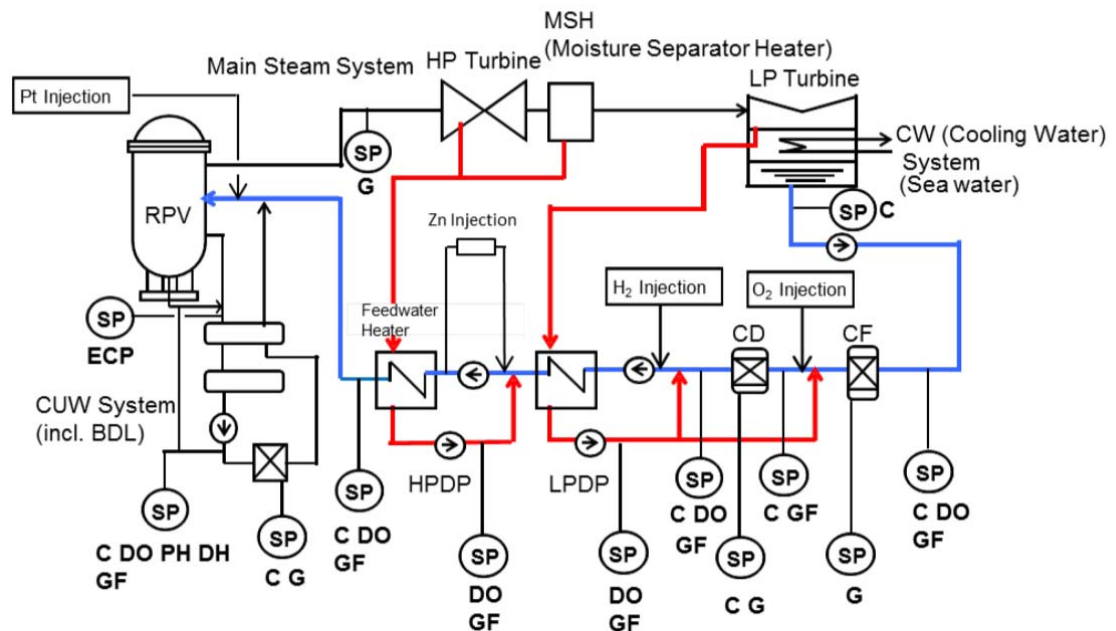
system as a consequence of changing the operating chemistry. Chapter 16 of the PCSR (Ref. 98) also presents a summary of some of the aspects of the SAM system design. Overall, I consider the level and type of information presented in the PCSR on the SAM system is appropriate for a generic safety case. Again, although high-level, I consider the claims made by Hitachi-GE are reasonable.

639. In common with the other UK ABWR systems which deliver chemistry control, *Topic Report on Design Justification in Chemistry Aspect for Primary Water Systems* (Ref. 18) is the document which presents the majority of the information to justify the adequacy of the design of the SAM system, from a chemistry perspective. As well as Ref. 18, the following documents also provide pertinent information:
- *Monitoring Parameter in Design Basis Faults* (Ref. 99);
 - *Monitoring Parameters for Severe Accident Management in UK ABWR* (Ref. 100);
 - *Sampling System – System Design Description* (Ref. 101); and
 - *Topic Report on Sampling and Monitoring Philosophy for Chemical and Radiochemical Parameters* (Ref. 102).
640. During the earlier stages of GDA, based on my assessment (RQ-ABWR-1038, -1198, -1351 and -1352 (Ref. 49) of earlier revisions of Refs. 18 and 99 – 102 it was clear the information provided was insufficient to be able to undertake a meaningful assessment. I therefore raised RO-ABWR-0081 (Ref. 25), to address the gaps in the earlier information presented. Hitachi-GE provided an updated version (Rev. 3) of Ref. 18, including more specific and detailed information on the SAM system for UK ABWR, a number of SAM system Process and Instrumentation Diagrams (P&IDs) (Ref. 102, 104) and an additional report, *Topic Report on Sampling Capabilities of Chemistry-related Parameters in Design Basis Faults* (Ref. 105). I targeted my assessment of these documents on the following aspects of Hitachi-GE's justification for SAM system design:
- the system's ability to enable representative samples to be taken;
 - how the consideration of worker radiological safety has influenced the design;
 - the impact of sampling on the generation of radioactive waste; and
 - how the design enables safe and representative sampling of important chemistry parameters during and post-design basis accidents.

641. I have therefore structured my assessment of the SAM system which follows below, according to these themes. My assessment of the sampling provisions provided in the UK ABWR design during accidents is presented separately, in Section 4.7 of my report. In support of my assessment of the SAM system I also commissioned a TSC to undertake an independent review of relevant good practice for BWR sampling systems. I also requested my TSC to review a selected number of relevant Hitachi-GE's submissions. I have drawn on the outcome of my TSC's work (Ref. 106) in various places throughout my assessment of the SAM System. The most relevant SAPs to undertake my assessment of the SAM system were ECH.3 and ECH.4.

Overview of the SAM System

642. The SAM system is large and relatively complicated, providing sampling and monitoring points across a number of buildings for a range of UK ABWR systems. While not exhaustive, this includes the CFDW, CUW and HD systems. This means that in common with other BWRs, there is no single device or location where sample lines are routed. Instead, a number of local sub-systems are used. Figure 29, below, provides a general overview of some of the identified sampling points for the UK ABWR RCS, which are part of the SAM system.



(SP) : Sampling Point C: Electric Conductivity Meter
 DO: Dissolved Oxygen Meter PH: pH Meter DH: Dissolved Hydrogen Meter
 G: Grab Sampling GF: Grab Sampling and/or Filter Holder
 ECP: Electrochemical Corrosion Potential

Figure 29: Identified sampling points in the UK ABWR “primary circuit” (Ref. 18)

643. The SAM system is designed to be able to undertake both online and grab sampling. This capability is supported further by other measurements taken in, and by other systems, during normal operations and/or fault and/or accident conditions.
644. For GDA, Hitachi-GE’s approach to justifying the design of the UK ABWR SAM was to start with the J-ABWR reference plant. A gap analysis of selected parts of the J-ABWR SAM system was undertaken, to consider whether the change in the operating chemistry proposed for UK ABWR has any significant impact. In this regard, Ref. 18 an impact assessment, outline sampling schedule, and a comparison of the SAM system against what Hitachi-GE considers to be relevant good practice for BWR sampling system designs. The main outcome of this work is three specific modifications for the UK ABWR generic design. The modifications include adding:
- hydrogen meter(s) to the CUW system;
 - ECP sensor(s) to the CUW system; and
 - hydrogen meter(s) to the CFDW system.
645. Based on these modifications, it is clear Hitachi-GE are giving a wider consideration to how the UK ABWR generic design needs to be optimised, to enable the “new” operating chemistry to be implemented.
646. Ref. 18 also provides a breakdown of the main chemistry parameters sampled and monitored for UK ABWR, along with a justification for the methods selected. For the majority of the parameters, I am satisfied with the choice between on-line and grab sampling. The one minor exception is Hitachi-GE’s justification for chloride and sulphate. Based on independent information provided to me by TSC (Ref. 106), common practice for Nordic and US BWRs is to conduct on-line measurements for these parameters. Hitachi-GE argues this is not necessary for UK ABWR because conductivity is continuously measured. Considering the approaches adopted

elsewhere, I do not necessarily agree with this assertion. However, for GDA I am satisfied, because this does not undermine the generic design as the option to implement on-line measurements for chloride and sulphate is not foreclosed. I expect a future licensee to consider such requirements as part of responding to **Assessment Finding AF-ABWR-RC-16, below.**

647. During GDA, Hitachi-GE did not provide information covering all aspects of the design of the UK ABWR SAM system. More detailed information, based on J-ABWR, for a small number of the most safety significant SSCs connected to the SAM system, was provided. I focussed my GDA assessment on the CUW system. This is because this is the system which provides the functionality to sample reactor water. As I expect, it is clear from the generic safety case that implementing adequate control over the chemistry of the reactor water is fundamental to the demonstrating that the operating chemistry is being controlled within the limits and conditions defined in the generic safety case.

Representative Sampling

648. The ability to ensure a sample is suitably representative requires consideration of multiple factors. In general, Hitachi-GE has considered the main ones I expect to see. For many of these, I am content with Hitachi-GE proposals, including:
- The general rule is that sampling lines should be as short as possible, to minimise the potential for “line effects”. Hitachi-GE recognises this important requirement. The CUW system sample line lengths are between 70 and 100 metres^{†††}, which appears consistent with the lengths of similar sample lines for other BWRs.
 - For materials selection, Hitachi-GE’s general approach is to use SS to minimise the impact of corrosion; either Type 304 in lower temperature regions or Type 316L (NG) in higher temperature parts, to minimise the impact of corrosion product dissolution from the sample line. Titanium is also used where the corrosion product burden is higher and/or where the concentration of non-radioactive cobalt is measured. Cobalt-free alloys are selected for sampling line valve seats.
 - There are several aspects associated with the adequate selection of sample point source locations. I have assessed the more general aspects of Hitachi-GE’s chosen sampling source locations and I consider they appear to be broadly consistent with sources of good practice available to me (Ref. 44) and also with the findings of my TSC (Ref. 106).
649. However, for a number of other points, I am not satisfied with what is proposed. These include:
- In terms of sample flow velocities, SAM system sample line piping has typical linear flow velocities of 0.5 m/s. Hitachi-GE has calculated Reynolds numbers >8000 based on these conditions. They argue this means the sample lines experience turbulent flow, meaning deposition is limited. This does not appear to be consistent with the sample line linear flow velocities (1.3 – 1.8 m/s) presented in other sources of good practice (Ref. 44) and those suggested by my TSC (Ref. 106), which are required to avoid deposition.
 - In terms of the more detailed geometrical considerations which need to be taken into account when selecting and designing sampling source points, Hitachi-GE states sampling source nozzles should be mounted on the pipe side surface of horizontal and straight sections of process piping. They also prescribe other requirements associated with pipe bends and elbow sections.

^{†††} Sample lines are routinely flushed during normal operations.

The practice of preferring to locate sample source nozzles in horizontal piping is inconsistent with other sources of good practice (Ref. 44) and the independent view of my TSC (Ref. 106), who recommend a vertical pipeline (with a preference for upwards flow).

- Owing to the low concentrations being sampled, and risks associated with nozzle breakage, Hitachi-GE argues isokinetic sampling is not necessary. Whilst I agree the latter point is an important consideration, I am not aware this has been reported in BWRs. Instead of performing isokinetic sampling, to sample particulates, the UK ABWR design uses a small sample nozzle insertion at the source point. This is essentially trying to achieve an appropriate alternative to isokinetic sampling, which does show Hitachi-GE recognises there are additional challenges when sampling for particulates. This approach is based on Japanese OPEX. Alternative approaches which may achieve similar ends appear to be available. This is essentially a matter associated with the specific design of sample nozzles.

Worker Conventional and Radiological Safety

650. Ref. 18 states that some of the specific hazards considered in the design of the UK ABWR SAM system are the nature and quantity of radioactivity, and high temperature and pressure. I agree these are very important considerations for maintaining operator safety during sampling.
651. With respect to radioactivity, Hitachi-GE's justification is focussed on the physical locations of sample line end points i.e. the rooms where they are located and the facilities provided. This includes the arrangements for: Heating Ventilation and Air Conditioning (HVAC), sampling sinks, filter holder shielding and radiation alarms. Based on the information presented, these are all standard approaches I would expect to see in the design of any LWR sampling system.
652. Still on the topic of radioactivity, Ref. 18 also provides (more limited) information about the design of the sampling lines themselves and other features required for sampling. This includes:
- A conclusion that delay coils are not necessary. Hitachi-GE argues they are not required because the radionuclides which have the largest contribution to radiation doses during at-power normal operations are ^{16}N and ^{15}C ; both of which have very short radioactive half-lives (7 seconds and 2.5 seconds, respectively), compared to the relatively long transit time (approximately 200 s) of fluid in the sampling lines. The absence of delay coils is consistent with the practice of US BWRs and is also consistent with my TSC's findings (Ref. 106).
 - Radiation shielding is provided where sampling is undertaken using filter holders, but radiation doses are still expected to be high. As Hitachi-GE also recognises, there are other techniques available to achieve the same measurements. For example, when measuring the concentrations of corrosion products. These techniques may result in reduced worker doses. I expect a future licensee to consider such requirements as part of responding to **Assessment Finding AF-ABWR-RC-16, below.**
653. With respect to high energy (temperature and pressure) sampling lines, evidence is presented to show the balance of competing factors in optimising the location of sample coolers has been considered. High temperature piping is provided with coolers to reduce effluent below 40 °C and the location of pressure reducing equipment has been considered both with the requirement to maintain analysis equipment within calibrated limits and to ensure a safe environment to undertaken local grab sampling.
654. The information provided on the SAM system during GDA to justify its adequacy to handle the levels of radioactivity both expected to be, and permitted to be in the plant

by the generic safety case during normal operations, was less forthcoming. For example, there was little information to link the SAM system design to the radiological source term, shielding requirements and an explanation of the measures in place to prevent an operator being exposed to elevated radiation doses during the process of taking a sample (this is different to local radiation alarms). I therefore raised RQ-ABWR-1512 (Ref. 49) seeking the necessary clarification.

655. In response, Hitachi-GE confirmed the SAM system is available during all anticipated transients, including fuel failures. The case for the system's adequacy to operate during these periods is made by using the DB source term values (i.e. the expected worst case) and performing estimated worker dose calculations. These calculations show the estimated worker doses are below the relevant Basic Safety Objective (BSO). I did note some inconsistencies in the calculations and with, for example, the Generic Technical Specification (Ref. 90) limits. Hitachi-GE explained these differences in the context of conservatism, and importantly clarified that the SAM system can operate up to the relevant Technical Specification limit(s) on radioactivity (Ref. 92). On this basis and considering the estimates indicate worker doses would be below the BSO, I am satisfied with Hitachi-GE's response for this GDA stage.
656. Despite this conclusion, in this regard, there are several inconsistencies presented in the generic safety case. These relate to Hitachi-GE's arguments regarding the adequacy of the measures to prevent workers receiving elevated radiation doses during sampling activities, and also in support of Hitachi-GE's assertions that the SAM system remains available for sampling during transients:
- The same calculation performed for OG system grab sampling indicates worker doses are also expected to be below the BSO, but in the event the relevant Technical Specification limit value is reached, OG system grab sampling is no longer permitted. The same requirement does not exist for sampling reactor water in the CUW system.
 - Ref. 18 identifies the requirement to isolate the containment under certain fault and/or accident conditions. This means the SAM sampling lines which penetrate the containment would also be isolated. This leads to uncertainty about the availability of the SAM system under these scenarios. Furthermore;
 - *Basis of Safety Case on Containment Isolation Systems* (Ref. 107) identifies the requirement to isolate the SAM system sampling lines on detection of: level, temperature and/or pressures being outside of defined limits. The same requirement is not identified for radioactivity.
657. This means the SAM system generic design does not provide any direct engineered controls which prevent the operator sampling reactor water in scenarios where radioactivity levels become elevated. The justification provided by Hitachi-GE during GDA for this aspect of the generic design isn't robust. If the system does remain available (which is unclear at present) for sampling during accident and/or fault conditions (considered in further detail in Section 4.7.6 of my report), it is not clear whether adequate measures are in place to prevent operators receiving high doses during sampling activities. **This has been captured as part of Assessment Finding, AF-ABWR-RC-16, below.**

Radioactive Waste Generation

658. Ref. 18 identifies three main types of sampling: continuous monitoring with process line recycling, continuous monitoring and no process line recycling, and local grab sampling. The latter two will generate radioactive waste as the effluent is not returned to the plant. My assessment has therefore focussed on these to determine whether the design minimises the volume of radioactive waste generated.

659. Hitachi-GE's preferred approach is to recycle sample effluent directly back to the process where the operating conditions (pressure, temperature and chemical conditions) allow. This is positive and the approach I expect to see as part of any wider justification that radioactive waste generated by sampling activities has been minimised. Where effluent generated by continuous monitoring cannot be directly recycled back to the process, it is transferred to the Low Chemical Impurities Waste System (LCW). For local grab sampling, effluent generated by laboratory activities is treated using the High Chemical Impurities Waste System (HCW) before recycling for re-use in the RCS.
660. Ref. 18 presents information on the volume of radioactive waste effluent generated by sampling activities. Based on the information provided during GDA, it was clear that effluents generated from sampling make a relatively small contribution to the overall waste volumes processed by the LCW. For the CUW system in particular, Hitachi-GE explained there are other operational practices that are implemented to reduce radioactive waste. This includes limiting the flow rate, or even stopping flow altogether. There is clearly a balance to be struck between minimising waste and worker doses, and also maintaining sample representivity. During GD, there was some uncertainty regarding a number of aspects associated with this practice. Despite these uncertainties, Hitachi-GE argue the UK ABWR generic design has sufficient flexibility to enable a future operator to choose to recycle effluent from the CUW system. No further information was provided to support this assertion though. Based on Nordic BWR experience, my TSC (Ref. 106) identified the importance of maintaining a constant volumetric flow rate when performing filter-holder sampling (for corrosion products), to ensure representative sampling. I expect a future licensee to consider such practices provide the necessary justifications as the design of the UK ABWR SAM system is developed further. **This has been captured as part of Assessment Finding, AF-ABWR-RC-16, below.**
661. For local grab sampling, Ref. 18 was not particularly clear about the impact effluent generated by these activities, has on the HCW. In response to RQ-ABWR-1511 (Ref. 49) Hitachi-GE clarified the impact was very low, only 0.2 m³ per day of effluent is generated by this route and sent to the HCW for processing. Hitachi-GE assert that since the HCW is specifically designed to process identified input streams, there is no impact on the HCW from the discharge of high chemical impurities effluent from the "hot laboratory". On this basis, in this regard, I am satisfied with the information provided by Hitachi-GE during GDA.

Sampling for DZO

662. Of all the reactor water chemistry parameters, DZO can be one of the more challenging to representatively sample. Given the requirement to sample for DZO is new for ABWRs, I specifically assessed Hitachi-GE's justification for why the UK ABWR design will enable DZO to be representatively sample. Specifically for DZO sampling, Hitachi-GE argues:
- Reactor water DZO concentrations can be measured by an integrated sampling method (filter holder). This is already included in the J-ABWR SAM system design. The method is successfully used by US BWRs which inject DZO.
 - Sampling source points for DZO are situated downstream of the CUW system heat exchangers, therefore minimising the potential for DZO interactions with sample line piping. Hitachi-GE also states the sample line design has a cooler fitted as close to the isolation valve point as practically possible, to minimise any line effects. These measures follow good practice guidance.
663. Considering these points, and that my TSC's independent review of BWR sampling systems (Ref. 106) and other OPEX (Ref. 44) supports Hitachi-GE's assertions, I am satisfied with the information provided on DZO sampling during GDA.

ECP Measurements

664. Hitachi-GE's generic safety case for UK ABWR is that, as well as calculating ECP by radiolysis modelling, it will also be directly measured on the plant using specific ECP sensors to provide long-term verification of the effectiveness of OLNLC, and to provide validation of the ECP modelling. This is particularly important for UK ABWR given no other ABWR has operated with this chemistry.
665. Unfortunately, for some regions of BWR plants where SCC mitigation is required, it may not be possible to install ECP sensors, owing to geometric constraints and the effects of radiation degrading the sensors over time. To address this, many operating BWRs have a system called the Mitigation Monitoring System (MMS). The MMS is an external reactor water sample system connected either to the CUW system or directly off the recirculation line. It is predominately used to monitor ECP, but some existing BWRs also use the MMS to monitor the distribution and deposition of platinum. The system contains ECP sensors exposed to reactor water by a once through flow path with the outlet routed back to the reactor. Hitachi-GE argues (RQ-ABWR-1447 (Ref. 49)) that installing an MMS is not suitable for UK ABWR because of changes to the CUW pump location means they cannot be used to provide motive flow (as in existing plants) and OPEX has shown that the measurements taken by the MMS is often not representative or reliable. Based on independent information provided to me by my TSCs (Ref. 43), Hitachi-GE's case appears to be robust. On this basis, I agree an MMS does not appear to be necessary or practicable. However, this does not alleviate the need to find other suitable and sufficient means of achieving in-process monitoring of ECP and monitoring platinum deposition and distribution.
666. The main justification for the number and location of ECP monitoring points for UK ABWR is presented in Ref. 18. Five potential locations are identified, based on existing plant experience. These include both in-core and external pipework locations, as well as considering modified flanges or Local Power Range Monitor (LPRM) assemblies. These are presented in Figure 30.

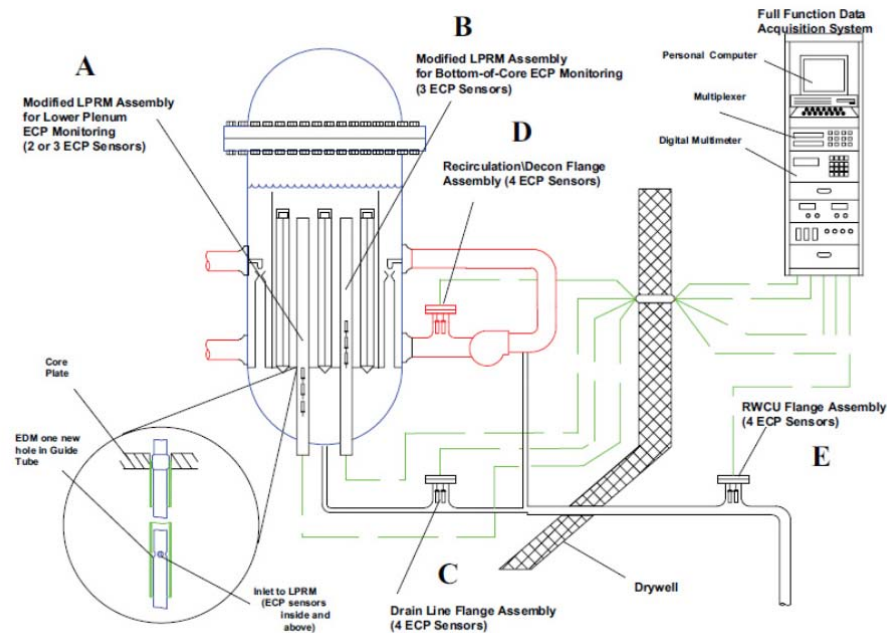


Figure 30: Typical ECP sensor locations for BWRs which Hitachi-GE regard as relevant good practice (Ref. 45)

667. Ref. 18 acknowledges in-core monitoring of ECP is the best indicator of SCC mitigation in the core region and that "location D" is not relevant to the UK ABWR design because it doesn't have external re-circulation piping. For the UK ABWR generic design Hitachi-GE has only selected one location (location E, in the CUW system outside of the RCCV, upstream of the re-generative heat exchanger) to directly measure ECP. This location has been selected on the basis that in terms of the hydrogen to oxidant molar ratio, it represents the most conservative location compared to the lower plenum (bottom of the RPV). In Ref. 18 Hitachi-GE also acknowledge it would be desirable to monitor ECP at other locations in the RCS, but do not specify doing so.
668. I agree with the necessity of monitoring ECP in the CUW system, given the claim made that the operating chemistry mitigates SCC in this system (see Section 4.3.2.1). However, I do not agree that this location alone provides suitable and sufficient ECP monitoring locations, because:
- Direct in-core ECP monitoring has been excluded, but Hitachi-GE claims the operating chemistry mitigates SCC in the lower plenum (RPV bottom head). The generic design does not include ECP monitoring in the lower plenum.
 - The number and location of ECP monitoring points is insufficient to be able to conduct the benchmarking tests that will be required to determine the ECP response and therefore feedwater hydrogen requirements of the plant.
 - The number and location of ECP monitoring points is insufficient to be able to validate any radiolysis and ECP models.
 - It would be difficult to use the CUW system ECP data on its own to provide evidence to demonstrate the plant is operating within the limits and conditions defined in the safety case.
669. In parts of RQ-ABWR-1496 (Ref. 49) I therefore asked Hitachi-GE about the extent of any modifications that may need to be made to the generic design to be able to implement ECP monitoring at the other locations identified (except location D). The response indicates that existing BWRs have implemented such measurements without extensive modifications being necessary, and that the technology exists to be able to do similar in UK ABWR. Therefore, whilst I am not satisfied the generic design includes suitable and sufficient ECP monitoring locations, and I am not content that the expectations of SAP ECH.3 have been met, it is clear that such options have not been foreclosed. Given the overall importance of adequate ECP monitoring, especially during the first few cycles, **I consider this to be an Assessment Finding:**

AF-ABWR-RC-15: Considering the UK ABWR generic safety case claims, and the importance of being able to determine the UK ABWR operating chemistry's effectiveness at mitigating materials degradation risks, the licensee shall justify that suitable and sufficient locations have been included in the UK ABWR design to adequately monitor for:

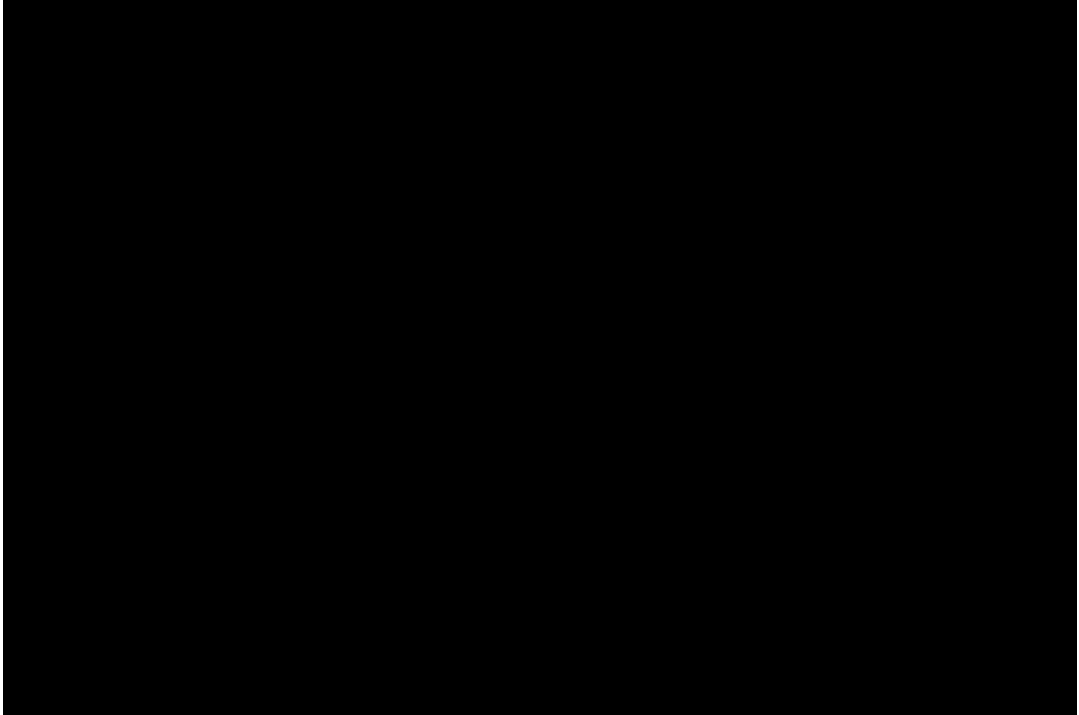
- Electrochemical Corrosion Potential; and
- The deposition and distribution of platinum.

The monitoring locations selected shall consider the requirement to undertake feedwater hydrogen benchmarking tests throughout the operating life of the plant. This shall include the need to validate the radiolysis and electrochemical corrosion potential models used to demonstrate the plant is operating within the limits and conditions defined in the UK ABWR safety case, throughout its operating lifetime.

Monitoring Platinum Deposition and Distribution

670. While ECP monitoring provides vital information on the effectiveness of measures to mitigate SCC, it provides no direct information regarding the behaviour of the platinum

injected into the plant. One of the consequences of the generic design not incorporating an MMS is that it is not available to monitor the deposition or distribution of platinum. The approach for UK ABWR is that sampling of SSCs (either by scraping or other methods) at various locations throughout the RCS will be done periodically throughout an outage. Ref. 32 considers how to determine viable sampling locations and identifies eight “recommended” locations where platinum deposition should be monitored. These are shown in Figure 31.



671. Each “recommended” location is either sampled directly or the platinum loading at some is determined using an estimating technique, based on measured data for the platinum deposited at two adjacent locations where scraping or sampling by other methods has been performed. Hitachi-GE’s proposals to monitor platinum deposition and distribution are not new and have been implemented before for existing BWRs (Ref. 43). This is because, for BWR plants which have an MMS, experience has shown that monitoring the behaviour of platinum using the MMS is unreliable.
672. One of the consequences of Hitachi-GE’s proposed method is that the deposition and distribution behaviour of platinum can only be determined during the outage. Given feedwater hydrogen is identified as an OR and ECP sensors will provide an on-line means for measuring the ECP response, I consider this is an acceptable approach for noble metals.
673. Ref. 32 also considers the first four outages and specifies which location should be sampled during each outage. Most are recommended to be sampled only after the first outage, apart from locations six, seven and eight, which should be sampled during each of the four outages. This appears to be because platinum deposition at these locations can be more easily monitored by removing either an LPRM probe, ECP sensor or channel fastener. The reason why only the four outages are considered isn’t stated, nor is any justification provided for why sampling the majority of locations only after the first outage will be adequate. Furthermore, given the specific claim that the operating chemistry will mitigate SCC in the bottom head of the RPV and the high flows expected owing to the reactor internal pumps, noble metal behaviour is currently uncertain. From Ref. 32 it is unclear how this has been taken into account by Hitachi-GE. The RPV bottom head is identified as a “recommended” location, but the report

states platinum deposition in this part of the plant will be estimated (despite Figure 31 indicating scraping will be undertaken).

674. Most significantly, none of the “recommended” locations are actually included as part of the generic safety case. There are currently no safety case requirements to undertake noble metal deposition and distribution monitoring for UK ABWR. This has been left for the future licensee to decide. Whilst I agree with the philosophy of noble metal monitoring suggested, I am not convinced the generic safety case is sufficient. I judge the expectations of SAP ECH.3 are not satisfied. I consider the process of developing and implementing the necessary arrangements to monitor platinum deposition in UK ABWR during outages is a matter of normal business for a future licensee. However, for the reasons stated above, I consider an adequate justification for the locations and frequency of platinum deposition monitoring, has not been provided. **This has been captured as part of Assessment Finding, AF-ABWR-RC-15.**

SAM System Capability during Commissioning

675. For commissioning, Hitachi-GE states the SAM system may not be available during the earlier phases. In these circumstances, they argue drain lines in the CUW system are suitable “surrogate” sampling locations to enable all relevant chemistry parameters for reactor water quality to be controlled. Hitachi-GE’s argument is based on a comparison of conductivity and aggressive anion measurements taken at a “surrogate” location and one of the main sampling points available in the CUW system when the SAM system is in service (Ref. 18).
676. The evidence provided is very limited and does not give a wider consideration to the appropriateness of the design of these “surrogate” locations to sample other key chemistry parameters, including DZO and particulate corrosion products. Hitachi-GE’s overall conclusion is that the UK ABWR design does not foreclose the option for a future licensee to implement a range of different commissioning chemistries. I consider a future licensee needs to give further consideration to this aspect of the overall design of the sampling provisions provided for UK ABWR. **This has been captured as part of Assessment Finding, AF-ABWR-RC-16, below.**

SAM System Overall Summary

677. Overall, I am content Hitachi-GE has provided sufficient information for my assessment during GDA. Hitachi-GE’s approach has been to base the design of the UK ABWR SAM system on the J-ABWR design, then to undertake a gap analysis and review of the system against Hitachi-GE’s view of relevant good practice for BWR sampling systems. From this I conclude:
- The SAM system appears consistent with the majority of the relevant good practice on BWR sampling systems collated independently by my TSC and information available to me from other sources.
 - Positively, there is some evidence to demonstrate Hitachi-GE has begun to consider changes necessary to optimise the sampling and monitoring systems based on the “new” operating chemistry. I specifically note the important changes to include feedwater hydrogen and ECP measurement.
678. Despite this conclusion, during my assessment of the SAM system, I have identified important gaps either in the design, or the justification provided for it. Further work and important decisions will be required of a future licensee. Some of the more significant matters arising from my assessment above include, the availability and capability of the SAM system across the full range of its claimed operability, and the measures in place to ensure worker doses during sampling are reduced SFAIRP. Despite these matters arising, based on the information available to me during GDA, I am satisfied the gaps should not undermine the generic design. Considering the cumulative

significance of all of the matters arising from my assessment though, and the “new” operating chemistry proposed for UK ABWR, **I consider this to be an Assessment Finding:**

AF-ABWR-RC-16: There were limitations in the information provided during GDA and gaps identified by ONR’s chemistry assessment in both the design, and justification, of the UK ABWR sampling and monitoring system. To address these gaps, the licensee shall provide an adequate justification to demonstrate the UK ABWR sampling and monitoring system will enable the “new” proposed operating chemistry to be controlled within the limits and conditions defined in the safety case. This should include all modes of operation and severe accident conditions, and provide a robust demonstration that relevant risks are reduced SFAIRP.

679. In terms of my overall judgement against the relevant SAPs, in the context of GDA, because there appears to be no fundamental changes required for the SAM system which may undermine the generic design, I am broadly satisfied the expectations of SAPs ECH. 3 and ECH. 4 have been met. As the project progresses though, a future licensee will need to continue to ensure the design of the SAM system continues to fulfil the expectations of SAPs ECH. 3 and ECH.4.

4.3.5.4 Chemistry Control Systems Summary

680. In broad terms, the approach taken by Hitachi-GE to justify the adequacy of the UK ABWR chemistry control systems for the RCS was to consider the existing J-ABWR reference plant systems and identify if additional measures are required. However, in general, the level of information available at the generic stage of the design was insufficient to undertake a complete assessment of all relevant aspects, particularly for the chemical injection systems, and SAM system. Despite this, what I am able to conclude for GDA is that:
- The impact of the proposed change to the UK ABWR operating chemistry on existing plant systems is generally small, although there are some exceptions. These have been captured in my Assessment Findings, **AF-ABWR-RC-13 to -16** and one Minor Shortfall, **MS-ABWR-RC-06**, raised during my assessment of this topic.
 - Hitachi-GE have started to consider how best to adapt the J-ABWR reference plant design to be able to adequately control any additional requirements implementing the “new” UK ABWR operating chemistry brings, including both chemical injection, and monitoring requirements; and
 - At this generic stage of the design, I do not see any reason why a future licensee would not be able to exercise suitable and sufficient control over the operating chemistry, subject to adequately resolving the Assessment Findings, and continued development of the UK ABWR design and safety case.

4.3.6 Radiolysis Gases Generated During Normal Operations

681. The formation of radiolysis gases (hydrogen and oxygen) during BWR operations cannot be eliminated, nor avoided. A 900MWe BWR typically produces around 190 m³ of radiolysis gases for each hour of operation in a stoichiometric mixture, where two thirds of the mixture is hydrogen and the remaining third is oxygen (Ref. 44).
682. As already described in Section 4.2, because of their high distribution coefficients, radiolysis gases are transported as a mixture with the saturated steam out of the RPV. For BWRs, this typically results in concentrations of radiolysis gases in the main steam of about 20 ppm oxygen and 2.5 ppm hydrogen, at pressures greater than 10 bar and at the temperatures of saturated steam (Ref. 44). These gases must be safely managed and removed by the design. This function is principally performed by the UK

ABWR OG system, which recombines the hydrogen with oxygen to form water. My assessment of the OG system is presented in Section 4.3.6 of my report.

683. For a large dynamic system like a BWR, there are a number of scenarios where the radiolysis gases may behave differently than the steam. When the steam condenses this can result in the accumulation of radiolysis gases, leading to potentially explosive mixtures in plant piping or vessels. Scenarios which may lead to the accumulation of radiolysis gases include:
- Changing pipework/component geometries (i.e. horizontal, vertical and/or sloping) and the changes in flow conditions this brings.
 - the amount of heat loss (thus causing steam condensation) from pipework/components to the surrounding environment.
 - the degree to which condensed water is able to escape (i.e. the amount of drainage from the pipework).
684. Radiolysis gases are therefore a particular hazard within the RCS, and connected systems of a BWR. Strictly, the hazard they propose is not caused directly by the operating chemistry, but the hazard can be influenced by it.
685. For operational BWRs, historically there have been a number of hydrogen combustion events as a result of the accumulation of radiolysis gases. Considering the significance of the hazard, during GDA I raised a joint RO with ONR's internal hazards inspector, RO-ABWR-0044 (Ref. 25). The scope of my chemistry consideration of radiolysis gases was confined to the "process", defined in Ref. 25 as, "*the structures, systems and components which provide the primary containment and which when taken together comprise the nuclear reactor and associated systems*". During GDA, I therefore sampled a number of aspects of Hitachi-GE's for radiolysis gases to determine whether Hitachi-GE had:
- Identified all potential "vulnerable" areas in the UK ABWR design which may be susceptible to the formation of flammable atmospheres, either directly or following the accumulation of radiolysis gases.
 - Identified and implemented all reasonably practicable measures to address any vulnerable areas in the UK ABWR design.
 - the approach to describing and documenting, in the UK ABWR generic safety case, the risks and hazards presented by radiolysis gases under normal operations.
686. It is important to note that the scope of my assessment in this section considers the hazard and risk posed by radiolysis gases generated during normal operations, only. Managing hydrogen generated under fault or accident conditions is assessed separately in Section 4.7 of my report.
687. The starting point for my assessment is to consider relevant standards and guidance. For the topic of radiolysis gases some of the most important aspects of RGP is the guidance contained in the Dangerous Substances and Explosive Atmospheres Regulations (DSEAR) Approved Code of Practice (ACoP) (Ref. 108), and also the expectations of SAP EKP.1 - inherent safety. In applying SAP EKP.1 in this context the goal is clear; achieving an inherently safe design would mean eliminating the possibility of radiolysis gas accumulation in UK ABWR.

4.3.6.1 Hitachi-GE's Approach to Safely Managing Radiolysis Gases

688. *Topic Report on Safe Management of Radiolytic Gases Generated Under Normal Operations* (Ref. 109) is the principal submission in Hitachi-GE's document hierarchy for radiolysis gases. It presents the main safety claims, arguments and evidence, and it describes Hitachi-GE's methodology for assessing the potential for radiolysis gas

- accumulation in the UK ABWR design. Ref. 109 also summarises the overall outcome of applying the methodology to UK ABWR.
689. Ref. 109 is underpinned by, *Supporting Information for the Topic Report on Safe Management of Radiolytic Gases Generated Under Normal Operations* (Ref. 110). This report provides the comprehensive technical information to demonstrate how the methodology described in Ref. 109 has been applied to the UK ABWR design.
690. The starting point in Hitachi-GE's radiolysis gas accumulation methodology is to base the approach on Rev. 3 of a set of guidelines entitled, *Guideline on Radiolysis Gases (Hydrogen/Oxygen) Pipe Damage Prevention of BWR Piping* (Ref. 111). Ref. 112 states the guidelines were first developed in 2007 by the "Review Committee on Non-condensable Gases", set up by an organisation in Japan known as the Thermal and Nuclear Power Engineering Society. The committee then fell under the auspices of the Japan Nuclear Technology Institute (JANTI), now known as the Japan Nuclear Safety Institute (JANSI). However, Rev. 3 of the guidelines is still commonly referred to as the "JANTI Guidelines". Hitachi-GE's UK ABWR generic safety case adopts this terminology.
691. Ref. 112 contains the English translation of the JANTI Guidelines. The JANTI Guidelines are based on engineering experience, analysis and experiments. As well as presenting the guidelines themselves, Ref. 113 also documents the outcome of a benchmarking exercise conducted by Hitachi-GE to compare the JANTI Guidelines with equivalent guidance from the US and Germany. Hitachi-GE asserts the outcome of this exercise confirms that the JANTI Guidelines provide an appropriate and acceptable method to minimise the risk caused by radiolysis gases during normal operations.
692. Ref. 109 states that for the assessment of the UK ABWR radiolysis gas accumulation hazard, the JANTI Guidelines are used in conjunction with DSEAR to demonstrate how the hazard is addressed, using a hierarchical approach. Furthermore, in Ref.109, Hitachi-GE clarify that the emphasis of the JANTI Guidelines is to prevent explosions (by designing plant to prevent accumulation of explosive concentrations of radiolytic gases), or to limit their effects to the plant of origin. Hitachi-GE also make further links to the UK regulatory context, drawing specifically on some of the requirements of DSEAR and introducing specific sections of the DSEAR ACoP as relevant good practice. Ref. 109 states, "...In particular, it recommends maintaining hydrogen concentrations to below 25% LFL where practicable. This is considered UK RGP." On this basis, it is clear Hitachi-GE's guidelines have captured key aspects of what is regarded as relevant good practice in the UK.
693. Hitachi-GE has also expanded the scope of their methodology to include non-steam systems. The JANTI Guidelines specifically exclude them. Hitachi-GE defines steam and non-steam systems in Ref. 109. They also state that in their view, assessing non-steam systems is only necessary when there is a significant amount of radiation in the presence of water or resin, resulting in a higher risk of hydrogen accumulation. This means that UK ABWR non-steam systems contained in the Reactor Building (R/B) and Turbine Building (T/B) are not assessed in detail by Hitachi-GE, whereas radioactive waste systems located in the Radioactive Waste Building (RW/B) are. I am satisfied this is a proportionate and reasonable approach to take.
694. Figure 32 provides an overview of the main steps involved in Hitachi-GE's radiolysis gas assessment methodology, and how it relates to relevant requirements of the DSEAR regulations (Ref. 113). Hitachi-GE say the process and Ref. 109 are concerned with identifying where radiolysis gases could accumulate within UK ABWR systems (and are referred to as "the vulnerable areas"), and therefore the potential for deflagrations and detonations of radiolysis gases, which could threaten the delivery of fundamental safety functions.

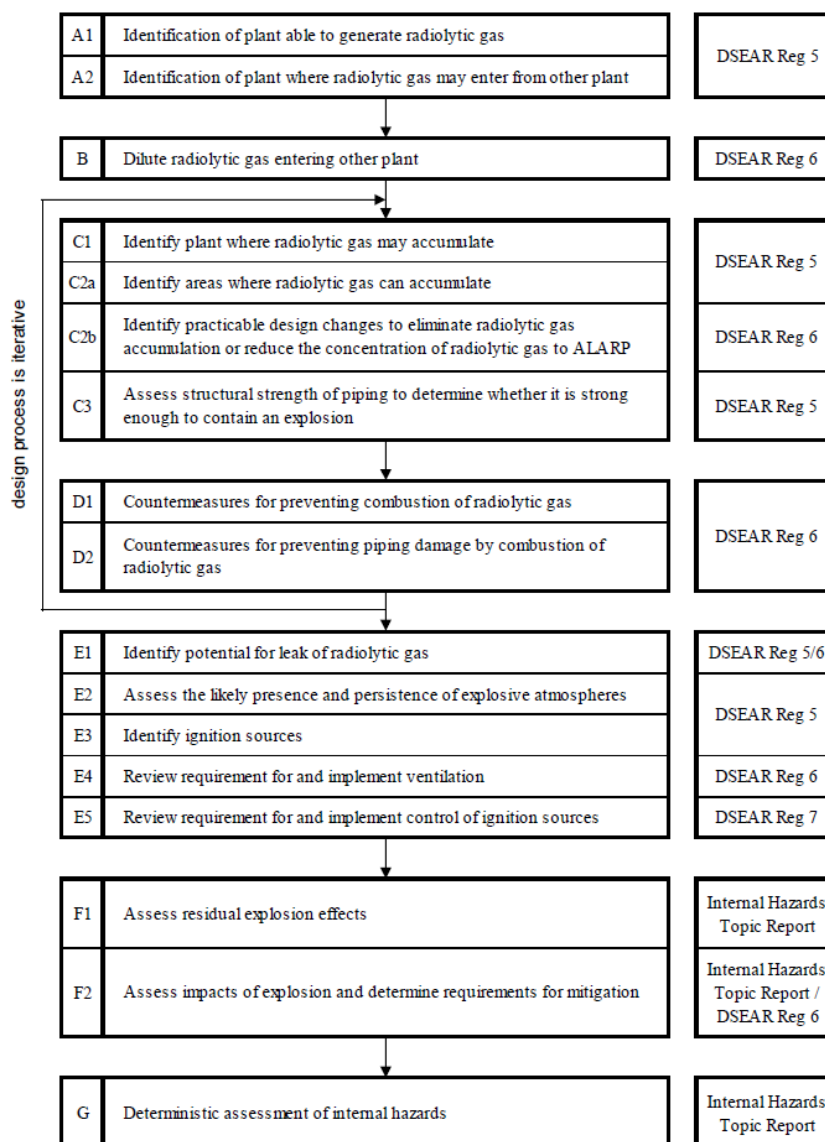


Figure 32: Hitachi-GE's assessment process for radiolytic gases as described in Ref. 109.

695. During GDA, my assessment of Hitachi-GE's radiolysis gas accumulation methodology considered Steps A1, A2, B, C1, C2a and C2b. The remainder of Hitachi-GE's process (Steps 3C – G) is largely concerned with aspects outside the scope of my chemistry assessment; they are considered in Ref. 114. Examining Figure 32 in more detail, it is clear that the guidance contained within the DSEAR ACoP has informed its development. For example:

- Step A of the process uses fractional LFLs (25%) as the basis for the threshold radiolysis gas concentrations for hazard identification. It does not use the LFLs themselves.
- In Step C2a, Hitachi-GE recognises that the JANTI Guidelines are only capable of indicating whether a particular scenario would lead to a concentration of radiolysis gases which exceeds the LFL. To address this, where the accumulation hazard cannot be eliminated by design alone, Hitachi-GE's method includes an extra step to perform additional calculations to confirm the concentration is below 25% of the LFL, in these scenarios.

696. The above points are particularly encouraging and once again demonstrate Hitachi-GE has clearly recognised applicable relevant good practice in the UK regulatory context.

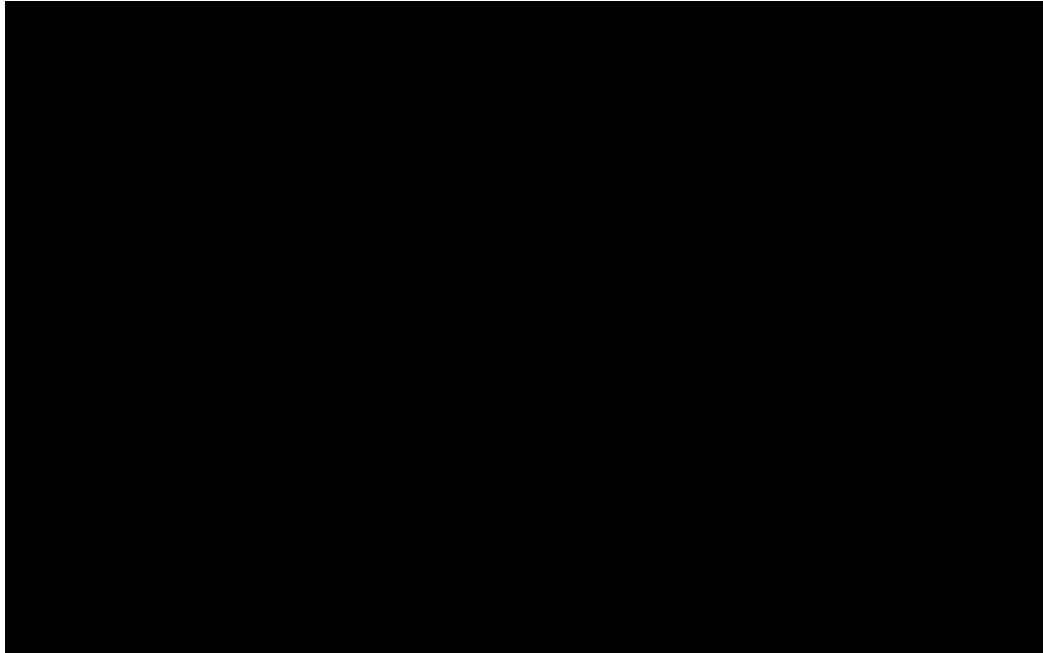
697. As well as at-power operations, Hitachi-GE's methodology also considers the potential for radiolysis gas generation and/or accumulation during various stages in start-up, shutdown and reduced power operations, for systems contained in the RB and TB. They treat the RW/B differently because the systems and processes in this building operate on a batch-wise basis, and therefore do not follow the same operating logic. For start-up and shutdown operations, Hitachi-GE presents various arguments why the risk of accumulation of radiolysis gases can be considered negligible. There are several common themes which underpin all of Hitachi-GE's arguments on this topic:
- For large parts of the start-up and shutdown sequences radiolysis gas generation is negligible since the core is not critical.
 - Various operations only occur for a short period of time (between four and 10 hours depending on the operation).
 - For some stages in the shutdown sequence, radiolysis gas is generated at a low rate since reactor thermal power is very low (<2%).
 - During the shutdown sequence residual steam and gases are vented from reactor systems.
698. As a set of high-level principles, I broadly agree with the above assertions. However, I have sample assessed the detail of Hitachi-GE's approach to radiolysis gas accumulation risk assessment during other (i.e. not at-power) modes of reactor operations, below.
699. In summary, based on my general overview assessment of the main steps in Hitachi-GE's radiolysis gas risk assessment methodology, in the context of a generic safety case, I am satisfied it appears fit-for-purpose. The process should be able to identify all potential ("vulnerable") areas in the UK ABWR design (within the "process") which may be susceptible to the formation of flammable atmospheres, as a result of radiolysis gas accumulation. Hitachi-GE's methodology is based on established practice applied to Japanese BWR plants; it has been benchmarked against other similar international approaches, and captures specific UK expectations. Importantly, it also addresses the specific, generic design of the UK ABWR, and expands the scope to areas not normally considered. The process drives the consideration of removing the hazard first, consistent with the hierarchy of controls and SAP EKP.1. I have assessed specific aspects of Hitachi-GE's application of the methodology to the UK ABWR generic design, in detail, below.

4.3.6.2 Operating Chemistry Impact on Radiolysis Gases

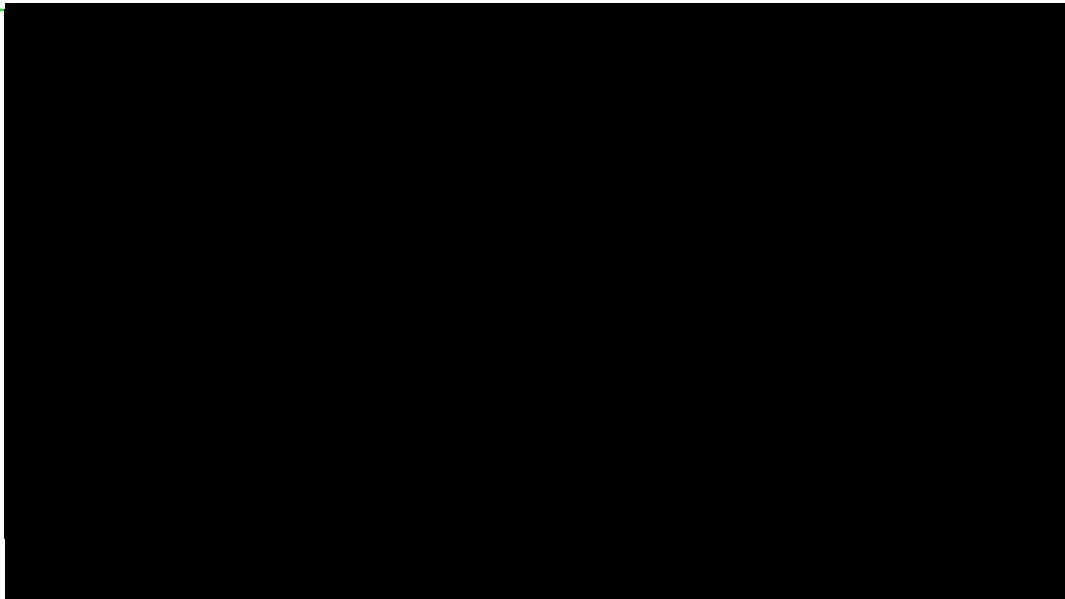
700. Ref. 109 presents a summary of Hitachi-GE's relevant safety claims and maps them to underpinning arguments and evidence. Chapter 23 of the PCSR (Ref. 11) also makes the link between chemistry control and its impact on the generation and control of flammable gases. In this regard, the "top claim" and claim made by Hitachi-GE in Ref. 109 are:
- "[RC SC4]: *The UK ABWR pipework and system design will mitigate the build-up of hydrogen concentration and mitigate flammability risk and radioactive release from reactor coolant system*".
 - "[RC SC4.1]: *Pipework and system design will minimise the build-up of hazardous concentrations of hydrogen*".
701. As I would expect, and as explained earlier, because the hazard of radiolysis gas accumulation is not directly caused by the operating chemistry, in this regard, Chapter 23 of the PCSR (Ref. 11) does not make any explicit claims on it. From a chemistry perspective, I am satisfied with claims [RC SC4] and [RC SC4.1] and I note there is a general alignment between the PCSR and the underpinning TR (Ref. 110). However, what is clear from the wording of [RC SC4] and [RC SC4.1] is that the claims in Chapter 23 of PCSR (Ref. 11) are relatively high-level. Nevertheless, I consider this is

largely restricted to being a presentational matter, and in the context of a generic safety case, I am satisfied with the approach taken.

702. In other parts of the UK ABWR generic safety case, Hitachi-GE make important assumptions (or more accurately, implicit claims) about the impact of HWC on radiolysis gases. These assumptions cover: the generation rates of radiolysis gases, the corresponding hydrogen concentrations in main steam, and the level of margin this corresponds to within the UK ABWR design basis. Hitachi-GE claim that as a result of HWC, the concentration of hydrogen in UK ABWR main steam will be "*less than 2 ppm*". The principal source of supporting evidence used by Hitachi-GE, are data presented in Figure 33, below. These are data from the J-ABWR plant KK-7, measured when hydrogen benchmarking tests (mini-tests) were conducted at full rated power, to evaluate the likely responsiveness of the ABWR design to HWC.



703. These measured data are also supported by calculated data presented by Hitachi-GE, which shows the average hydrogen concentration in UK ABWR main steam is expected to be 1.85 ppm, with a range between 1.73 and 1.97 ppm. These calculated data are some of the outputs from Hitachi-GE's work on radiolysis modelling. I have previously assessed the adequacy of this work in Section 4.3.2.1 of my report, and I have therefore already accepted their validity. Taken together, these data do show that, as claimed by Hitachi-GE, one of the impacts of the operating (i.e. injecting hydrogen) is to lower the concentrations of hydrogen in main steam, relative to NWC plants.
704. During GDA, I also asked (RQ-ABWR-0783 (Ref. 49)) Hitachi-GE about the UK ABWR design basis for radiolysis gas production rates. The standard design value used by Hitachi-GE for total radiolysis gas (hydrogen and oxygen) production rate is \blacksquare $\text{Nm}^3/\text{hr}/\text{MWt}$. This is shown as the red line in Figure 34 below. To justify this rate, Figure 34 shows the relationship between reactor thermal power output and radiolysis gas production rate for ten BWRs and one ABWR. The average value for all of these plants is $0.064 \text{ Nm}^3/\text{hr}/\text{MWt}$. Hitachi-GE says this ensures a 36% safety margin is included within the UK ABWR design basis, with respect to radiolysis gas production rates.



705. Applying this rate to UK ABWR conditions gives a main steam hydrogen concentration of 3 ppm (RQ-ABWR-1321 (Ref. 49)). As the main steam hydrogen concentration used by Hitachi-GE in their radiolysis gas accumulation analysis is 2 ppm, it is important to note here that the design value of \blacksquare $\text{Nm}^3/\text{hr}/\text{MWt}$ has clearly not been used. In this context, I therefore do not accept Hitachi-GE's claim that the UK ABWR design provides a 36% safety margin. Nevertheless, there are several mitigating factors, which I believe for at-power operations, provides some confidence that a value of 2 ppm provides some conservatism:

- Data in Figure 34 are taken from a relatively large number (eleven) of BWR plants. The average radiolysis gas production rate is $0.064 \text{ Nm}^3/\text{hr}/\text{MWt}$.
- Measured plant data in Figure 33 shows implementing HWC for the ABWR design is expected to lower (relative to NWC conditions) the main steam hydrogen concentration. The same trend is observed for other operational BWRs.
- Hitachi-GE's radiolysis calculations using the core designs provided during GDA, which are based on the feedwater hydrogen ORs defined in the generic safety case, show main steam hydrogen concentrations are expected to vary between 1.73 and 1.97 ppm for UK ABWR, when implementing HWC.

706. For these reasons, for at-power operations, I am content that not applying the \blacksquare $\text{Nm}^3/\text{hr}/\text{MWt}$ design value is acceptable for GDA. I have assessed the appropriateness of using 2 ppm as the assumed main steam hydrogen concentration during other phases of operation, in Section 4.3.6.4 of my assessment, below. That section provides my overall conclusion on the level of margin in Hitachi-GE's radiolysis gas assessment.

4.3.6.3 Application of Hitachi-GE's Radiolysis Gas Methodology – At-power Conditions

707. Ref. 109 summarises the outcome of Hitachi-GE's application of the process depicted in Figure 32. The outcome is presented for the systems considered by Hitachi-GE in the R/B, T/B and RW/B. The results show that the UK ABWR generic design has either been modified throughout the GDA process, and/or show how the J-ABWR reference plant has been modified (and those changes implemented in the UK ABWR design), following application of their radiolysis gas methodology.

708. In Ref. 109, Hitachi-GE also presents arguments and links to evidence, to justify why they believe the risk from radiolysis gases generated under normal operations has

been reduced SFAIRP. Ref. 109 states that for the “vulnerable areas” identified by Step C2a of their methodology, the majority have been completely eliminated by design changes to pipework configurations, to prevent the accumulation of radiolysis gases. The report also identifies some areas which Hitachi-GE term “*residual vulnerable areas*”. These are areas where the hazard of radiolysis gas accumulation cannot be completely eliminated by design alone. The identified “*residual vulnerable areas*” include parts of the spent sludge system, air off-take systems and extraction steam system. A summary of the outcome is shown in Table 9.

Building	Potential Vulnerable Areas	Residual Vulnerable Areas
Reactor Building	32	0
Turbine Building	113	46
Radioactive Waste Building	17	3

Table 9: Summary of the outcome of applying Step C of Hitachi-GE’s radiolysis gas assessment methodology

709. Ref. 109 also makes it clear that, during detailed design and plant operations, any changes made to piping configurations, should apply the same methodology to ensure risks remain reduced SFAIRP. It is clear, even at this summary level, that the consideration of radiolysis gas accumulation and applying the defined methodology has resulted in a number of design improvements for the UK ABWR. At this level, I am therefore content with Hitachi-GE’s application of their methodology, and the resulting conclusions and reasonably practicable modifications made to the UK ABWR generic design. In support of my overarching conclusion, I have assessed some more detailed aspects of the work produced by Hitachi-GE during GDA, below.
710. Table 9 demonstrates Hitachi-GE claim there are no “*residual vulnerable areas*” identified in the Reactor Building. Considering the majority of the UK ABWR steam systems are located here, this is a particularly significant and encouraging conclusion. However, Ref. 109 also identifies two areas which require further consideration and assessment during the detailed design phase. These are:
- the Reactor Pressure Vessel Head Spray (RPVHS) line in the CUW system; and
 - the Reactor Vessel Instrument condensing chambers in the nuclear boiler System.
711. OPEX shows that one of the most serious hydrogen combustion events to have occurred in BWRs happened at Brunsbüttel, Germany, during 2001. The design of this BWR included a RPVHS line. The root cause of the hydrogen combustion at Brunsbüttel was identified as radiolysis gas accumulation inside the pipework. Given the seriousness of this event, and the fact that the UK ABWR design also includes a similar RPVHS line, I focussed part of my assessment of Hitachi-GE’s case for radiolysis gas management in the Reactor Building on this particular aspect of the generic design.
712. Although the inclusion of a RPVHS line is common in operational BWRs, many plants do not use the line and some have even blanked it off. During GDA I therefore requested (RQ-ABWR-0786 (Ref. 49) Hitachi-GE to provide a suitable justification for this line. Ref. 116 presents Hitachi-GE’s ALARP assessment for the RPVHS line. It concludes the “ALARP option” is for the RPVHS line to remain in the generic design. Ref. 116 explains the functions provided by the line include cooling of the RPV flange

during outages and purging of the RPV. Importantly, Ref. 116 clarifies that the line does not support any safety functions. Hitachi-GE states the RPVHS line is included for operational reasons – to shorten the outage to enable the 30 day outage period assumed during GDA to be met. It is argued that removing the RPVHS line would therefore result in a significant cumulative commercial impact over the lifetime of the plant, for the future operator. It is also noted that the UK ABWR generic design has already been modified to remove the possibility of a similar event occurring, by including a bypass (vent) line. Hitachi-GE's overall conclusion, therefore, is that the risk of radiolysis gas accumulation and subsequent combustion in the RPVHS line is very low, and that removing of the line is judged not to be reasonably practicable. Conversely, Ref. 116 also states that:

- While a similar bypass line has also been retro-fitted to the J-ABWR, there are an appreciable number of Japanese BWR plants which do not use the RPVHS at all.
- There are other alternative technical methods to speed-up RPV head flange cooling, if a future operator chose to do so.
- While the process of RPV head flange cooling may be important to maintaining adequate levels of safety, the speed at which cooling is achieved appears to be less safety significant, if at all.

713. For GDA, I accept Hitachi-GE's rationale to include the RPVHS line in the design. It avoids foreclosing options a future licensee may wish to implement based on operational decisions. I also accept that in retaining the RPVHS line in the generic design, modifying it to install a vent (bypass) line should prevent the accumulation of radiolysis gases. However, applying SAP EKP.1, relevant good practice in this context may be to eliminate the RPVHS line from the UK ABWR design. To be able to conclude this matter, important operational decisions first need to be made by a future licensee. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-17: The UK ABWR generic design includes a reactor pressure vessel head spray line. The line has been modified to include a vent to prevent the accumulation and potential combustion of hydrogen and/or oxygen. The reactor pressure vessel head spray line performs no claimed safety functions and its use is not universal across other Boiling Water Reactor plants. The licensee shall review their detailed design, outage plans and procedures to decide whether retaining the reactor pressure vessel head spray line within the UK ABWR design reduces relevant risks SFAIRP.

714. As identified by Hitachi-GE, the second area requiring further consideration at the detailed design stage is the RVI condensing chamber. During GDA, the assumption is that a change is made to this component, to implement a "flat top" configuration for the condensing chambers. This is an important assumption because it means the accumulation of radiolysis gases can be eliminated. If the assumption were not made, these chambers would therefore become a "*residual vulnerable area*". The assumption and its consequences are clear in Ref. 109. For GDA I am therefore content this is an acceptable position. However, this does mean that based on the generic safety case, I would expect a future licensee to implement this modification. I am satisfied this should form part of normal business for the future licensee, as part of the detailed design process.

715. Table 9 also shows Hitachi-GE identified the largest number (113) of "vulnerable areas" in the Turbine Building. Of the 113, a small number (14) were discounted because they were "*fully ventilated*" (i.e. accumulation cannot occur due to the piping geometry and resultant flows), or calculations showed it was not possible to reach 25% of the LFL. A further 53 "vulnerable areas" were eliminated by implementing reasonably practicable changes to the UK ABWR generic design. The changes are

mainly concerned with altering piping layouts and/or geometries. For example, Ref. 109 provides examples of numerous locations in UK ABWR steam systems containing branched pipes that have been modified to a horizontal orientation, to eliminate high-points in the design, where it may have been possible for radiolysis gases to accumulate up to and beyond 25% of the LFL. In this regard, a selection of the reasonably practicable changes include:

- sections of MSR pipework in the MS system;
- some sections of pipework and some instrument lines (level) for the Steam Jet Air Ejector (SJAE) in the Air Off-take (AO) system;
- level instrument lines in various parts of the HD system;
- several instrument lines and some sections of pipework in the Extraction Steam (ES) system;
- several sections of pipework in several locations in the Auxiliary Steam system, and;
- OG system header piping, air purge piping.

716. Where applicable, Ref. 109 also clearly states where a modification made for UK ABWR has resulted from the application of Hitachi-GE's radiolysis gas accumulation methodology produced specifically for GDA. It also explains that applying the JANTI Guidelines alone would not have led to identifying that location as a "vulnerable area", because JANTI only assesses accumulations up to the LFL, as opposed to fractional LFLs. As well as examples where pipework has been changed to a horizontal orientation, Ref. 109 gives several examples where changes have been made to pipework lengths, by considering the location of valves in the design. The most prominent example I noted during my GDA assessment is some of the piping connected to the SJAE condenser.
717. During my assessment, I did note some occasions where some aspects of the full methodology outlined above was not presented in the submissions. In particular, Ref. 109 identifies several occasions where "*independent ventilation analysis*" (performed using CF/D), to determine the concentrations and/or accumulation rates of radiolysis gases, was required. During GDA, I asked (RQ-ABWR-1323 (Ref. 49)) Hitachi-GE a number of questions about the approach taken to perform these analyses. Their response was satisfactory, and therefore, I did not consider it proportionate or necessary to sample the detailed CF/D analysis performed, particularly as the overall objective of Hitachi-GE's methodology is to eliminate "vulnerable areas" by design, as opposed to justifying safety based on complex analysis.
718. For the instrument lines in the AO and ES systems identified as "*residual vulnerable areas*", during my assessment I also noted Ref. 109 provides additional evidence which demonstrates further consideration was given to other options, by Hitachi-GE. This is clearly very positive and consistent with the requirements of providing a robust demonstration that risks are reduced SFAIRP. I also noted that part of Hitachi-GE's justification for retaining these lines relies on the outcome of a piping strength evaluation. However, this is outside of the scope of my chemistry assessment.
719. Overall, I judge Hitachi-GE's claim that 53 locations in the TB previously identified as "vulnerable areas", with respect to radiolysis gas accumulation, have been eliminated from the design, as being justified. From my assessment of Refs. 109 and 110 for the Turbine Building examples, it is clear that Hitachi-GE have provided the vast majority of the evidence required to support the claims made. In the context of radiolysis gases generated during normal operations, there is clear and compelling evidence to demonstrate the UK ABWR generic design meets the goal of SAP EKP.1, and implements relevant parts of guidance contained in the DSEAR ACoP. I am satisfied relevant good practice has been implemented by Hitachi-GE. However, the important exclusion from my conclusion here are large parts of the OG system, which are not assessed by Hitachi-GE in Refs. 109 and 110. This is because the JANTI guidelines

purposefully exclude this system. This exclusion is not consistent with UK regulatory expectations. My assessment of the chemistry aspects of the OG system is presented in Section 4.6.2 of my report.

4.3.6.4 Application of Hitachi-GE's Radiolysis Gas Methodology - Different Operating Modes

720. In Refs. 109 and 110 Hitachi-GE also consider the potential for radiolysis gas accumulation during other modes of operation. For example, calculations are performed for shutdown to assess the potential for hydrogen accumulation in Safety Relief Valve (SRV) branched piping. The calculations show the concentration of hydrogen remains below 25% of the LFL. Furthermore, there are examples where "vulnerable areas" have been identified based on radiolysis gas concentrations estimated during start-up and shutdown, as opposed to full power operations. For example, inlet piping on steam safety valves for feedwater heaters in the HD system. In the broadest sense, I am therefore satisfied that Hitachi-GE's methodology gives consideration to other reactor operating modes, however, there are some specific aspects of their approach I assessed in more detail during GDA, which are presented below.
721. I have already concluded that I am satisfied with Hitachi-GE's general method for assessing the potential for radiolysis gas accumulation, i.e. the determination of whether "adequate ventilation" exists in branched pipework. The method is based on assessing the relationship between the steam velocity (and therefore reactor pressure, temperature and power) and the flow this induces in the pipework, to determine the extent of the "ventilation". During start-up and shutdown operations, and/or reduced power operations, the steam velocity will be lower, meaning some branched pipework may no longer be "fully ventilated". In addition, when purposefully injecting hydrogen to implement HWC, there may be times when the steam velocity may not be the limiting factor. During GDA I therefore asked (RQ-ABWR-1321 (Ref. 49)) Hitachi-GE a number of queries aimed at exploring the sensitivity of these aspects of Hitachi-GE's methodology.
722. Hitachi-GE provided reasoned arguments why their approach is suitably conservative. The arguments refer to the steam flow rates when HWC is started (at 5-10% thermal power), and point to the fact that the amount of hydrogen purposefully added is always lower than the amount generated by radiolysis, because the volume of hydrogen gas entering the main steam is directly proportional to the rate of steam production. These statements are logical, but I note Hitachi-GE's methods depend on the assumed starting concentration of hydrogen in the main steam. During GDA, I therefore specifically asked Hitachi-GE to provide evidence, in the form of measured plant data, for main steam hydrogen concentrations for BWR plants which inject hydrogen during start-up. The only set of measured plant data available for an ABWR running HWC is from KK-7 (shown previously in Figure 33). As previously described, these data were obtained at full reactor power. Strictly speaking, hydrogen purposefully injected into the reactor should not be classed as a radiolysis gas because it hasn't been generated as a result of water radiolysis, although it can accumulate in exactly the same way. I would therefore consider this to be an arbitrary distinction. During GDA, Hitachi-GE was unable to provide the evidence requested. However, they did acknowledge that it may be possible for the main steam hydrogen concentrations for UK ABWR to rise above (the assumed) 2 ppm, during start-up operations. Hitachi-GE argues that if this were the case, the impact of this deviation on their radiolysis gas accumulation assessment is not significant. Based on the evidence provided during GDA, I do not consider it is credible for Hitachi-GE to make such a claim.
723. The radiolysis gas accumulation assessment presented during GDA (based on an assumed main steam hydrogen concentration of 2 ppm) would be expected to be representative of how UK ABWR might behave in the majority of circumstances. As

previously explained in the earlier part of my assessment of radiolysis gases covering the impact of the operating chemistry, this conclusion is most applicable to at-power conditions. However, as acknowledged by Hitachi-GE themselves in some of their responses during GDA, for other operating modes and conditions, and when implementing HWC, the radiolysis gas accumulation analysis may not be entirely representative, and therefore may not be a suitably conservative basis on which to base all aspects of the generic safety case on. This is because:

- Hitachi-GE's radiolysis gas accumulation methodology assumes a main steam hydrogen concentration of 2 ppm, as opposed to 3 ppm. 3 ppm corresponds to the radiolysis gas generation rate of 0.1 Nm³/h/MW which Hitachi-GE says is the UK ABWR design basis value.
- If the UK ABWR hydrogen gas concentration in the main steam did exceed 2 ppm during start-up operations, the plant would be operating outside of the assumptions made, and therefore safety analysis performed in, the generic safety case. During GDA Hitachi-GE has not been able to provide evidence that during all anticipated (and permitted) operating modes, the hydrogen concentration in UK ABWR main steam will not exceed 2 ppm. In fact, Hitachi-GE acknowledges that on occasions, it may exceed 2 ppm.
- The information presented during GDA is based on a specific core design, which may differ to the core design selected by the future plant operator.

724. My overall conclusion, therefore, is that I am not convinced the radiolysis gas accumulation assessment provided by Hitachi-GE during GDA has been performed on an appropriately conservative basis, to account for the impact of operating with HWC during all operating modes. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-18: During GDA, ONR's chemistry assessment has established that some of Hitachi-GE's assumptions regarding the accumulation of radiolysis gases (hydrogen and oxygen), may not be suitably conservative for all operating modes, and also to account for the implementation of Hydrogen Water Chemistry. Based on these conclusions, and considering their detailed design and reactor operating strategy (including potential prolonged reduced power operations), the licensee shall review and update the UK ABWR safety case, to provide safety analysis for radiolysis gas accumulation which is performed on a demonstrably, suitably conservative basis, and show that relevant risks are reduced SFAIRP.

725. During GDA, similar, but related queries also arose regarding other operations which have the potential to undermine some of the assumptions and analysis performed for radiolysis gas accumulation in the generic safety case. These have also been captured as part of **Assessment Finding, AF-ABWR-RC-18, above**. They are namely:

- I also asked Hitachi-GE whether operating BWRs exhibit transients (i.e. any peaks or spikes) in radiolysis gas production. Hitachi-GE argues radiolysis gas production rate is directionally proportional to core power, as is the rate of steam production. Scientifically this is a reasonable argument to make. However, in Section 4.6.2 of my report I assess OG system. As part of that assessment I have noted OPEX which shows some operational BWRs have experienced hydrogen transients (in the OG system). In addition, I also note that Hitachi-GE's argument is based on the concentration of radiolysis gases, and not the absolute values of hydrogen a spike may result in. Hitachi-GE's case for GDA therefore excludes the possibility for spikes in radiolysis gas production. For a generic safety case, I accept this is a reasonable position, but I would expect a future licensee to consider this aspect of the radiolysis gas accumulation assessment in more detail and provide further evidence to justify this is an appropriate assumption to make.

- In terms of the impact of reduced power operations, Hitachi-GE state that based on their “time to accumulation calculations” conducted for start-up and shutdown operations (i.e. reduced power), the shortest time they consider a steam system may reach 25% of the LFL in, is 49 days. Based on the assumptions used in the generic safety case Hitachi-GE say no requirement has been identified where 49 days operating at reduced power would be necessary. Whilst the analysis performed for GDA appears to be valid based on this assumption, it does highlight the more important point that depending on decisions a future operator may take in terms of maximum permitted timeframes for reduced power operations, further consideration may need to be given to other risk reduction measures. This may include additional engineered protection and/or operational practices. This could be important, but requires key operational decisions to be made by a future licensee, which is why it has **been captured as part of Assessment Finding, AF-ABWR-RC-18, above**

4.3.6.5 Additional Risk Reduction Measures

726. To manage the hazard of radiolysis gas accumulation, some BWRs have adopted other measures, including temperature surveillance at certain locations and the ability to sample and measure for radiolysis gases. During GDA I asked (RQ-ABWR-1322 and -1323 (Ref. 49)) Hitachi-GE to justify why these measures are not appropriate, or “ALARP” to implement, for UK ABWR. Hitachi-GE say that temperature monitoring to detect the possible accumulation of radiolysis gases is not required for UK ABWR because:
- In most cases applying their methodology has resulted in the elimination of the accumulation hazard, by design.
 - Some of the “*residual vulnerable areas*” which remain are non-steam systems in the RB, meaning the accumulation mechanism is different and therefore temperature surveillance is not an appropriate measure to implement.
 - A large majority of the “*residual vulnerable areas*” are instrument pipes, eliminating the possibility for accumulation is not technically feasible as the measurement could not be taken reliably. The necessity for the instrumentation has been justified elsewhere.
727. For the same reasons given above, Hitachi-GE also says the ability to sample and analyse for radiolysis gas accumulation is not required. On this basis I accept Hitachi-GE’s position for GDA – the goal of SAP EKP.1 is either met or there are valid technical reasons why such measurements should not be implemented.
728. As identified in Step C2b of Hitachi-GE’s methodology, venting is also a relevant risk reduction measure that could be implemented. During GDA, I asked (RQ-ABWR-1322 (Ref. 49)) Hitachi-GE whether, applying their methodology, there were any circumstances where venting had been identified as a specific requirement. Hitachi-GE’s response indicated that aside from normal purging activities conducted during shutdown, there are none. This is because the accumulation hazard has been eliminated by design changes. Based on my assessment above, I am satisfied with Hitachi-GE’s response - the goal of SAP EKP.1 has been met.
729. As Hitachi-GE also recognise in their GDA submissions, it is clear that the conclusions regarding the necessity for additional risk reduction measures, are only valid based on the assumptions made in the generic safety case. Whilst I am content with the position for GDA, once the future licensee conducts further work to assess what impact reduced power operations and purposefully injecting hydrogen (in particular during start-up), may have on the analysis conducted in the generic safety case, I would expect the technical feasibility and reasonably practicability of some or all of these additional risk reduction measures, to be re-visited and justified on an appropriate basis.

4.3.6.6 Radiolysis Gases Generated During Normal Operations Summary

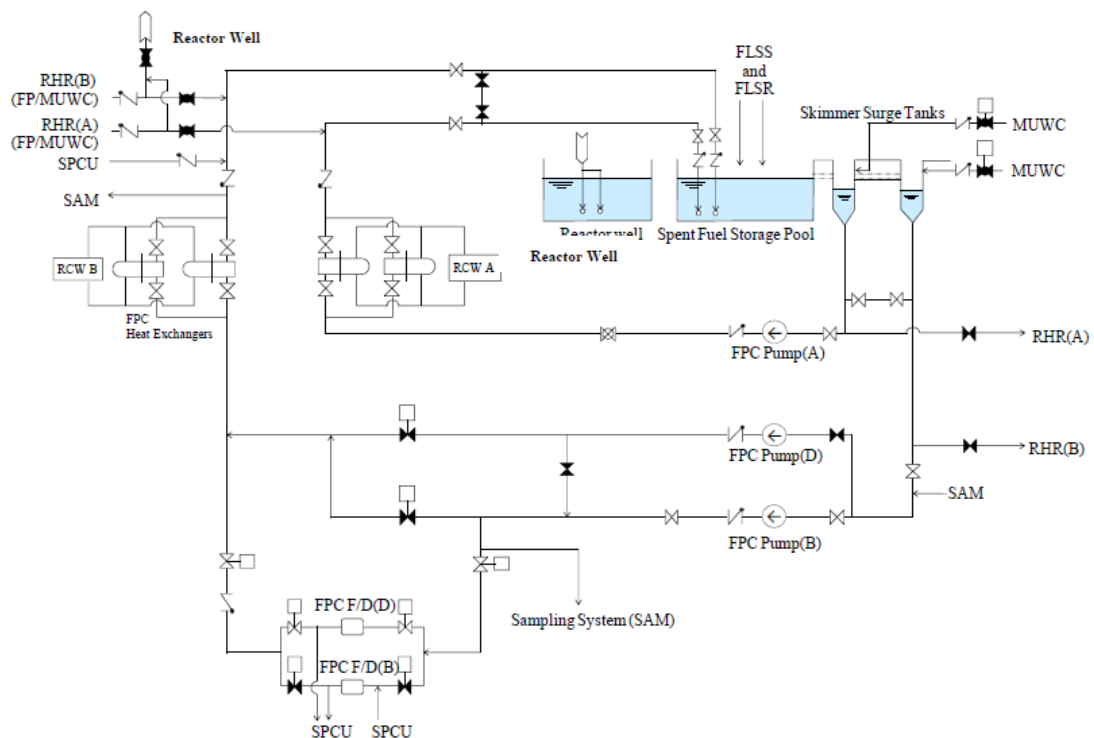
730. During GDA, Hitachi-GE placed a considerable amount of effort into providing a generic safety case which gives appropriate consideration to the hazards and risks which need to be controlled (and for risks, reduced SFAIRP), to safely manage the potential accumulation of radiolysis gases during normal operations. To a large extent this has been achieved, and the justification and supporting evidence provided in the generic safety case is clear and comprehensive. My specific conclusions on this topic, at this stage are:
- Hitachi-GE's claims and arguments are justified and supported by suitable and sufficient evidence. The one important exception is the OG system, which I have assessed separately in Section 4.6.2 of my report.
 - Hitachi-GE's radiolysis gas assessment methodology considers the accumulation mechanisms I expect to see.
 - Subject to satisfactorily resolving the two relevant Assessment Findings I have raised, **AF-ABWR-RC-17 and –18**, I am satisfied Hitachi-GE's methodology is an appropriate starting point for a future licensee to undertake similar assessments on the detailed UK ABWR plant design and layout.
 - Application of Hitachi-GE's methodology has already resulted in numerous modifications to pipework geometries for the UK ABWR design (most notably to systems contained in the T/B), which completely eliminate the potential accumulation of radiolysis gases.
 - It is clear, therefore, that relevant good practice in this context is met. The expectations of SAP of EKP.1 have been satisfied and the relevant guidance contained in the DSEAR ACoP (Ref. 108) has been appropriately applied. In the context of a generic safety case, and subject to resolving Assessment Findings **AF-ABWR-RC-17 and –18**, this provides confidence the risks associated with the accumulation of radiolysis gases during normal operations can be reduced SFAIRP.

4.4 Spent Fuel Pool and Suppression Pool

731. During a reactor outage typically around one quarter of the fuel in the core is replaced with new fuel. In rare instances the entire core may need to be removed to allow for inspections of the RPV, or other maintenance activities. The approach to refuelling adopted for UK ABWR is similar to all other BWRs. PCSR Chapter 19 (Ref. 117) provides information on fuel handling activities, including new fuel and refuelling. This chapter includes details of the system designs, functionality during normal operation and faults, safety categorisation and classification, and assumptions, amongst others. More detailed information is contained within the *Basis of Safety Cases on Spent Fuel Storage Pool and Fuel Pool Cooling, Clean-up and Makeup Systems* (Ref.118).
732. The SFP stores all the spent fuel assemblies removed from the reactor after being irradiated, within racks submerged in the pool, to maintain a sub-critical geometry. New fuel and any damaged fuel are also stored in the SFP. To allow the decay heat of irradiated fuel to reduce, a minimum cooling period within the SFP of 10 years is considered; the SFP is sized on this basis. Used fuel may be removed after this period, using a spent fuel cask. The SFP is permanently filled with demineralised water and cooled and purified by the FPC. Additional cooling can also be supplied by the RHR system and several systems can provide makeup water to the SFP to compensate for evaporation or leakage, including during fault conditions.
733. During refuelling operations the reactor well is filled with water from the Suppression Pool (S/P) (with additional supply also provided via the CST if necessary). The RPV head, associated equipment and upper reactor internals are removed and transferred to the dryer separator pit (DSP). The DSP is located on the opposite side of the reactor well from the SFP and is also filled with water from the S/P. The gates between the

SFP and DSP are removed, allowing the reactor well, DSP and SFP to become hydraulically linked. Fuel can then be removed from the reactor to the SFP, or vice versa, under water by using the fuel handling machine. This sequence is reversed to complete refuelling and close the reactor ready for the next operating cycle.

734. Of particular relevance to my assessment is the FPC. The primary function of this system is to cool the fuel held within the SFP, but it is this system that maintains the required chemistry within the SFP water. It consists of two independent closed circuit divisions, 'A' and 'B'. Overflowed water from the SFP flows into the skimmer surge tanks where it is circulated by the FPC 100% capacity pumps in each circuit. Two 50% capacity heat exchangers are installed on each circuit to remove heat to the Reactor Cooling Water (RCW) system. The clean-up components, consisting of a F/D, are installed in division 'B' to remove impurities from the circulating water. Division 'B' includes a spare pump and F/D to allow for maintenance. Each F/D is designed to provide 100% of the required clean-up capacity. The FPC is shown in Figure 35.



KEY: RHR – Residual Heat Removal system; SPCU – Suppression Pool Clean-Up system; FPC – Fuel Pool Cooling; SAM – Sampling system; MUWC – Make-up Water and Condensate system; RCW – Reactor Cooling Water system; F/D – Filter Demineraliser; FLSS – Flooder System of Specific Safety Facility

Figure 35: Fuel Pool Cooling and Clean-up System (FPC)

735. The Suppression Pool (S/P) is a large pool of water located at the bottom of the PCV. The pool is part of the concrete civil structure but lined with SS for the part immersed in water and coated CS for the nominally dry part above the water level. The water within the S/P is used for multiple purposes, including during both normal operations and during faults. The S/P is the water source for filling the reactor well and DSP during refuelling, as well as the water source for tests of the ECCS during normal plant operation. In order to maintain the required water quality, the water is circulated in the Suppression Pool Clean-Up system (SPCU). The SPCU transfers the S/P water through the FPC clean-up system F/Ds where soluble and insoluble impurities are removed. Given the connectivity between the S/P and SFP clean-up systems, and commonality of chemistry requirements, I consider both of these systems together in

this part of my assessment. The S/P is an important part of a number faults, and the chemistry related aspects of this are considered later in my report in Section 4.7. The S/P is shown in Figure 36. PCSR Chapter 16 (Ref. 98) provides further information on the S/P.

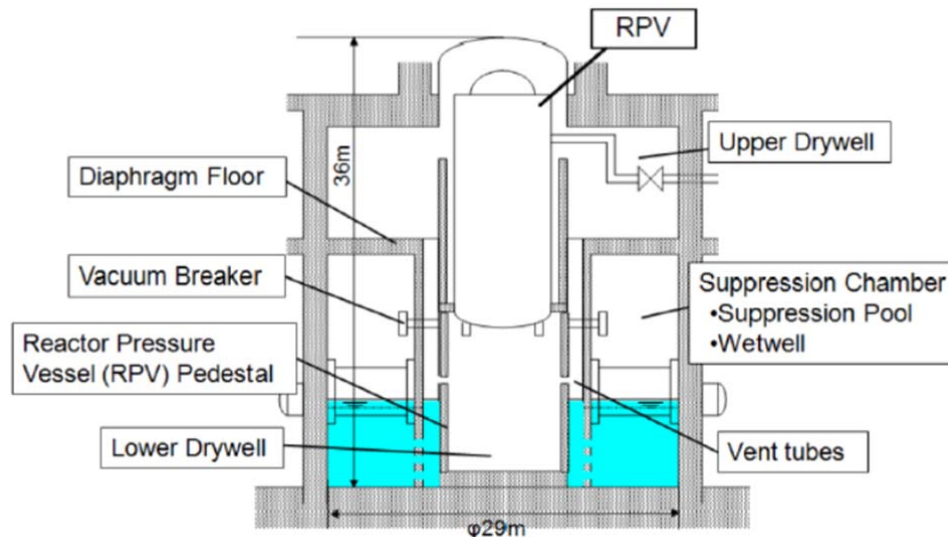


Figure 36: Suppression Pool schematic

736. While some changes have been made to the cooling aspects of the FPCSS design in response to specific UK requirements, no changes are proposed to the operating chemistry, materials or clean-up provisions for UK ABWR. These remain based on the J-ABWR reference design.
737. In this part of my assessment I consider whether the defined chemistry requirements for the water within the SFP and S/P are adequate, to reduce relevant risks SFAIRP, and also whether the engineered systems are adequate to control that chemistry. By default, this means my assessment does not consider all aspects of the SFP, S/P or refuelling activities. Further assessment of these systems and processes is also reported in other ONR assessment reports (for example Refs. 119, 120, and 121).
738. The main SAPs (Ref. 2) of relevance are those related to chemistry (ECH.1 to 4), ageing and degradation (EAD. 1 and 2) and integrity of metal components and structures (EMC.13). Aspects of radioactive waste and radiological protection were also relevant. I have also considered the guidance provided within the associated TAGs (Ref. 4).

4.4.1 Operating Chemistry

739. PCSR Chapter 23 (Ref. 11) includes dedicated sub-sections on the operating chemistry within the SFP and S/P. The fundamental basis for chemistry control within these systems is to maintain high purity water, without the need for any chemical additives. The basis for this choice is discussed further below. Several explicit “top claims” are made by Hitachi-GE against the operating chemistry within the SFP:
- “[RC SC8] *The UK ABWR Spent Fuel Storage Pool chemistry regime will contribute to the maintenance of the integrity of the structure and liner in the pool by operating within the limits and conditions.*”
 - “[RC SC9] *The UK ABWR Spent Fuel Storage Pool chemistry regime will contribute to the maintenance of the spent fuel integrity by operating within the limits and conditions.*”

- “[RC SC10] *The UK ABWR Spent Fuel Storage Pool chemistry regime will ensure that the radionuclide releases and exposure to worker is ALARP.*”
- “[RC SC11] *The UK ABWR Spent Fuel Storage Pool chemistry regime will ensure that the radionuclide releases and exposure to public is ALARP.*”

740. These “top claims” are supported by two specific claims. The first of these supports “top claims” [RC SC7] and [RC SC8], and is about minimising impurities. The second supports “top claims” [RC SC10] and [RC SC11], and is about minimising radioactivity:

- “[RC SC8.1] *Impurities will be kept below levels that may result in increased corrosion rates and minimised to ALARP in normal operations by the FPC.*”
- “[RC SC10.1] *Radionuclides in the Spent Fuel Storage Pool will be minimised to ALARP by the FPC and CUW.*”

741. I am satisfied that collectively, these claims are sufficient to describe the main safety related impacts of the operating chemistry with the SFP. I do note there is no specific “top claim” relating to maintaining the clarity of the SFP, which is important for conducting safe refuelling operations. The requirement is considered in the generic PCSR and other safety case documentation, but is not explicitly claimed. This means I am content it is only a presentational matter with the generic PCSR, as opposed to a specific gap in the wider generic safety case.

742. An important aspect of the UK ABWR SFP chemistry is that there is no requirement to include a soluble neutron poison in the water. This is because the SFP storage racks are designed to maintain sub-criticality, even in the most penalising case of a fuel handling accident. As explained in Ref. 122, they include fixed neutron poisons within the racking and geometric separation. It is beyond the scope of my assessment to consider the supporting analysis which leads to this conclusion. However, the conclusion of ONR’s assessment presented in Ref. 120 agrees that there is no need for a soluble neutron poison within the water. Excluding a soluble neutron poison in the SFP water therefore means that there are no similar requirements placed upon the S/P, which greatly simplifies the chemistry control requirements, and allows that water to be used for many other purposes across the plant.

743. PCSR Chapter 23 (Ref. 11) also considers the S/P chemistry requirements. The corresponding “top claim” is very similar to the one made on the SFP and relates to maintaining the integrity of the S/P:

- “[RC SC12] *The UK ABWR Suppression Pool chemistry regime will contribute to the maintenance of the integrity of structure and liner in the pool by operating within the limits and conditions.*”

744. Like the SFP, claim [RC SC12] is achieved by reducing the ingress of impurities. I consider these to be reasonable claims to make on the S/P, in relation to the function of supplying water as part of refuelling operations. It is relevant that Hitachi-GE has acknowledged the important link between these systems. Further arguments and evidence to support the claims are provided in several supporting documents, including, *Topic Report on Design Justification in Chemistry Aspect for Primary Water Systems* (Ref. 18), and in response to a number of RQs I raised during my assessment (Ref. 49). The operating chemistry requirements for the SFP and S/P are given in the *Water Quality Specification* (Ref. 20). I assess relevant aspects of these documents below, from the perspective of materials and radioactivity relevant risks.

4.4.2 Materials Degradation

Stored Fuel

745. The major material in contact with the SFP water is the zirconium fuel cladding. This includes both new and irradiated material, including fuel which has iron based oxide deposits formed during operation. This cladding continues to be the barrier to release of radioactivity held within the fuel and therefore it is important that the SFP environment continues to maintain the integrity of this cladding. Any fuel damaged during operations within the reactor is also stored within the pool.
746. The generic safety case claim is that the SFP operating chemistry, of essentially pure water, is adequate to store fuel in any anticipated condition. Ref. 123, "*GE14 fuel integrity evaluation during interim storage*", provides information on the longer term water storage of spent fuel (up to 140 years). At typical SFP temperatures the cladding oxide formation rate is negligible. However, if the SFP water quality deteriorates, the potential for accelerated cladding corrosion may arise. No quantitative data is presented here, or elsewhere regarding the impact of such degraded water quality. The PSCR (Ref. 11) correctly notes that such impurities may include:
- impurities mixed into the pool from the open surface;
 - impurities brought into the pool with the fuel or components on which the impurities adhere to;
 - CPs and FPs from the core during refuelling;
 - mixing material during refuelling and other operations; and
 - residual chemicals used for cleaning or flushing water after pool cleaning.
747. Another potential source is leakage of Reactor Building Cooling Water System (RCW) coolant from the FPC heat exchangers.
748. The only reference for any data of the impact of such impurities in Ref. 123 is a single set of experiments, conducted with relatively high impurity levels but not covering all impurities identified above. *Management of Damaged Fuel* (Ref. 91) considers the storage of damaged fuel within the SFP. The impact of chemistry is mainly concerned with the release of additional radioactivity to the water, rather than any specific considerations for the water quality requirements. In effect, it is argued that the requirements for intact and damaged fuel are the same. While I am satisfied that the premise of storage under pure water conditions is satisfactory, I am dissatisfied with how the safety case considers and justifies the basis for the impact of abnormal chemistry conditions. The response to RQ-ABWR-1028 (Ref. 49) indicates that the impact of impurities in the SFP will be described in *Topic Report on Impurity Ingress* (Ref. 30). This is only partly true, with the consideration of chloride, for example therein, only reflecting SCC risks. Overall, no data, therefore evidence, is provided to support the impact on fuel integrity, particularly over the longer timescales. This is important in determining what limiting values for impurities are tolerable. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-19: The UK ABWR generic safety case for the spent fuel pool assumes pure water is used as the storage medium, under adequately controlled conditions. During GDA, ONR's chemistry assessment established there were some limitations in the associated evidence provided; including a demonstration of the impact abnormal chemistry operating conditions may have on the integrity of the spent fuel itself, and spent fuel pool structures. The licensee shall review and update the UK ABWR safety case to provide evidence to demonstrate the impact deteriorated spent fuel pool water quality has on the integrity of the stored fuel itself, and/or the spent fuel pool structures, system and components. This shall include a proportionate justification to show adequate controls are in place to reduce relevant risks SFAIRP.

749. *Water Quality Specification* (Ref. 20) includes the requirements for the SFP water quality, and similarly for the S/P water. They are presented in Annex 5 of my report. Information is provided in Ref. 20 on the basis for these; the expected values are based upon a statistical review of J-ABWR OPEX, whereas the limit values are taken

directly from the corresponding reactor water limits. The requirements for the S/P are identical, except the expected values are slightly higher.

750. I queried a number of these ORs as part of RQ-ABWR-1335 (Ref. 49). Unfortunately a number of the responses provided have not been reflected within Ref. 20. I also note that the values given in the water quality specifications, and reflected in the PCSR, are different to those given in Refs. 55, 91 and 123. The requirements in the latter reports are more stringent. Collectively, I am not content these ORs are adequate, because:
- The defined limit values are not substantiated, with the current basis being to use reactor water limits (which are defined for other purposes, specifically for SCC mitigation of SS not long term storage of zirconium alloys).
 - pH as a control parameter is not justified.
 - Using direct conductivity has not been demonstrated to be suitable to detect small increases in impurities.
 - Silica is not specified despite being a control parameter in the reactor water and the main source of this impurity likely to be during outage periods, and;
 - No controls are specified to ensure water clarity, such as iron, suspended solids or turbidity, despite the importance of this parameter.
751. Considering the importance of these controls in delivering the safety case claims, I expect the future licensee to demonstrate the gaps identified in the generic safety case have been addressed. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

SFP and S/P Liners

752. The SFP, including the cask pit and DSP are lined with SS (Ref. 118). The justification for this material is provided in Ref. 55. Hitachi-GE considers that both SS and CS could be potentially suitable, but the former is selected mainly on the basis of the lower corrosion rate. While I am not content that the material selection optioneering properly reflects the benefits and detriments of the different choices examined, I am content with the decision. This is consistent with other SFPs in other nuclear facilities, including the J-ABWR reference design.
753. Ref. 55 provides Hitachi-GE's materials justification for the S/P liner. The liner material selected for the S/P is discontinuous, using CS and SS in different locations. I am content with the adoption of SS for those parts that are in continuous direct contact with suppression pool water, given the environmental conditions expected. The upper and lower drywell air spaces use CS treated with a zinc paint and epoxy coating to prevent corrosion. Hitachi-GE's case is that the materials selection decision is based on the conservative assumption that any corrosion resistant coating applied, had failed. They argue that this is very unlikely based on OPEX. The performance of the coating will be managed throughout the 60 year design life of UK ABWR primarily by in-service inspection. I consider this to be adequate to justify the use of CS for parts of the RCCV liner and I would consider developing the necessary arrangements to manage the in-service inspection to fall under normal business for a future licensee. From a chemistry perspective, the impact of the S/P coating on iodine behaviour during accidents is an important factor, considered further in Section 4.7 of my report.

FPC and SPCU Systems

754. Most of the surfaces in contact with the water in these systems are austenitic SS. The spent fuel rack is constructed from boronated SS, to meet criticality control requirements (Ref. 118). The main exception to this is the pipework upstream of the F/Ds within both the FPC and SPCU system. These pipework sections are made from CS.

755. In general, the low operating temperature and high water purity expected means that corrosion rates will be low and would need to suffer a sustained increase to be significant. This is particularly true for the SFP and FPC which is recirculated and cleaned-up. The S/P and SPCU system are expected to be stagnant for long periods and therefore may be more prone to degradation if degraded water quality was present. I discussed the defined ORs earlier in my assessment in the context of fuel cladding integrity within the SFP. My conclusions reached there remain relevant here. Namely, I would not consider the limit values defined to have been demonstrated to be adequate. In particular long term operation at those values would almost certainly lead to damage and failures in the CS and SS components of these systems. **This has been captured as part of Assessment Finding, AF-ABWR-RC-19.**
756. The material choices for these systems are identical to the J-ABWR reference plant. These choices are justified in Ref. 55. The case put forward by Hitachi-GE is that overall SS is the material of choice, although the material degradation risks in the SFP environment are not significant. I am content this is a suitable decision, and is consistent with other operating nuclear reactors, both BWR and PWR. However, given the low temperatures and high water purity maintained within the SFP and S/P, Hitachi-GE argues it is acceptable to use the less corrosion resistant CS in upstream pipework that directly connects to the F/Ds. Hitachi-GE argues any corrosion products released by this pipework will be efficiently removed before contaminating the bulk water.
757. The key argument is that the deciding factor in the choice is the clarity of the SFP water, which needs to be maintained “*completely clear*” to allow safe handling of irradiated fuel. Whilst CS meets all the other requirements, its higher corrosion rate would lead to a risk of particulate iron build-up in the SFP over time, ultimately leading to a “cloudy” SFP, which would hinder safe operations⁵⁵⁵. As demonstrated by available OPEX, this is an important consideration. During GDA I asked (RQ-ABWR-1464 (Ref. 49)) Hitachi-GE whether consideration had been given to the impact of this increased corrosion product burden on the capacity of the F/D, generation of radioactive waste and worker doses. Overall, while some additional clarity on the generic case was provided, I remain unconvinced over the use of CS for the upstream sections of the FPC. In particular, considering the continuous use of this system and the extended design life of UK ABWR (60 years as opposed to 40 years for J-ABWR), insufficient evidence has been provided in the UK ABWR generic safety case to demonstrate why it is not reasonably practicable to use SS throughout the SFP for all main piping. As explained earlier, normal practice for the majority of LWRs is to use entirely corrosion resistant materials for these applications. Despite the lack of evidence, if adequate evidence can be provided to substantiate the decision to use CS, this might prove to be the “ALARP option”. I do not consider this matter needs to be resolved during GDA, given any subsequent material changes would not undermine the fundamental generic design. I expect the future licensee to fully justify and provide adequate evidence to underpin this choice, taking all relevant factors into account. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-20: Normal practice for spent fuel pool clean-up systems is to select entirely corrosion resistant materials. Some large sections of the UK ABWR spent fuel pool clean-up system main piping uses carbon steel. The licensee shall review and update the UK ABWR safety case to demonstrate that using carbon steel for sections of spent fuel pool clean-up system main piping is not detrimental to water quality, doses to workers and the generation of radioactive waste, and show that relevant risks are reduced SFAIRP.

⁵⁵⁵ There are other relevant factors, including the integrity of the piping. Assessing the potential impact of the materials selection on the overall integrity of the system was beyond the scope of my GDA chemistry assessment.

758. Considering the SPCU system is only used intermittently, and the lower safety significance of this system, I am content with the adoption of CS for some identified parts of the piping. This assumes that resolution of Assessment Finding, **AF-ABWR-RC-19**, does not identify any matters which undermine this conclusion.

4.4.3 Radioactivity

759. Both the SFP and the S/P in UK ABWR will contain some level of radioactivity as a result of normal operations. This may occur as a result of both refuelling operations or from contamination from other systems. The SFP in particular will contain some radioactivity transferred for the reactor, or from releases of stored irradiated fuel. The radioactivity released from the spent fuel, which may include a range of FPs or CPs, including damaged fuel, is removed by the FPC (or SPCU). Hitachi-GE recognise that the allowable concentration of the radionuclides in the SFP is determined by the radiation level at the operating floor, where maintenance work is carried out, in addition to the potential environmental impacts of discharging evaporated SFP water. Similar requirements do not apply to the S/P given the inaccessible location, but radioactivity still needs to be minimised, particularly following the return of water used in the various pools during an outage.
760. The amount of radioactivity expected to be present within the SFP and S/P in UK ABWR is calculated by Hitachi-GE in accordance with their PrST methodologies, as detailed in my assessment (Ref. 79). I was content with this approach and resulting values in the context of the generic safety case. This confirms that the expected concentrations within the S/P are very low.
761. No radioactivity is produced within the SFP itself and the impact of the chemistry on subsequent release or distribution is small, provided the chemistry is controlled within boundaries close to the expected conditions of essentially pure water. From a direct generation of radioactivity perspective the chemistry ORs defined within Ref. 20 can therefore be considered adequate.
762. One parameter that is important is the SFP temperature. The higher the temperature the more radioactivity that is released from stored irradiated fuel. There was some confusion over the expected temperature for UK ABWR. This was clarified in RQ-ABWR-1028 (Ref. 49). In summary the expected temperature is 35 °C, rising to 45 °C under conditions of a full core offload. The temperature is limited to 52 °C for worker access and 65 °C is the ultimate limit to protect the concrete SFP structure. It is beyond the scope of my assessment to consider the adequacy of these upper values (for worker access and concrete integrity), however the increase in radioactivity between these temperatures will be important, but not to an extent that cannot be safely managed. This is within the range of BE and DB values assessed in Ref. 79.
763. The behaviour of radioactivity within the SFP is considered by Hitachi-GE in, *Topic Report on radioactivity behaviour in UK ABWR* (Ref. 80). In particular, Ref. 80 provides an outline of how radioactivity within the SFP changes during a refuelling cycle. This highlights several important points that merit further consideration.
764. At shutdown the reactor water radioactivity will mix with the SFP (and DSP, reactor well and dryer separator pit) water. The concentration that is present in the reactor water when the RPV is opened is therefore important to control as this directly influences the resultant concentrations in the operating deck water pools. The *Water Quality Specification* (Ref. 20), details the consideration given by Hitachi-GE to whether controls are necessary during these operations, based on information described in *Topic Report on Start-up and Shutdown Chemistry* (Ref. 36). The conclusion reached is that there is the need to provide limits on both ^{131}I and ^{60}Co in the reactor water prior to removal of the RPV head or reactor cavity flooding respectively. No limit is required for ^{133}Xe ; given this noble gas nuclide is rapidly

removed to the gas phase during operations. Collectively these controls are claimed to be sufficient to ensure worker doses are minimised during outages, particularly in the case of any fuel damage. I welcome the inclusion of such controls for UK ABWR, which has not been common practice for Japanese plants (although other methods have been employed). I have previously considered these limits in an earlier part of my assessment, under Section 4.3.1.2. Overall, I am content such limits are specified, but have unresolved questions regarding the completeness and adequacy of the values proposed by Hitachi-GE during GDA. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

765. In addition, a separate control is specified for the concentration of ^{60}Co within the SFP water. The purpose of this control is to limit ORE for workers in the vicinity of the SFP. Radioactivity deposited on the fuel surfaces may over time undergo dissolution and spallation, thus introducing an ionic and particulate CP burden to the SFP water. There is uncertainty in this, but the values used by Hitachi-GE are based upon the BE and DB values for the UK ABWR and are therefore calculated based on the corresponding reactor water concentrations rather than having an explicit link to radiological protection. ^{60}Co is selected as a representative surrogate for all CP nuclides. I am satisfied that such a control is beneficial, but would expect the future licensee to consider the specified values which at the moment appear quite arbitrary and are not linked to a particular hazard. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**
766. I also note that, for the purposes of estimating the quantity of deposits released from the spent fuel cladding for the decommissioning EUST, Hitachi-GE conservatively assume that spallation of the entire loose fuel deposit material will occur and a fraction of this will accumulate on the base of the SFP. OPEX provided by Hitachi-GE suggests that such conditions are unlikely, with most operating plants exhibiting low levels of such accumulated material. I therefore consider this to be a reasonable assumption, as per Ref. 79.
767. Radioactivity in the SFP is continuously removed by the FPC. During refuelling operations the concentrations of radioactive species may temporarily increase, but a steady state level is reached thereafter. This FPC will rapidly remove all soluble and insoluble radioactivity in the SFP water that was introduced by the mixing with reactor water. I consider the adequacy of the FPC in the next section (4.4.4) of my assessment. The exception to this is for the ^3H concentration within the SFP (or S/P) water which would continuously increase at every refuelling outage as the RPV inventory is successively added. As such ^3H levels in the SFP water will reach a steady state after typically three or four fuel cycles, controlled by evaporative loss from the pool and the liquid discharge management policy applied by the future licensee. Managing ^3H was considered as part of my assessment in response to RI-ABWR-0001 (Ref. 79), which agreed that such a steady state value could be reached. Ref. 36 also provides consideration of whether an OR is necessary to limit the SFP ^3H concentration. On the basis that; ^3H concentrations in BWRs are relatively low (compared to PWRs for example), OPEX from operating plants suggest that this can be readily managed and calculations of the impact on worker dose show that high concentrations are necessary, Hitachi-GE conclude that an OR related to the SFP ^3H concentration is not necessary. I am satisfied that sufficient evidence has been provided for GDA to support this conclusion.
768. Leaking fuel pins may also provide a continuous source of FPs to the SFP. Any soluble and particulate material lost from the defective pin will be treated via the FPC, while volatiles will be released from the pool and filtered and extracted by the building HVAC system. The impact of storing damaged fuel in the SFP is considered in Hitachi-GE's *Management of Damaged Fuel* report (Ref. 91). Ref. 91 provides evidence which demonstrates that the impact on SFP radioactivity is limited. It is noted that in exceptional circumstances, means are available in the extant generic design to further

minimise any impact, for example by varying the F/D resins or installing temporary treatment equipment. I am satisfied with this argument. However, no justification is provided for why radioactivity (aside from ^{60}Co) does not need to be controlled or limited within the SFP. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

4.4.4 Chemistry Control Systems

769. The F/Ds within the FPC function to clean-up both the SFP on a continuous basis and the S/P, as required. The overall engineered design basis this system is presented in the *Basis of Safety Cases on Spent Fuel Storage Pool and Fuel Pool Cooling, Clean-up and Makeup Systems* (Ref. 118). The important features of the system related to chemistry control are then expanded upon in, *Topic Report on Design Justification in Chemistry Aspect for Primary Water Systems* (Ref. 18). During my GDA assessment of this system, I asked several queries (RQ-ABWR-1028, 1335 and 1457 (Ref. 49)) related to the adequacy of the F/D clean-up technology selected. As the technology used in this system is the same as the CUW system, my assessment of the clean-up technology selected for the CUW system, reported earlier (Section 4.3.5.2) in my report, is also directly relevant here.

Design Basis

770. Each of the two 100% capacity parallel F/D trains located in Division 'B' consists of a single F/D vessel. These remove both soluble and insoluble impurities from the water flow, using pre-coat ion exchange resins as in the CUW system. This system is designed primarily to fulfil removal of low-level chronic contamination of the SFP, but also has sufficient capacity to handle more acute impurity ingress events. Ref. 19 provides a typical calculation to support this, as well as explaining factors such as sizing. I am content that this is a reasonable basis for the system design, and note that acute contamination of the SFP is unlikely, having never been experienced in Japanese plants. I do note that the FPC does not have a high capacity to remove soluble species, as Hitachi-GE claims that there is no need to take into account events such as sea water ingress that markedly and quickly decrease water quality. I take further comfort from the redundancy provided by the two 100% capacity F/D trains in this regard. **This has been captured as part of Assessment Finding, AF-ABWR-RC-19.**

771. The safety function associated with clean-up of the SFP water is classified as Safety Category C and the components to deliver it are designed to meet Safety Class 3 requirements (note that other functions carry higher safety significance, for example the cooling functions are Category A, Class 1). This has an impact on the design, whereby the F/Ds are not part of Division 'A' of the FPC. The split into two divisions allows the Category A cooling functions to have sufficient redundancy, which is not required for the Category C clean-up function. It is argued that, for the clean-up function to be unavailable Division 'B' would need to fail, and even if it did the deterioration in water quality would be slow. However, the response assumes that the resolution of this matter would require the installation of an addition F/D within Division 'A'. This would be a significant design change and is clearly not reasonably practicable. However, similar functionality could be achieved by the addition of connections which would allow Division 'A' to use the existing F/Ds installed in Division 'B', which is not considered. While this may be advantageous from an operation perspective, I am content that it is not a significant safety risk and therefore do not consider further. However, I do consider that this matter should be addressed by the future Licensee. **I consider this to be a Minor Shortfall:**

MS-ABWR-RC-07: During GDA, the UK ABWR generic design of the spent fuel pool cooling and clean-up system was modified to add an extra train of cooling. As part of this modification extra clean-up capacity was not added to the system. Only one train

of the system is therefore capable of cleaning-up the spent fuel pool water. The licensee should review and update the UK ABWR safety case to justify whether there are reasonably practicable measures that can be applied to allow operation of the existing clean-up provisions using both cooling trains, or using other similar means.

772. Hitachi-GE has assigned the SPCU System a categorisation and classification of C3. Based on the chemistry claims placed on this system I am content with this, for similar reasons as described above.
773. In principle there are several technologies which could be utilised to fulfil the clean-up requirements for the FPC. Ref. 18 provides an overview of these, including the historical development and OPEX from other BWR plants. From this it is clear that pre-coat F/D is one of the most common system configurations in BWR plants around the world. Hitachi-GE contends that, based on OPEX from existing plants in Japan, the water quality can be achieved by this technology. As this is the same technology as the CUW, backwash and pre-coating facilities are able to be shared, which enables efficiencies and the system scale to be minimised. I am content with these arguments, provided the adequacy of the system can be demonstrated.
774. The FPC flows are defined to meet the higher of either, the whole volume of water in the SFP twice a day or the whole volume of water in the upper pools (i.e. reactor well, DSP, dryer separator and SFP combined) once a day. This equates to a flow rate of around $\blacksquare \text{ m}^3 \text{ hr}^{-1}$. The flow through the F/Ds is controlled locally but can treat the full flow. The SPCU system flow through the F/Ds is the same. This equates to a lower linear velocity through the beds than the CUW system. This value is not set based on chemical performance, rather to ensure sufficient turn-over of the SFP water, particularly during outages. I judge this flow rate to be adequate from a chemistry perspective, given the likely scale, nature and severity of abnormal chemistry that could occur in the SFP.

Adequacy of the System to Deliver Chemistry Requirements

775. Ref. 118 indicates the basis of the system design in terms of clean-up requirements. As noted earlier, the values given here are smaller than those within the *Water Quality Specification* (Ref. 20). Hitachi-GE claims a DF of 10 for all species, but do not provide evidence or OPEX to support this. The basis for this claim is therefore the same as that for the CUW system described in Section 4.3.5.2. I note that this is for a different system (CUW) operating under different conditions (flow and chemical environment). Importantly the resin type used within the FPC is different, operating in the carbonate form, which will affect the chemical performance.
776. Ref. 18 does provide a typical calculate that, in the event that the F/D was to operate with the maximum chloride or sulphate ion concentration experienced in J-ABWRs for the entirety of a cycle (equivalent to the expected values for UK ABWR), the resin would be required to be changed around two or three times per cycle. This compares to the normal exchange rate of once per cycle. If the system were to operate at the limiting OR concentrations, the operating period would be measured in hours, rather than days. Consideration is given to factors which might negatively impact this performance, from which I agree with Hitachi-GE's assertion that the SFP conditions are much less onerous than for the comparable CUW system and significant deterioration in water quality is unlikely to occur. Similarly, I do note that some evidence for FPC performance is described in Ref. 79. However, this too is limited and does not distinguish between soluble or insoluble species and concentrates on radioactive species and not those which might challenge the FPC performance. As noted earlier, the lack of consideration given to quantifying the potential for impurity ingress events is important here.

777. I was particularly interested in the filtration performance of the F/Ds. During operation, the SFP water in UK ABWR will continuously overflow into the Skimmer Surge Tank and insoluble impurities on the surface of the SFP with a diameter over [REDACTED] mm are entrained in the tank mesh strainer. All other particulate is passed through the F/D. Despite asking for this in multiple RQs (RQ-ABWR-1028, 1335 and 1457 (Ref. 49)), Hitachi-GE was unable to provide me with a demonstration that the F/D can deliver an adequate filtration performance. In effect the performance is inferred from other data, which may or may not reflect the behaviour of UK ABWR. I consider this to be an important aspect of the FPC system that is inadequately justified in the safety case.
778. Overall, for the same reasoning as described earlier in my report for the CUW system (Section 4.3.5.2), and based on my assessment above, I do not consider this to be a sufficient demonstration of the adequacy of the chemical performance of the FPC clean-up function. While I acknowledge some aspects of the case presented as adequate (for example the basis of the design, functions, capacity and on-line measurements), I do consider there are other aspects where insufficient evidence is presented. These relate to providing evidence which support the expected clean-up performance. Despite these deficiencies in the safety case, I remain content that sufficient information has been provided to suggest that the fundamental design of the FPC could be shown to be adequate from a chemistry perspective. This will likely require further OPEX from a future licensee. My judgement at this stage is mainly based on my own experience of other similar plants and information contained outside the Hitachi-GE's responses. **This has been captured as part of Assessment Finding, AF-ABWR-RC-14.**

Sampling and Monitoring Systems

779. As with other systems where there is a requirements to control the operating chemistry, appropriate and representative sampling is a key activity for operation of the FPC. Sampling of this system was one of the examples provided during GDA. The sampling provisions for this system are based on the same design principles and considerations discussed earlier in my assessment for the CUW system specifically (Section 4.3.5.2). My conclusions there, in particular those associated with Assessment Finding, **AF-ABWR-RC-16**, remain relevant here.
780. Three sample points, the F/D inlet, F/D outlet and heat exchanger outlet are identified for the SFP in Refs 104 and 124. In general, I am content with the information provided for this system, given that current status of the generic design. However, it is clear that further consideration needs to be given to this system. For example, from the information presented, I note that sample flows for the FPC are low, compared to the CUW system. No information is given on piping length to indicate how much flushing may be required to ensure representative grab sampling. This may also affect sampling for corrosion products, given the likely low flows and particulate deposition. I am satisfied that Assessment Findings, **AF-ABWR-RC-02 and -16** capture my specific expectations in this regard.
781. It is also notable that various parameters are monitored using on-line instruments. This includes temperatures (SFP, pump inlet and heat exchanger outlet), differential pressure (each F/D) and conductivity (inlet and outlet of each F/D and outlet of the heat exchanger). Temperature and conductivity excursions trigger alarms in the control room. The inclusion of such measures, particularly conductivity, reflects the importance of maintaining suitable control over the water chemistry.
782. I am broadly satisfied with the consideration given to sampling and monitoring within the FPC for this stage of the generic design. My conclusion broadly aligns with the independent view provided by my TSC (Ref. 106) and information available to ONR (Ref. 44). While I have identified areas where further development will be required by

the future licensee, I am content that the basic generic design and philosophy adopted for UK ABWR is sufficient.

4.4.5 Spent Fuel Pool and Suppression Pool Summary

783. Hitachi-GE have provided a generic safety case for the chemistry-related aspects of the SFP and S/P systems which, from a chemistry perspective, considers all of the main nuclear safety related risks and identifies a set of ORs to bound the limits and conditions which define the safe operating envelope. Furthermore, they have considered aspects relating to the demonstration of the adequacy of the engineered systems to support the operating chemistry defined. The generic design itself is largely based upon the J-ABWR reference design, which has been operated successfully for a number of years. In that regard, the scope and content of the generic safety case broadly meets my expectations.
784. I have identified a number of areas where the future licensee will need to review, revise, expand or complete the safety justifications provided in the UK ABWR generic safety case. These include matters relating to:
- providing evidence for the impact of abnormal chemistry on stored spent fuel;
 - the adequacy of the defined ORs;
 - the selection of CS for some large sections of FPC main piping; and
 - adequately justifying the chemical clean-up performance of the FPC.
785. During my assessment these matters have been formally captured either as individual, or part of wider Assessment Findings, including: **AF-ABWR-RC-02, -14, -16, -19 and -20**, and also one Minor Shortfall, **MS-ABWR-RC-07**. I am broadly satisfied these matters can be satisfactorily resolved by a future licensee, and the fundamental generic design proposed for UK ABWR could be demonstrated to be acceptable, from a chemistry perspective, and reduce relevant risks SFAIRP.

4.5 Component Cooling Water Systems

786. Component Cooling Water (CCW) systems support safe reactor operations by providing a heat sink for numerous SSCs, ensuring they are able to deliver their claimed safety functions. CCW is a generic term used to describe multiple systems which are all closed cooling circuits that continually re-circulate water, as a means to remove reject process heat. For UK ABWR, the CCW system consists of five independent sub-systems:
- Reactor Building Cooling Water System (RCW);
 - Turbine Building Cooling Water System (TCW);
 - HVAC Emergency Cooling Water System (HECW);
 - HVAC Normal Cooling Water System (HNCW); and
 - Emergency Equipment Cooling Water System (EECW)
787. Of the five UK ABWR CCW sub-systems mentioned above, my assessment during GDA focussed on the RCW and the HECW. These systems supply cooling water to UK ABWR SSCs of the highest nuclear safety significance and each uses similar materials. They are therefore considered representative of all of the CCW systems.
788. The RCW system is a very important system and is separated into three divisions. The system itself carries a nuclear safety A1 categorisation and classification. Each division provides cooling to a number of nuclear safety significant SSCs, but also SSCs which carry no nuclear safety classification. The system is designed to function during both normal operations and some fault conditions. Amongst others, some of the A1 SSCs the RCW provides cooling for include components in the RHR system, CUW system

and Emergency Diesel Generators. Each division of the RCW comprises the following main SSCs:

- Main piping.
- Three pumps and three heat exchangers arranged in parallel. Heat exchange capacity provided by all three divisions is required for normal operations and fault conditions.
- A surge tank which is shared between the RCW and HECW, which allows for volume expansion in each system.

789. The RCW also features two shared tanks to perform chemical additions. One tank is shared between RCW Division 'A' and the TCW; the other is shared between RCW Divisions 'B' and 'C'. Makeup water is added to the RCW using the Make-up Water Purified (MUWP) system. An overview of one of the divisions of the RCW is provided in Figure 37 below.

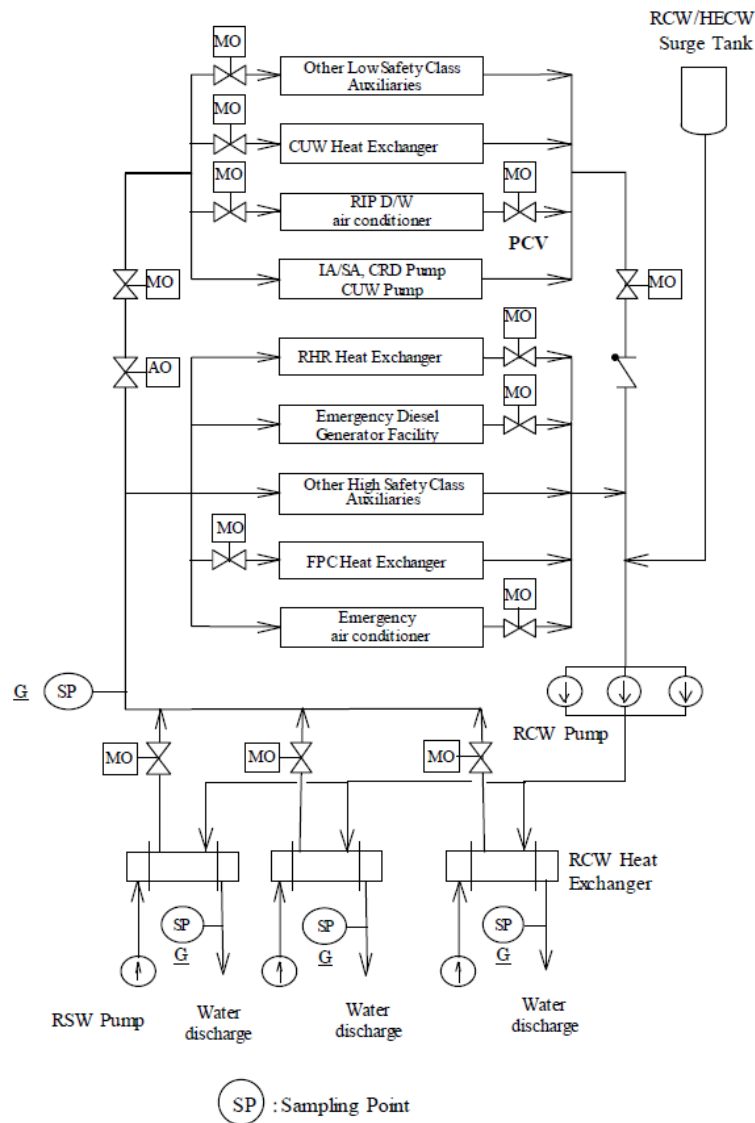


Figure 37: Overview of the UK ABWR RCW showing one division (Ref. 19).

790. Fundamentally, the HECW is very similar. It is divided into three non-identical divisions. The system functions across all normal operating modes and during some fault conditions. It provides chilled water to the cooling coils used in HVAC systems

and other local cooling units. Some of the SSCs the HECW system provides cooling for include the Class 1 EDG local cooling units and Heat Exchanger Building Emergency HVAC. Each division of the HECW comprises the following main SSCs:

- two cooling loops. One supplying chiller-cooling coils and one supplying chiller-cooling units;
- four pumps and four chillers, each of which are cooled by the RCW;
- one chemical addition tank to add corrosion inhibitor; and
- a connection to the shared RCW/HECW surge tank.

791. Hitachi-GE's overall case for the UK ABWR CCW system is presented in a number of documents at various levels within their hierarchy of safety case documentation. Chapter 16 of the PCSR (Ref. 98) provides details of the system engineering and design, but there are important links to other PCSR chapters, including Chapter 23 (Ref. 11). Beneath the PCSR there are a number of more detailed documents. The principal documents of relevance to my assessment include:

- *Topic Report on the Design Justification in Chemistry Aspects for Ancillary System* (Ref. 19);
- *Basis of Safety Cases on Reactor Building Cooling Water Systems* (Ref. 125); and
- *Reactor Building Cooling Water System System Design Description* (Ref. 126).

792. Similar to other aspect of my assessment, the principal SAPs relevant to my assessment of the CCW systems are ECH.1, ECH.3, ECH.4, EAD.1 and EAD.2.

4.5.1 Operating Chemistry

793. Given the function(s) CCW systems fulfil, Hitachi-GE make the following "top claim" and claim in Chapter 23 of the PCSR (Ref. 11):

- "[RC SC15] *The chemistry of the component cooling water contributes to minimise the corrosion of its system materials to maintain their integrity and heat transfer function by ensuring a corrosion controlled environment when operated within limits and conditions.*"
- "[RC SC15.1] *Corrosion of SSCs of carbon steel, stainless steel and copper will be minimised by the addition of nitrite based corrosion inhibitors*".

794. I consider the claims presented by Hitachi-GE are reasonable and are the main chemistry claims I would expect to see for such CCW systems, given the objective for chemistry control in these systems is predominately to mitigate the risk of materials degradation.

795. There are several potential chemistry regime choices that can be applied to closed cooling water systems. Hitachi-GE's approach for GDA was to present one single chemistry regime for the RCW and HECW and argue that, because this chemistry regime works for these two sub-systems, the same chemistry regime can be applied to all five CCW sub-systems to achieve the same objectives, and therefore fulfil the claims made in the generic PCSR (Ref. 11). Hitachi-GE contend this approach is appropriate because:

- RCW and HECW sub-systems have the highest safety significance;
- other CCW sub-systems are fabricated from similar (largely the same) materials; and
- it does not foreclose the option for a future licensee to implement a different CCW system chemistry regime to the one selected and justified during GDA.

796. *Design Justification Report on Chemistry Aspects of Ancillary Systems* (Ref. 19) is Hitachi-GE's principal submission which contains the justification for the chemistry regime selected for UK ABWR CCW systems during GDA. Ref. 19 states the objectives of chemistry control in the CCW system are to:

- minimise corrosion;
- control microbiological growth;
- control the deposition of suspended solids; and
- prevent scale deposition.

797. I agree these are suitable objectives and I am satisfied that they appear appropriate considering the chemistry claims ([RC SC15 and 15.1]) made by Hitachi-GE on the CCW system. Ref. 19 also presents a broad range of information on different CCW system operating chemistries. Anti-microbial agents are briefly discussed, but their specific application to UK ABWR is not explored further. The information largely consists of Japanese OPEX and a review of authoritative industry guidance on closed cooling water system chemistry choices. As presented, the information is more akin to a technical literature review. An objective safety case style approach is not adopted. Nevertheless, I am satisfied that sufficient information is presented to constitute a suitable starting point for a future licensee, when deciding which potential closed cooling water system chemistry regime(s) to apply for UK ABWR. On the basis of the information presented in Ref. 19, Hitachi-GE argues the following chemistry regime is a suitable choice for the UK ABWR CCW system:

- A nitrite-based corrosion inhibitor with sodium hydroxide used as a pH modifier, as required.
- Potential options for the addition of an azole are identified to suppress copper corrosion (i.e. for the HECW), as required.

798. Hitachi-GE selects nitrite as the preferred corrosion inhibitor on the basis that it is the most widely used corrosion inhibitor both for Japanese BWRs and internationally, for CCW systems fabricated from CS. In Ref. 19, Hitachi-GE provides further specific technical details to justify why the anti-corrosion properties delivered by the nitrite system are adequate for UK ABWR and also to underpin the likely performance of azoles. Based on this information, my own knowledge of typical closed cooling water chemistry regimes applied at nuclear power plants and the information available in authoritative industry guidelines, I am satisfied that for GDA, Hitachi-GE has provided sufficient evidence to demonstrate an appropriate CCW system chemistry regime, which delivers the claims made ([RC SC15] and [RC SC15.1]), has been provided for UK ABWR.

799. Clearly, the adequacy of the chemistry regime selected is determined by the materials selected. Both are inextricably linked. Section 4.5.2, below, presents my assessment of Hitachi-GE materials selection justification for the UK ABWR CCW system.

4.5.2 Materials Degradation

800. Hitachi-GE's materials selection justification for the UK ABWR CCW system is presented in *Materials Selection Report for Station Auxiliary Systems* (Ref. 55). For the RCW and HECW, Ref. 55 presents justifications for the following "representative SSCs". The materials selected for them are then applied to all UK ABWR CCW sub-systems. The scope of the RCW and HECW SSCs covered by Ref. 55 includes various piping, valves and heat exchangers. I am content these are a representative selection. In summary, the materials selected for UK ABWR for these SSCs, and the basis for Hitachi-GE's decision for each of them is:

- RCW system main piping – CS. The system operates at relatively low temperatures (mostly <60°C) and pressures. Localised corrosion mechanisms

are not a major concern. Chemical corrosion inhibitors are effective at mitigating general corrosion.

- RCW system heat exchangers – CS and titanium. Titanium is selected for its superior corrosion resistance to account for the fact that the RCW system also rejects heat to the sea water cooled circuit of the RSW system.
- RCW/HECW system surge tanks – SS. The material selected for UK ABWR is different to J-ABWR (which uses lined CS). SS has been selected for its superior corrosion resistance.
- HECW system chiller – copper is selected for the evaporator and condenser tubes on the basis of thermal conductivity requirements.

801. On this basis, I am satisfied Hitachi-GE has applied appropriate materials which take due account of the operating environment and main materials degradation threats expected in the UK ABWR CCW system. The materials selected are consistent with authoritative sources of industry guidance on chemistry control in closed cooling water systems (Ref. 45). Positively; Hitachi-GE has selected a more corrosion resistant material for the RCW/HECW surge tank. I consider this is an appropriate change to make to account for the increased 60 year design life of UK ABWR. The HECW evaporator and condenser tubes use copper. This is one of a very limited number of places in the UK ABWR design where copper alloys are applied. I am satisfied the use of copper in this application is appropriate as there is no direct contact with the RCS, and copper alloys are widely used in the same applications for other plants (Ref. 45).

4.5.3 Radioactivity and Impurity Control

802. A combination of Ref. 19, *Topic Report on Impurity Ingress* (Ref. 30), *Basis of Safety Cases on Reactor Coolant Pressure Boundary* (Ref. 128) and *Basis of Safety Cases on Core Cooling Systems Leaking Detection System* (Ref. 128) present the following information relevant to impurity ingress and chemistry control:

- Seawater ingress from the Reactor Building Service Water System (RSW) in to the RCW is mitigated by design. The RSW operates at a lower pressure than the RCW.
- The Process Radiation Monitoring (PRM) system has monitors on RCW coolant return lines that can detect the ingress of radioactivity from RCS systems (i.e. RHR, CUW and FPC systems) into the RCW (which operates at a lower pressure than the RCS).
- RHR and CUW systems are fitted with isolation valves (the RCW provides cooling water to RHR and CUW system heat exchangers) which can be closed on detection of RCW in-leakage.
- The RCW surge tank is fitted with a high-level alarm.
- The RCW is fitted with process monitors.
- The CCW system can be drained to the LWMS if clean-up is required.

803. Hitachi-GE argue that on the basis of these engineered measures and the RCW surveillance requirements completed once every 24 hours to demonstrate compliance with the Generic Technical Specification (Ref. 90), the UK ABWR design gives adequate consideration to the potential for radioactivity ingress into the RCW. On this basis, I am satisfied with the arguments put forward by Hitachi-GE for GDA.

804. To be able to implement adequate chemistry control in the UK ABWR CCW system, the generic safety case identifies several ORs. These are listed in Annex 5 of my report. The ORs selected by Hitachi-GE are largely based on J-ABWR experience and authoritative sources of industry guidance on chemistry control in closed cooling water systems (Ref. 45). I am largely satisfied with the majority of the ORs suggested by Hitachi-GE during GDA for CCW systems. There are also several proposed improvements, for example, identifying chloride as a control parameter. These are clearly aligned with the requirements for the operating chemistry identified above.

However, there are two important gaps I have noted in the ORs presented during GDA:

- Direct conductivity is identified as a control parameter, but has no limit value. Furthermore, the *Water Quality Specification* (Ref. 20) does not provide an expected value.
- The absence of any radiochemical parameters, either as control or diagnostic parameters.

805. Although I regard selecting and justifying the final choices of chemistry regimes for the UK ABWR CCW system, including deriving and justifying the necessary ORs, to be a matter of normal business, the fact that these parameters are missing from the ORs presented in the generic safety case, falls outside of that work. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

4.5.4 Component Cooling Water System Sampling Provisions

806. In terms of sampling the RCW, Ref. 19 states this is performed to control the dosing of corrosion inhibitors to the system and to detect the ingress of radioactivity from the RCS. Ref. 19 identifies the following locations in the RCW where grab sampling can be undertaken:

- There are a total of four grab sampling points per division. One is located on the common header line which feeds the RCW parallel trains of heat exchangers.
- The other three grab sampling points are located one per RCW heat exchanger, at the outlet side. This is where the RCW interfaces with the seawater cooled RSW system. See Figure 37 above. Grab sampling points are marked "GSP".

807. For the HECW system, Ref. 19) explains there a total of two grab sampling points per division, one per cooling loop. Each sampling point is located downstream of the chiller coils, but upstream of the chiller units. Both are located downstream of the chilled water pumps.

808. Overall, Hitachi-GE argues selecting these locations means representative grab sampling of the RCW and HECW can be achieved. On this basis, I agree with the arguments put forward by Hitachi-GE. However, I note Ref. 19 lacks clarity on whether these grab sampling points are also used to sample for radioactivity which may have ingressed into these systems. This not undermine the generic design nor safety case, so I am satisfied this aspect should be progressed as part of normal business by a future licensee as they develop their choices for, and plans to implement CCW system operating chemistry choices.

4.5.5 Component Cooling Water Systems Summary

809. Overall, I am satisfied that for GDA, Hitachi-GE have provided sufficient evidence to justify the chemistry claims made on the UK ABWR CCW system. With the exception of some gaps identified in the ORs provided (captured by Assessment Finding, **AF-ABWR-RC-02**), I am satisfied the generic safety case presents sufficient information for a future licensee to be able to begin to make informed decisions regarding the final operating chemistry regime choice(s) to implement for the UK ABWR CCW system. I am satisfied Hitachi-GE have demonstrated the generic design provides enough flexibility not to foreclose these options to a future licensee, and that with the exception of responding to the relevant part of Assessment Finding **AF-ABWR-RC-02**, I consider the majority of the future work a licensee will need to perform in this area is a matter of normal business.

810. On this basis, in the context of generic safety case, I am satisfied that the expectations of SAPs ECH.1, ECH.3, ECH.4, EAD.1 and EAD.2 have been broadly met.

4.6 Radioactive Waste Treatment Systems

811. The UK ABWR possesses a number of systems designed to manage radioactive waste to support safe reactor operations. As there are no formal operating chemistry requirements for these systems, the approach taken for my assessment in this part of my report is different. My primary objective in assessing these systems is to make a judgement on the adequacy of their design, with respect to any specific claims made on the chemical performance required. This means the scope of my assessment of these systems is relatively narrow. A much more detailed assessment of these systems is presented in ONR's radioactive waste management assessment report (Ref. 119).
812. It is also important to note there were key differences in the level of design information presented by Hitachi-GE during GDA for the liquid and gaseous radioactive waste treatment systems. The liquid radioactive waste treatment system could be considered preliminary, because it is based on the J-ABWR design. However, for UK ABWR, during GDA, Hitachi-GE approached this system as a concept design, because the design of the structure housing it has not yet been defined. On the other hand, the gaseous radioactive waste treatment system is considered to be a preliminary design and was therefore much more developed for GDA. The breadth and depth of my assessment of these systems corresponds to this key difference. It is therefore much more detailed for the gaseous radioactive waste treatment system. I consider each of these systems separately below.

4.6.1 Liquid Radioactive Waste

813. The principal system in the UK ABWR design which manages liquid radwaste arisings is the Liquid Waste Management System (LWMS). The LWMS receives and treats radioactive liquid effluent from a number of buildings. The LWMS is divided into five main sub-systems, based on effluent source and means of treatment. These include:
- Low Chemical Impurities Waste System (LCW). This treats low conductivity waste from reactor building floor drains and the SAM system.
 - High Chemical Impurities Waste System (HCW). This treats high conductivity waste primarily from the CD bottom drain, the chemical analysis laboratory and the SAM system.
 - Spent Resin and Sludge system. This treats spent resin and sludge secondary wastes arising from the LCW and HCW.
 - Controlled Area Drains system. This collects effluent from non-radioactive systems in radiation controlled areas of the R/B and T/B, feeding in to the HCW for treatment on occasions when environmental discharge requirements aren't met.
 - Laundry Drains system. This processes effluent from the laundry facility and hot shower drains.
814. Hitachi-GE's overall case for the UK ABWR LWMS is presented in a number of documents at various levels within their hierarchy of safety case documentation. Chapter 18 of the PCSR (Ref. 129) is the principal top-level document, but there are links to other PCSR chapters, including Chapter 23 (Ref. 11). Chapter 23 of the PCSR (Ref. 11) does not introduce any new claims on the liquid radwaste systems. It represents the following two claims made in Ref. 129 and explains which chemistry claims they are directly linked to:

- “[LWMS SFC 4-12.3] *The LWMS facilities shall be designed to ensure that doses to both the workers and the public from normal operation of the UK ABWR LWMS are ALARP.*”
 - “[LWMS SFC 5-9.1] *The LCW and HCW systems treated effluent shall meet the re-use criteria specified in the Water Quality Specification.*”
815. I am satisfied the claims presented are reasonable. The linkage provided between PCSR chapters here is encouraging. Considering claim [LWMS SFC 5-9.1], of the five sub-systems which comprise the LWMS, for GDA I targeted my assessment towards the LCW and HCW.
816. Beneath the PCSR there are also a number of more detailed documents:
- *Basis of Safety Case on Liquid Waste Processing in the Radioactive Waste Building* (Ref. 130);
 - *Topic Report on ALARP Assessment for the UK ABWR LWMS and WILW/WLLW Systems* (Ref. 131);
 - *Liquid Waste Management System Process Description* (Ref. 132);
 - *Liquid Waste Management System Design Description* (Ref. 133)
 - *Material Selection Report for Radioactive Waste Systems* (Ref. 55)
817. My assessment largely focussed on *Topic Report on ALARP Assessment for the UK ABWR LWMS and WILW/WLLW Systems* (Ref. 131) and *Liquid Waste Management System Process Description* (Ref. 132). Part of Hitachi-GE’s effluent management strategy for UK ABWR is to recycle liquid effluent treated by the LCW and HCW for re-use in the RCS. This means the chemical performance of these systems needs to be able to meet the requirements of the *Water Quality Specification* (Ref. 20), to be able to produce effluent which meets reactor water standards. The approach to the design and operating philosophy for the LCW and HCW are therefore similar, so I have structured this sub-section on liquid radioactive waste, to present my assessment of these sub-systems together.

LCW Description

818. The LCW receives effluent from the R/B floor and effluent drains considered to contain low levels of chemical impurities. LCW effluent sources include the RCS, the FPC, plant make-up water system and the SAM system. The system comprises the following main treatment stages and equipment:
- Four effluent collection tanks. Two are used during normal operations and all four are used during outages.
 - Two parallel 50% duty hollow fibre filters to remove insoluble material.
 - Two parallel 50 % duty mixed bed ion-exchange demineralisers, with a downstream strainer.
 - Two sampling tanks. Each are equipped with level indicators and alarms and pumps to enable re-circulation of tank contents.
819. The overall design basis and operational philosophy for the LCW design provided during GDA is as follows:
- A batch process arranged in two parallel trains. The process typically operates once every two days during normal (reactor at-power) operations.
 - Installed capacity is designed to be capable of treating 30 m³/h at a constant flow rate when the batch process operates.
 - Collection tanks can be recirculated. Level control instruments prevent further effluent receipt until discharge.
 - Discharge of the sample tank to the CST is initiated by operators on receipt of adequate sample results (as defined in the *Water Quality Specification* (Ref.

20). Out of specification effluent is re-circulated back to the collection tanks to undergo further treatment.

HCW Description

820. The HCW is designed to process effluent with high levels of chemical impurities. The main sources of HCW effluent are the CPS CD bottom drains and the chemical analysis laboratory (“hot laboratory”) drains. LCW effluent which requires further treatment can also be sent to the HCW for processing. The system comprises the following main treatment stages and equipment:

- Two effluent collection tanks.
- An evaporator which is used to concentrate effluent in order to remove Total Organic Carbon (TOC). This consists of: the evaporator itself, an evaporator demister, an evaporator heater and an evaporator condenser.
- A Concentrated Waste Tank, where evaporator concentrate is routed.
- A Distilled Water Tank, where evaporator distillate is routed.
- A demineraliser which operates at 100 % capacity if deemed necessary from sampling the HCW Distilled Water Tank. Conductivity monitors are installed at inlet and outlet to monitor performance.
- Two sample tanks, each equipped with level indicators and alarms and pumps to enable re-circulation of the tank contents.

821. The overall design basis and operational philosophy for the HCW design provided during GDA is as follows:

- The HCW operates as a batch process with a constant batch volume. The operating frequency for the HCW is anticipated to be low during normal operations, running once every eight or nine months.
- Collection tanks may be recirculated and sampled. They are designed with the ability to chemically condition the effluent using either acid or base to neutralise effluent before it is transferred to the HCW evaporator. pH ranges between 6 and 10 are anticipated.
- Evaporator concentrate is periodically sampled to determine the concentration of suspended solids and “aggressive” ions. Concentrate is discharged from the evaporator by gravity.
- Evaporator distillate can be recirculated to the LCW collection tanks for demineralisation if the HCW demineraliser is unavailable.
- A chemical corrosion inhibitor is added to the evaporator.
- Discharge of the sample tank to the CST is initiated by operators on receipt of adequate sample results (as defined in the *Water Quality Specification* (Ref. 20). Out of specification effluent is re-circulated back to the collection tanks to undergo further treatment.

4.6.1.1 Chemistry Related Aspects

822. The basis of Hitachi-GE’s case to demonstrate the concept designs of the LCW and HCW are able to process effluents to meet the requirements of the WQS (claim [LWMS SFC 5-9.1]), is to provide OPEX from J-ABWRs for pre- and post-treated effluents (RQ-ABWR-1043 (Ref. 49). Given the design of the LWMS presented during GDA is at the concept stage, I consider this is a reasonable approach to take for this stage of the project. Furthermore, considering the information presented above on the LCW and HCW, I have noted the following positive aspects presented in Hitachi-GE’s concept designs for these systems:

- The flexibility to chemically condition HCW effluent. However, I note that the justification of treatment options for this system is the subject of an Assessment

Finding raised in ONR's radioactive waste management assessment report Ref. 119.

- The consideration of grab sampling source locations (RQ-ABWR-1220 (Ref. 49) and how the LCW and HCW may operate to achieve representative sampling.
- Evidence (Ref. 110) which demonstrates the potential for hydrogen accumulation and combustion in the UK ABWR liquid LWMS has been considered.

823. These design features and the OPEX presented by Hitachi-GE demonstrate the concept designs of the LCW and HCW have flexibility, when it comes to processing liquid effluent. The main chemical treatment processes I would expect to see, i.e. demineralisation by ion exchange to remove soluble species, filtration to remove insoluble species and evaporation, are present. Despite this, even at the concept design stage, there are important aspects required to justify the design of these systems which I consider are either missing or not clear. From a chemistry perspective, some of the gaps I have identified during GDA include:

- A lack of clarity on the chemical and radiochemical characteristics of the effluent (RQ-ABWR-1353 (Ref. 49)) and how this is linked to, and directly informs the design basis of the HCW and LCW.
- Why the installed clean-up capacity for the LCW and HCW concept designs is adequate to avoid/minimise the need to continuously re-circulate effluent to meet requirements in the *Water Quality Specification* (Ref. 20). Including the impact this has on secondary wastes, worker doses etc. It isn't clear how frequently out of specification (at the end of the treatment process) effluent will be encountered, what this is most commonly triggered by (i.e. high TOC etc.) and how the design of the LCW and HCW has accounted for this.
- Linked to the above two points, a lack of clarity on ORs. It isn't clear what the "limits" are for each system in terms of the chemical and radiochemical specification of effluent which each can handle.
- Linked to the above points, a justification of the decontamination factors which each stage of chemical clean-up needs to achieve, and what can be achieved in practice.
- Sampling end points (i.e. sample sinks, enclosures, line lengths etc.). This is particularly important when assessing the design against claim [LWMS SFC 4-12.3]. There also appears to be a reliance on administrative controls to prevent re-introduction of out of specification effluent to the CST and therefore RCW system.
- Why there appear to be important design differences between the LCW and HCW for the same installed chemical treatment process.

824. Considering Hitachi-GE is at concept design stage with these systems, there is time for a future licensee to address these gaps and optimise the design. Furthermore, because the scope of the chemistry interest in these systems is relatively narrow, I do not consider it is appropriate to raise an Assessment Finding to address these specific gaps, at this time. Given ONR's radioactive waste management assessment report takes a much more detailed interest in the design and justification of thereof, of these systems, I note here that some of the detailed points I have raised above may relate to Assessment Findings raised in that topic.

4.6.1.2 Liquid Radioactive Waste Systems Summary

825. Considering the design of the LCW and HCW is at the concept stage, undertaking a meaningful assessment of these systems during GDA has been challenging. Nevertheless, the claims presented are what I would broadly expect to see. For chemistry, I therefore consider SAP ECH.1 is partially satisfied. This is because I am satisfied the LCW and HCW appear to include the main chemical treatment processes

I would expect to see. Furthermore, Hitachi-GE has provided OPEX for J-ABWRs to demonstrate systems of a similar design are capable of cleaning-up the radioactive liquid effluents generated during reactor operations.

826. Even when taking into account the fact that the designs are at concept stage, considering Hitachi-GE's safety case claims, important information is missing, which means I consider it isn't possible to draw more detailed and meaningful conclusions against the expectations of SAPs ECH.1, ECH.3 and ECH.4. This is because there are important aspects required to justify the chemical performance of the design of these systems which I consider are either missing, or not clear. Considering the relatively narrow scope of the chemistry interest in these systems and the fact that there is time for a future licensee to address these gaps and optimise the design, because it is currently at the concept stage, I am satisfied a chemistry specific Assessment Finding to address these gaps is not proportionate. However, I note that ONR's radioactive waste management assessment report (Ref. 119) takes a much more detailed interest in the design and justification of thereof, for these systems.

4.6.2 Gaseous Radioactive Waste

827. The UK ABWR OG system extracts non-condensable gases (including hydrogen and the fission product noble gases) from the main condenser. It treats extracted stream first by dilution with further steam, and then by removing the hydrogen using a catalytic recombiner. The OG system then removes (by hold-up) the fission products from the gas stream using a series of temperature controlled charcoal adsorber beds. The treated gas, which by this point contains only negligible amounts of hydrogen and fission products, is then filtered for a final time and directed to the stack for discharge to the environment. An outline of the UK ABWR OG system is provided in Figure 38 below.

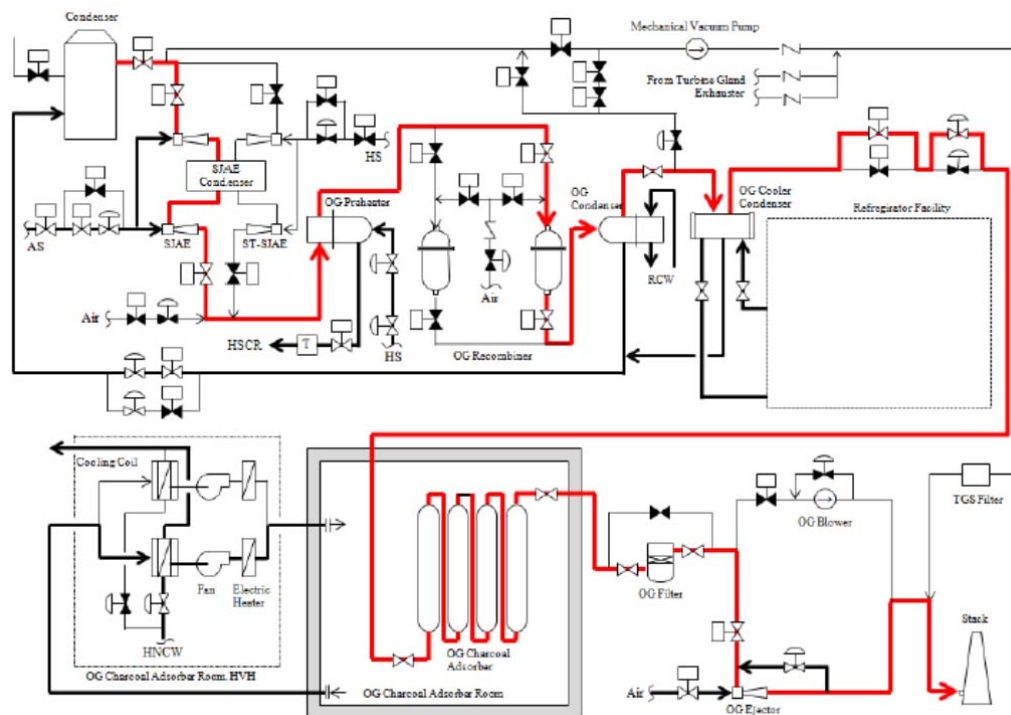


Figure 38: Outline of the UK ABWR OG System (Ref. 134)

828. Considering the design intent of the OG system, the UK ABWR generic safety case makes a number of important claims on the chemical performance of the system, both in terms of its ability to remove flammable gases and its efficiency in removing

radioactivity. Assessing the claimed performance and the design of the system therefore formed an important part of my overall chemistry assessment throughout GDA. My initial assessment (RQ-ABWR-0081, -0781, -0782 and -0783 (Ref. 49)) of Hitachi-GE's earlier GDA submissions to justify the design of the OG system revealed a number of inconsistencies and key areas, which from a chemistry perspective, had not been adequately justified to be able to demonstrate relevant risks are reduced SFAIRP. Furthermore, there were other important factors for the OG system:

- The OG system is purposefully designed to accumulate large amounts of radioactivity.
- The OG system is located outside of the main reactor containment and therefore represents a potential source of increased radioactivity which is available for release to the environment should the system fail to operate as intended, or during fault or accident conditions.
- The OG system processes significant quantities of hydrogen during normal operations.

829. At that stage in GDA, I considered Hitachi-GE had not adequately justified a number of key options and design features which other BWR OG systems implement. In collaboration with the Environment Agency and other ONR technical disciplines, I therefore raised RO-ABWR-0073 (Ref. 25). The RO asked Hitachi-GE to provide a robust demonstration that the design of the UK ABWR OG system reduces relevant risks SFAIRP. The principal SAPs relevant to my assessment of the chemistry aspects of Hitachi-GE's overall final submissions to justify the design of the OG system included: ECH.1, ECH.3, ECH.4, EAD.1 and EAD.2.

830. Chapter 18 of the PCSR (Ref. 129) provides an overview of the UK ABWR OG system and links to other PCSR chapters, including Chapter 23 (Ref. 11). Ref. 129 presents the following safety functional claims which are relevant to chemistry:

- “[OG SFC 4-7.1] *The OG minimises the dose to worker during normal operating conditions.*”
- “[OG SFC 4-11.1] *The OG minimises the release of radioactivity to the environment during the start-up, power and shutdown operations.*”
- “[OG SFC 4-11.2] *The OG reduces the risk of hydrogen combustion arising from the reaction of radiolytic hydrogen produced in the reactor.*”

831. Chapter 23 of the PCSR (Ref. 11) does not present any further specific and/or unique claims on the OG system. Instead, it provides a link to some of the higher-level claims already made in the chemistry area, which are equally applicable to the OG system. These includes claims [RC SC5], [RC SC6], [RC SC7] and [RC SC18]. This represents a good example of how the UK ABWR generic PCSR is generally very good at providing the links to other chapters where relevant supporting information is presented. In terms of the overall suitability of the claims presented, they broadly represent the topics I would expect to see, covering the chemical performance of the OG system with respect to radioactivity and also managing hydrogen. Again, the claims are very high-level though. For example, claim [RC SC18.1] refers to “*system design*”, but does not explicitly refer to the OG system.

832. Beneath the PCSR there are a number of more detailed documents which provide supporting evidence for the claims made on the OG system and to justify the design. The principal documents are:

- *Off-gas System Basis of Safety Case* (Ref. 134);
- *Topic Report on ALARP Assessment for OG System* (Ref. 135);
- *Technical Supporting Document on the OG ALARP Report* (Ref. 136); and
- *Off-gas System Design Description* (Ref. 137).

833. These are the documents I have undertaken a detailed assessment of below. The main focus of my assessment during GDA was the information contained in Ref. 135 and 136.

4.6.2.1 Management of Flammable Gases

834. On the topic of flammable gas management, the summary information contained in Chapter 23 of the PCSR (Ref. 11) states:
- Steam dilution by the SJAE reduces the risk of hydrogen combustion up until just before the recombiner by producing inerted conditions.
 - The OG system provides parallel redundant (two x 100%) catalytic recombiners, with one duty and one in standby, which provides mitigation in scenarios where recombiner failure, or loss of efficiency may occur.
 - The potential poisoning of the catalytic recombiner by a number of substances has been recognised and the use of known detrimental substances has been minimised.
 - The OG system has hydrogen detection monitors after the second stage SJAE and after the OG cooler condenser (after the recombiner). These detect high hydrogen concentrations which would occur as a result of degraded system performance. Compared to the J-ABWR reference plant, the UK ABWR has been modified to include an automatic isolation function (by closing the OG system isolation valves) in response to increased hydrogen concentrations after the OG cooler condenser (Ref. 135).
835. A much more detailed assessment of this information and the overall capability of the OG system to safely manage flammable gases (principally hydrogen) was undertaken by one of ONR's chemical engineering specialist inspectors during GDA. This includes an assessment of the claims made on the chemical performance of the OG recombiner. The outcome of this assessment is reported in ONR's radioactive waste management assessment report (Ref. 119). There are several key points that need should be highlighted here though, as part of my chemistry assessment:
- Certain parts of the OG system operate above the LFL for hydrogen during normal operations; and
 - The OG catalytic recombiner is the only safety measure capable of directly removing the significant quantities of hydrogen generated under normal operations. It therefore makes a significant contribution to delivering several nuclear safety functions.
836. Considering the above two points, with respect to the safe management of flammable gases, there are some key observations I have made during my assessment of the chemistry aspects of the OG system. My aim is not to replicate here the assessment reported in Ref. 119, but to highlight important matters I have noted during my own assessment, which relate to this topic. These matters are captured by a specific, single Assessment Finding raised in Ref. 119. They include:
- Hitachi-GE's safety categorisation and classification of the OG system recombiner is C3. While Hitachi-GE has provided a suitable justification for this during GDA, I would expect the future licensee to give further consideration to the reasonably practicability and benefits of increasing it.
 - The evidence provided to substantiate the performance of the recombiner (predominately in (Refs. 134, 135 and 136)) is largely theoretical. The recombiner is required to operate at a very high efficiency to meet the design basis (Ref. 119). Similar recombiners have been operated in the same system for J-ABWR. There should therefore be a much broader range of available evidence to adequately substantiate the performance of the recombiner. In the context of a generic safety case, while I am content with the information

provided at this stage, I consider that this aspect should be strengthened as the UK ABWR safety case is progressed by the future licensee.

- Similarly, considering the introduction of the OG system automatic isolation function in the UK ABWR generic design, the justification for the two x 100% parallel recombiner arrangement, and how and when the standby recombiner would be called on before a system trip is initiated, is not clear. The only real justification provided for the two x 100% parallel arrangement is that it is needed to ensure sufficient recombination capacity during start-up operations, when the thermal efficiency of the recombiners is lower, owing to the lower flow rates of hydrogen being processed.
- Considering the very high efficiency required of the recombiners, positively, the generic safety case does recognise the importance of implementing suitable measures to avoid introducing any “detrimental substances” which would degrade their performance. This is important given the OPEX shows reduced recombiner performance has been a relatively common occurrence. Furthermore, this is all the more important given the UK ABWR has introduced the automatic trip function when elevated hydrogen concentrations are detected. Capturing these important requirements, for example, when developing foreign material exclusion programmes and other associated operating instructions, should be progressed by the future licensee.
- As explained earlier in my report, by introducing the HOIS, Hitachi-GE has also recognised the need to optimise this aspect of the design to be able to implement HWC. This provides some additional confidence that the oxygen injection function into the OG system which the HOIS performs should secure the recombiner performance and efficiency required.

837. Finally, considering the very narrow scope of my assessment of the adequacy of the OG system to manage flammable gases, there are other significant aspects of the design which I have not assessed, but were within the scope of RO-ABWR-0073 (Ref. 25). This includes the ability of the OG system to be able to deal with the consequences of a potential combustion event. In addition, parts of the OG system are known to operate above the LFL. Hitachi-GE's approach to the design is essentially one of risk mitigation, as opposed to attempting to further “eliminate” such conditions. I have not assessed Hitachi-GE's justification for why this approach is reasonable and technically justified, nor why there are no other reasonably practicable measures that can be taken to further reduce the risk. These have been assessed by ONR's internal hazards specialist, and are presented in Ref. 114.

838. Overall, considering the conclusions reached in Ref. 119 by ONR's chemical engineering specialist, in parallel with the output of the limited scope of my assessment presented above, for a generic safety case, I am satisfied Hitachi-GE has presented suitable and sufficient evidence to substantiate the claims made on the chemical performance of the OG system recombiner. I am therefore satisfied that the broad expectations of SAP ECH.1, in this context, have been met. As I have identified above, I consider there is further work a future licensee needs to undertake to be able to develop these aspects of the generic safety case towards a more operations-focused approach.

4.6.2.2 Radioactivity Hold-up

839. With respect to the design of the OG system and the ability to hold-up radioactivity, the summary information contained in Chapter 23 of the PCSR (Ref. 11) states:

- The significant contributors to the source of radioactivity in the OG system are krypton and xenon noble gas radioisotopes.
- The majority of this inventory is adsorbed on the first of four charcoal adsorber beds.

- The charcoal adsorbers hold-up xenon and krypton isotopes for a minimum of 30 days and 40 hours, respectively.
 - Iodine is mainly retained in the RPV and if any is carried over with steam it will then be retained by the OG condenser. Any iodine that isn't removed by the OG condenser or OG cooler condenser will be removed by the first charcoal adsorber, where it will then undergo radioactive decay.
 - The charcoal adsorber design provides a conservative margin which accommodates deviations in operating conditions (temperature, pressure, humidity).
840. The majority of Hitachi-GE's evidence to support these implicit claims and assertions is presented in *Technical Supporting Document on the OG ALARP Report* (Ref. 136); The evidence and technical information contained in this report is drawn upon directly by *Topic Report on ALARP Assessment for OG System* (Ref. 135). As a document suite, I consider Ref. 135 and Ref. 136 are very well presented. The basis of Hitachi-GE's safety arguments is clearly articulated and there are very clear links to appropriate types of evidence. The scope of Ref. 135 is very comprehensive and covers the majority of technical topics I expect to see. Hitachi-GE worked hard on this aspect of the generic safety case during GDA. When judged against ONR's guidance on the demonstration of ALARP (Ref. 4), I consider this to be a good example of meeting the relevant expectations described in Ref. 4.
841. Considering claims [OG SFC 4-7.1] and [OG SFC 4-11.1], and the general information described above and presented in Chapter 23 of the PCSR (Ref. 11), my assessment of Ref. 135 and 136 focussed on Hitachi-GE's justification for the adsorber bed design, configuration and charcoal capacity. The UK ABWR OG system has four charcoal adsorber beds arranged in series. Each bed (vessel) contains 18 Te of activated charcoal, giving a total mass of 72 Te of charcoal evenly distributed across each of the four beds. The activated charcoal is very porous and a dynamic, continuous exchange process is established between the charcoal and the gas flow, whereby gas molecules are held-up as they pass through the beds. This allows the gaseous radioisotopes time to undergo radioactive decay. As this is a physical and dynamic process, the charcoal isn't consumed by the process and is therefore designed to last the full-lifetime of the plant (60 years) without being replaced. The use of charcoal adsorber beds in LWR OG systems is common practice, and what I regard as relevant good practice.
842. For these applications, the total mass of charcoal required is generally determined by selecting a desired target hold-up time for the radioisotopes of interest. Hold-up time increases with charcoal mass and as process gas flow rates become lower. For the UK ABWR OG system, target hold-up times for xenon and krypton selected by Hitachi-GE are 30 days and 40 hours, respectively. These are determined based on discharge requirements at the plant stack. This hold-up time is commonly referred to instead by the dynamic adsorption coefficient (or K-value). K-values are affected by a number of conditions including:
- the type of activated charcoal selected;
 - lower temperatures result in larger K-values;
 - lower pressures result in larger K-values; and
 - increases in moisture or relative humidity, decrease the K-values.
843. To substantiate the chemical performance of the UK ABWR OG system, in Ref. 136, Hitachi-GE present several sensitivity and modelling-based calculations, to demonstrate they have introduced a number of conservatisms to underpin their determination of the mass of charcoal required in the adsorbers. These conservatisms cover factors like: assuming a 20% increase in the relative humidity of the off-gas, decreasing the K-value to account for variabilities in charcoal procurement or measurement, and assuming the maximum system flow rate is almost double that expected. The results of these sensitivity calculations clearly show that the design

includes a significant margin. Hitachi-GE's modelling shows only around half of the total mass (of 72 Te) of charcoal is required to achieve the desired hold-up times, assuming a realistic relative humidity. Furthermore, in Ref. 135, Hitachi-GE also provides evidence in the form of measured data from site preparation tests for the KK-6 J-ABWR reference plant. For a charcoal mass of 72 Te these data show a measured krypton hold-up time of [REDACTED] as compared to the 40 hours required.

844. To take advantage of the fact that K-values increase with decreasing temperature, some BWRs operate their charcoal adsorbers at lower temperatures (around 0°C). This means less charcoal is required to achieve the same target hold-up times. To account for the importance of this relationship between temperature and hold-up time, the UK ABWR OG system charcoal adsorbers are located in a temperature controlled cell, which operates at 25°C. This is positive, however, during GDA I also asked Hitachi-GE (RQ-ABWR-1412 (Ref. 49) why it wasn't desirable to operate the charcoal beds at lower temperatures. Hitachi-GE argues that an additional gas cooler would be required and this introduces more active systems, lowering the overall fault tolerance of the system. On this basis I accept Hitachi-GE's case for the operating temperature selected for the OG system adsorber beds.
845. As described earlier, Chapter 23 of the PCSR (Ref. 11) also contains implicit claims regarding the behaviour of iodine in the OG system. The impact of iodine is considered By Hitachi-GE in Ref. 87. This report contains the necessary evidence to support Hitachi-GE's claims regarding iodine behaviour in UK ABWR. While I am not necessarily convinced by all of Hitachi-GE's assumptions, I am content with the most important conclusion, that the impact of iodine on the OG system is negligible.
846. Overall, I am satisfied that Hitachi-GE has provided suitable and sufficient evidence to justify the mass of charcoal required in the UK ABWR OG system. The evidence provided supports the relevant claims made in the PCSR (Ref. 11) regarding the radionuclide hold-up times. Hitachi-GE's evidence also clearly demonstrates that the majority of the radioactivity contained in the UK ABWR OG system will be adsorbed on to the first charcoal bed (RQ-ABWR-1412 (Ref. 49)), which contains around 67% of the entire system inventory. This is relevant to my assessment of the UK ABWR OG system charcoal bed configuration, presented in Section 4.6.2.3, below.

4.6.2.3 Charcoal Adsorber Bed Configuration

847. As previously explained, the UK ABWR OG system charcoal adsorbers are arranged in series. In Ref. 136, Hitachi-GE also examines the impact of charcoal adsorber configuration. This is mainly driven by the fact that the majority of radioactivity contained within the OG system is adsorbed onto the first bed (67%, as noted above). Furthermore, OPEX shows other BWRs have used different charcoal adsorber arrangements for the OG system. An important point to note here is that the OG system design used in the US ABWR includes both a "guard bed" and parallel main charcoal beds. In this regard, in Ref. 136, two main potential options are considered by Hitachi-GE for the UK ABWR OG system. The first is the potential to include a "guard bed" before the main charcoal adsorber beds. The second is the potential to adopt a parallel arrangement of the main charcoal adsorber beds.
848. Hitachi-GE's consideration of the potential benefits and detriments associated with arranging the charcoal adsorbers in parallel concludes it is appropriate and reasonable to retain the series arrangement. Hitachi-GE's (marginally) highest scoring option though is to retain the series arrangement, but incorporate a "guard bed". Despite the optioneering identifying this as the preferred option, incorporating a "guard bed" into the UK ABWR generic design is not taken forward by Hitachi-GE for GDA. The conclusions of Ref. 135 are carefully worded to essentially say that the series charcoal adsorber bed arrangement either with or without a "guard bed", and including other reasonably practicable design improvements identified for the UK ABWR OG system

design (i.e. automatic isolation, as described above), are both “*equally ALARP options*”. Hitachi-GE argues a “guard bed” provides no nuclear safety improvements and the decision is therefore purely commercial and should be left for a future licensee to determine. The “guard bed” would be relatively small (several tonnes of charcoal) and would serve to protect the main charcoal adsorber beds from detrimental materials (such as moisture). My assessment which follows examines the rationale for Hitachi-GE’s conclusion in much more detail.

849. The basis of Hitachi-GE’s case to retain the main charcoal adsorber beds in series is that the design already provides an appreciable margin, in terms of charcoal mass, to account for any potential variation in process conditions. From Ref. 136, the two most important fault scenarios where this arrangement appears to have the biggest impact, from a chemistry perspective, are Hitachi-GE’s assumptions underpinning the analysis of a charcoal bed fire, and moisture ingress into the charcoal beds as a result of a fault within the OG system. In both cases, assumptions are made about the impact this will have on K-values, and for the postulated fire fault, the potential for fire propagation. All of this work is underpinned by modelling, using the same approach I described previously in Section 4.6.2.2. (radioactivity hold-up).
850. Considering the inherent uncertainty surrounding the likely behaviour of the OG system during a postulated combustion event, I consider the approach taken by Hitachi-GE is reasonable, in the context of demonstrating what impact charcoal adsorber bed arrangement, may have on the radiological consequences. One of Hitachi-GE’s most important arguments for dismissing parallel processing is that it would provide reduced downstream charcoal capacity (relative to a series arrangement) in the event of a charcoal bed fire. Given the fact that most (67%) of the radioactivity is contained on the first charcoal bed, I asked Hitachi-GE (RQ-ABWR-1412 (Ref. 49)) why this factor was judged to be more important than the benefit of potentially halving the inventory of radioactivity present on the first bed, by implementing parallel processing. Hitachi-GE explained that the public dose in such events would indeed be reduced (from 6.8 to 3.4 mSv), but this was also considered to be a very specific, and low frequency (10^{-4} to 10^{-5} per year) event. From a chemistry perspective, I consider the majority of the arguments presented during GDA in this regard, were satisfactory.
851. The one aspect of Hitachi-GE’s case justifying the arrangement of the charcoal beds which I do not consider was particularly robust, is their analysis which assumes that twice the charcoal mass is required (i.e. it assumes two off four beds of 18 Te) to implement parallel processing. This is because Hitachi-GE’s analysis assumes an all duty and standby arrangement is required. Parallel processing could also be achieved without increasing the total charcoal mass, for example, by operating with two off two beds, and half the flow through each parallel train. Hitachi-GE’s work in Ref. 136 does consider this option, however, their overarching arguments regarding gross disproportion, specifically focus on the parallel processing option of all duty and standby (which as explained in NS-TAST-GD-005, *Guidance on the Demonstration of ALARP* (Ref. 4), is what I would consider to be a “deluxe” option, in this context).
852. For the moisture ingress event, Ref. 136 performs a calculation based on an assumed eight hour time period for OG condenser and OG cooler consider failure. Hitachi-GE calculates this could lead to the ingress of 400 kg of moisture into the adsorber beds. I have not assessed the detail of the appropriateness of the assumed 400 kg of moisture, but Hitachi-GE claims this means only 6% of the available adsorption capacity of the charcoal beds would be compromised in such an event. They assert that given the UK ABWR OG system charcoal beds are significantly oversized in terms of charcoal mass, the downstream adsorbers have sufficient capacity to hold-up any noble gases which would either become desorbed from the first bed, or would have otherwise been removed from the gas flow by the first bed. This is also one of the main

reasons why Hitachi-GE claims the addition of a “guard bed” results in no significant nuclear safety advantages.

853. Considering the majority of the benefits of operating with parallel charcoal adsorber beds could also be achieved by adopting a “guard bed”, during GDA, I asked Hitachi-GE a series of questions (RQ-ABWR-1412 and -1487 (Ref. 49)), with the aim of establishing whether the UK ABWR generic design forecloses the potential to install a “guard bed” at a later date. Hitachi-GE provided several responses which considered the option of installing a “guard bed” sized at either one or six Te^{****}. Some of Hitachi-GE’s initial work to respond to my queries was inconsistent with their earlier overarching conclusion in Ref. 135, namely, that installing a “guard bed” was the preferred (marginally) option and reasonably practicable to implement. With respect to the “guard bed”, Hitachi-GE’s final position for GDA is that:

- Radioiodine makes the largest contribution (97%) to estimated radiological doses following a potential adsorber loss of containment fault. Including a “guard bed” therefore shifts the focus of radiological risk from the first main charcoal adsorber bed to the “guard bed”. Hitachi-GE concludes there is therefore no net nuclear safety benefit and it is grossly disproportionate to change the generic design of the UK ABWR OG system during GDA, to add a “guard bed”.
- Based on the assessment of 400 kg of moisture that could potentially ingress into the adsorbers, a smaller (one Te) “guard bed” would be sufficient and could be accommodated within the footprint of the UK ABWR turbine building. On this basis, Hitachi-GE concludes the UK ABWR generic design does not foreclose the option of including a “guard bed” at a later stage.

854. Based on the overall arguments and evidence provided by Hitachi-GE regarding the arrangement of charcoal adsorber beds in the UK ABWR OG system, I conclude that:

- I agree that including additional main charcoal beds to operate with a parallel arrangement is demonstrably grossly disproportionate. However, as explained above, I do not agree that this is the only way to achieve parallel processing, or that this option is representative of other means of achieving the same benefits.
- I consider that similar benefits to parallel processing can be achieved by including a “guard bed” in the design. I disagree with Hitachi-GE’s conclusion that adding a “guard bed” to the UK ABWR generic design provides no net nuclear safety benefit.

855. My rationale for these conclusions is as follows:

- I consider including a “guard bed” in BWR OG system design appears to be a relevant good practice, mainly because other BWRs have included such features. The design intent is not to replace the charcoal in the main adsorbers for the full lifetime of the plant. Furthermore, the UK ABWR design excludes any specific measures to enable the charcoal beds to be regenerated by drying in order to restore their hold-up capacity, if a moisture ingress event were to occur. In addition, “guard beds” protect the main adsorber beds from the effects of other deleterious impurities, not just moisture.
- Hitachi-GE’s arguments regarding the gross disproportion aspects of installing a “guard bed” are not robust. The US version of the same plant design (US ABWR) has already done this, as have other BWRs. Hitachi-GE’s arguments appear to be based on implementing a “deluxe option”, which has not been suitably sized nor optimised for their specific OG system design. Even if these

^{****} The six Te case is based on the “guard bed” installed capacity for the US ABWR, scaled to the equivalent requirements for UK ABWR. The one Te case is based on Hitachi-GE’s assumption of 400 Kg of moisture following OG condenser and cooler failure.

arguments were valid, Hitachi-GE still scored the “guard bed” as the highest scoring option during their optioneering exercise and concluded it reduces risks SFAIRP (and therefore by definition is reasonably practicable to implement).

- Hitachi-GE’s arguments regarding the importance of radioiodine are inconsistent. Their case for iodine behaviour under normal operations is that it remains in the aqueous phase and therefore follows the condensate. Very little, if any at all, should therefore be present in the gas stream processed by the OG system. This is why Hitachi-GE’s generic safety case, quite reasonably, does not place claims on the hold-up time the adsorbers provide for radioiodine. Despite this, the fact that the adsorbers will hold-up radioiodine (and contribute 97% of the radiological dose) is one of the main arguments used by Hitachi-GE to argue for the benefits of retaining a series adsorber arrangement. For the “guard bed”, these factors are being used to argue the case the other way. This does not appear to be appropriate.
- The smaller size of a “guard bed” would bring benefits in many of the OG system fault scenarios considered by Hitachi-GE during GDA, for example, in recovering from a moisture ingress event, or in the case of having to replace the charcoal. This would result in lower worker ORE and radioactive waste generation.

856. Although I don’t agree with Hitachi-GE’s assertions and I consider there are both nuclear safety and commercial considerations when deciding whether to include a “guard bed” in the UK ABWR design, I am satisfied with the position reached for GDA. This is because the information presented during GDA confirmed that despite the UK ABWR generic design excluding a “guard bed” from the OG system, the option of installing one at a later stage is not foreclosed either. Considering Hitachi-GE’s own conclusion that incorporating a “guard bed” in the design of the UK ABWR OG system reduces risks SFAIRP, and their view that the decision to implement a “guard bed” is purely commercial (Ref. 135), I expect a future licensee to give this further consideration. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-21: The UK ABWR generic safety case concludes including a “guard bed” in the off-gas system design reduces relevant risks SFAIRP (and by definition is therefore reasonably practicable to implement). Despite this conclusion, the decision to include a “guard bed” is left for a licensee to make, but evidence is provided to demonstrate the generic design does not foreclose this is an option. The licensee shall review and update the UK ABWR safety case to provide the necessary information to demonstrate a “guard bed”, or an equivalent measure which delivers the same function, has been implemented in the UK ABWR off-gas system design.

4.6.2.4 Materials Selection

857. Ref. 55 presents Hitachi-GE’s justification for the materials selected for UK ABWR OG system main piping. Hitachi-GE has applied their materials selection methodology on the basis of considering the operating environment for several main locations within the OG system and determined the main risks to integrity are general corrosion, FAC, SCC and thermal embrittlement. Ref. 55 also demonstrates Hitachi-GE has considered wider OPEX for previous materials degradation issues in the OG systems for Japanese BWRs. Applying their methodology; Hitachi-GE has selected a mixture of CS, LAS and SS for UK ABWR OG System main piping.
858. The large majority of the system uses CS. The part which uses LAS does so on the basis of an increased risk of FAC. SS is selected for main piping around the recombiner on the basis of the high operating temperature (425°C). The materials selection for UK ABWR OG system is the same as J-ABWR. I haven’t assessed the detail of Hitachi-GE’s application of their materials selection methodology to justify this decision. However, on the basis of the operating environments defined for the system

and the materials degradation risks identified, from a chemistry perspective, I am satisfied with the materials selected for the UK ABWR OG system main piping.

4.6.2.5 Gaseous Radioactive Waste Summary

859. During GDA, Hitachi-GE provided a clear and coherent suite of documentation to justify the design of the UK ABWR OG system. Based on the claims presented in Chapters 18 and 23 of the PCSR (Ref. 11 and 129), I am largely satisfied that suitable and sufficient evidence has been provided to support them.
860. My assessment of the OG system focussed on the suitability of the design to justify the claims made on the chemical performance of the system, with respect to its ability to remove radioactivity, and the arrangement of the charcoal beds. In this regard, from a chemistry perspective, I am largely satisfied with the information provided by Hitachi-GE. The one important exception is the exclusion of a “guard bed” from the UK ABWR OG system design. This has been captured as part of Assessment Finding, **AF-ABWR-RC-21**.
861. Assessing the adequacy of the OG system design to safely manage flammable gases has largely been excluded from the scope of my assessment. This is because it has been largely covered by other ONR technical disciplines. Nevertheless, there are very important and relevant conclusions reached by my internal hazards (Ref. 114) and chemical engineering (Ref. 119) colleagues regarding the adequacy of this aspect of the design of the UK ABWR OG system. Based on these conclusions, it is clear that there is still a significant amount of important work for a future licensee to undertake to justify the UK ABWR OG system can safely manage hydrogen.
862. Overall though, in the context of a generic safety case, I am largely satisfied the requirements of SAP ECH.1 have been met. However, given I have raised Assessment Finding, **AF-ABWR-RC-21**, I am not satisfied the expectations of SAP ECH.3 have been fully met at this time.

4.6.3 Radioactive Waste Treatment Systems Summary

863. From a chemistry perspective, assessing the adequacy of the UK ABWR radioactive waste treatments systems has been limited in scope, either due to the level of design information available during GDA, or because few chemistry-related claims were presented. Despite this, I am content that a meaningful assessment of these systems has been undertaken during GDA. The most important conclusions from my assessment are:
- While I have not been able to draw many conclusions against the relevant SAPs for the LWMS, I am satisfied that the generic design does include the types of chemical processes and claims I expect to see. Nevertheless, even considering the early stages of the design, I did identify several gaps. The conclusions of the narrow scope of my chemistry assessment performed for the LWMS should therefore be read in conjunction with ONR’s radioactive waste management assessment report (Ref. 119).
 - The importance of the OG system in the overall UK ABWR design is clear from the relevant documentation presented in the generic safety case. I am broadly satisfied that sufficient information has been provided to substantiate the main chemistry-related claims made by Hitachi-GE, regarding flammable gas management and the hold-up of radioactivity. The important caveat is that the scope of my assessment performed during GDA of the adequacy of the OG system design to safely manage flammable gases, was rather limited. The conclusions of my chemistry assessment of these aspects of the OG system design should therefore be read in parallel with ONR’s internal hazards (Ref. 114) and radioactive waste management (Ref. 119) assessment reports.

- There are, however, some aspects concerned with the generic design of the UK ABWR OG system and its ability to process radioactivity, which will require further enhanced scrutiny beyond GDA. In particular, the decision to exclude a “guard bed”, or other means to provide similar functionality. However, I am satisfied that this option has not been foreclosed by the UK ABWR generic design. This is the subject of Assessment Finding, **AF-ABWR-RC-21**.
- I am satisfied the chemistry-related safety case claims for both systems can be further substantiated as the generic design and safety case is taken forwards.

4.7 Accident Chemistry

864. The assessment of risks arising from nuclear facilities needs to consider those arising both from normal operation and from accident conditions. Conservative design, good operational practice, and adequate maintenance and testing should minimise the likelihood of accidents. Nuclear facilities are therefore designed to cope with, or are shown to withstand, a wide range of faults without unacceptable consequences by virtue of the facility’s inherent characteristics or safety measures.
865. BWRs have three successive barriers which prevent the release of radioactivity during a reactor accident; the fuel cladding, the RCS pressure boundary and the containment structures. In normal operation, the primary barrier is the cladding of the fuel itself, which retains over 99% of the nuclear material in the reactor. Failure of the fuel cladding releases radioactivity into the RCS, where it can be released to the containment structures if this barrier also fails. Ultimately, failure of the entirety of these barriers would lead to the release of radioactivity to the environment. There are no generic statements that can be made about the quantity and levels of radioactivity released during an accident as it depends heavily on the precise sequence and conditions. However, most accidents within the design basis of the plant do not result in damage to the fuel, but may result in some limited radioactive release from the plant. In the most extreme, but unlikely cases, a Severe Accident (SA), by definition, will lead to large amounts of fuel damage and the potential for releases outside the plant.
866. Depending upon the accident sequence, other releases may occur from the reactor including large quantities of steam or flammable gases. As with radioactivity, they may be released into the containment, and the plant must also be able to deal with the hazards caused by them, such as increased pressures or combustion. These represent risks to maintaining the integrity of the containment structure and therefore the final barrier to release of radioactivity.
867. As well as the reactor, there are other significant sources of radioactivity within the plant. They include the spent fuel stored in the SFP and radioactivity contained in radioactive waste treatment systems. Accidents in these systems also need to be analysed, to ensure the protection in place is suitably robust.
868. The quantities of releases, their subsequent behaviour and effectiveness of many mitigation measures may depend upon the chemical properties and assumptions used within the analysis. This part of my assessment report therefore considers those assumptions, as an input to the overall ONR assessment of the safety analysis provided for the UK ABWR generic design.
869. Consistent with international practice, the safety features incorporated in the UK ABWR generic design are based on a defence-in-depth approach. This means that there are multiple layers of protection, all designed to ensure safety, independently of one another. Collectively, these layers of protection are in place to ensure that the plant can be safely shutdown following a fault or accident. To do so there are three main steps:

- To effectively shut down the reactor and ensure the nuclear chain reaction remains inhibited.
- To initially cool the reactor, and maintain cooling of the fuel to remove decay heat, and;
- To ensure radioactivity remains contained and is not released to the environment.

870. The UK ABWR generic design includes a number of engineered safety systems which provide these functions, and which are independent to the reactor systems used during normal operations. In this regard, I assessed the chemistry aspects of the following systems during GDA:

- the Chloride Ingress Protection System (CIPS);
- the Filtered Containment Venting System (FCVS);
- the Suppression Pool pH Control System;
- the Standby Gas Treatment System (SGTS);
- the Standby Liquid Control (SLC) system;
- the Flooder System of Specific Safety Facility (FLSS);
- the Flammability Control System (FCS); and
- the Passive Autocatalytic Recombiners (PARs) in the Secondary Containment.

871. The adequacy of these systems is considered much more broadly within the ONR Faults studies and Severe Accidents assessments for UK ABWR (Refs. 120 and 138). For my assessment of them, I consider the adequacy of any chemistry-related claims. Also, many systems which provide normal operational and/or safety functions for the reactor can also provide mitigation during an accident, including the RHR system, S/P and other systems able to cool the reactor core. My assessment of these systems presented earlier in my report is also of some relevance to accident conditions (Sections 4.3 and 4.4). The overall objectives of my assessment for UK ABWR for the accident chemistry topic was to consider:

- The overall adequacy of the generic safety case made by Hitachi-GE for chemistry-related effects during faults.
- The assumptions made about the behaviour of molten core material (known as corium).
- The assumptions made about the generation, behaviour and mitigation of flammable gases.
- The assumptions made about the generation, behaviour and mitigation of fission products.
- The engineered safety systems which deliver chemistry-related functions, including their link to the above assumptions, and;
- The requirements and capabilities to measure chemistry-related parameters during, or following an accident.

872. My assessment was informed by a specific TSC review of iodine chemistry and behaviour in BWRs (Ref. 189). I have also made use of TSC reviews conducted during previous GDAs, such as Refs. 139 and 140. The main SAPs (Ref. 2) of relevance to this part of my assessment are ECH.1, those relating to fault analysis, in particular FA.1, AV.2 and AV.6, and the relevant numerical targets (including NT.1 to 3). These principles address the expectations that suitable consideration is given to relevant chemistry effects within the faults analysis. I have also considered the guidance provided within the associated TAGs (Ref. 4).

4.7.1 UK ABWR Generic Safety Case Overview

Fault Analysis for UK ABWR

873. PCSR Chapter 24 (Ref. 16) provides an overview of the generic safety case for Design Basis (DB) faults in UK ABWR. The main aim of this chapter is to summarise how the safety systems provided in the design have been shown to control all transients and accidents that make up the design basis, in all operating modes. It summarises the process that was applied by Hitachi-GE to develop the list of specific DB faults, and how this was used to produce the Fault Schedule that lists all of the bounding design basis faults that were assessed during GDA. This includes faults from a number of groups including reactor faults at-power, faults occurring during other operating modes, and non-reactor faults, such as those within the fuel route and the radioactive waste systems. Collectively these cover all operating modes and all main sources of radioactivity within the plant. The DB analysis is supported by a number of topic reports. These provide the detailed underpinning and analysis. A number of these formed an input to my chemistry assessment, including:

- *Topic Report on Fault Assessment* (Ref. 141);
- *Topic Report on Fault Assessment for SFP and Fuel Route* (Ref. 142);
- *Topic Report on Design Basis Analysis* (Ref. 143);
- *Topic Report on Design Basis Analysis for SFP and Fuel Route* (Ref. 144); and
- *Topic Report on SBO analysis* (Ref. 145)

874. The basic conclusion of the DB assessment made by Hitachi-GE is that, even with conservative assumptions, the acceptance criteria applied to the analysis are met. ONR's assessment of DB faults is reported in Ref. 120.

875. Chapter 26 of the PCSR (Ref. 17) extends the DB analysis of Chapter 24 (Ref. 16) to include Beyond Design Basis (BDB) faults, that is, faults with frequencies outside the range of DB faults, and provides analysis to demonstrate that there are no "cliff edge" effects near the cut-off frequency. The BDB analysis summarised therein shows that such faults do not lead to melting or considerable damage of the fuel so that no significant environmental release of any radioactivity occurs. This is achieved through demonstration of the adequacy of safety provisions based on defence-in-depth and diversity in the design. PCSR Chapter 26 (Ref. 17) also includes the Severe Accident (SA) analysis. This is performed to analyse hypothetical fault sequences that do indeed result in significant radiological consequences, or unintended relocation of radioactive material. The SA analysis is undertaken for many reasons, but of relevance to my chemistry assessment some of the main purposes are:

- To understand the severe accident phenomena and progression including fission product release and transport behaviour.
- To demonstrate the effectiveness of engineered features, strategies, and procedures.
- To determine the magnitude and characteristics of the potential radiological consequences, including societal risk, and;
- To demonstrate that there is no sudden escalation of consequences just beyond the design basis.

876. A number of TRs support the DB and SA analyses, including, *Topic Report on Beyond Design Basis Analysis* (Ref. 146), and, *Topic Report on Severe Accident Phenomena and Severe Accident Analysis* (Ref. 147), respectively. Relevant aspects of them were considered during my chemistry assessment, but they are assessed more comprehensively by ONR in Ref. 138.

Chemistry Related Aspects

877. The analysis which is used to demonstrate the safety case for UK ABWR, for DB, BDB and SA analysis, includes many explicit and implicit assumptions regarding chemical behaviour and phenomenon. For example, the speciation and volatility of radioactive species during a fault can impact on the capability and capacity of the mitigation

systems to deal with it, and ultimately the radiological consequences to workers or the public. Similarly the production of flammable gases during a SA is mainly due to a chemical reaction between the fuel cladding and water, with the rate and extent of this being an important determinant. I assessed various aspects of these assumptions, and others, throughout GDA. This included the responses to a number of RQs and two related Regulatory Observations, RO-ABWR-0043 and RO-ABWR-0066 (Ref. 25 and 49):

- RO-ABWR-0043 was raised during Step 3 to request evidence to support the claims made in the first version of the generic PCSR (Ref. 25) relating to controlling the pH of the S/P water during accidents. This RO required Hitachi-GE to review their safety case claims, provide supporting evidence and justify that the UK ABWR design reduces risks SFAIRP.
- RO-ABWR-0066 was raised later in GDA. Alongside assessment of the pH control aspect of the S/P, a number of other chemistry related assumptions in the DB and SA analysis were being discussed with Hitachi-GE. This RO required Hitachi-GE to provide evidence that they were giving suitable and sufficient consideration to chemistry effects in the fault analysis. While there was some overlap with RO-ABWR-0043, the scope of this RO was much broader.

878. I assess Hitachi-GE's responses to these ROs in Section 4.7.4, along with the plant systems that are intended to be able to fulfil those assumptions in Section 4.7.5. These formed a large part of my assessment of accident chemistry, but I also chose to sample aspects relating to flammable gas control and the behaviour of molten core materials (corium).

4.7.2 Corium Behaviour

879. Under SA conditions the fuel, alongside some of the core internal components may melt. If cooling of the core is lost the high temperature that can be reached are enough to degrade the fuel, control rods, reactor internals and other components. At such high temperatures the zirconium fuel cladding is oxidized by water, so the main constituents of the molten core, or corium, include (by mass): uranium dioxide, zirconium dioxide, zirconium, iron, chromium and nickel, as well as a large proportion of the FPs. The chemistry of corium is complex, but it is the impact this has on the physical properties of the mixture that is important, because it can affect viscosity and heat transfer. This can influence the accident progression, and ultimately the radiological consequences.

880. A description of the corium behaviour expected during SAs in UK ABWR is given in the *Topic Report on Severe Accident Phenomena and Severe Accident Analysis* (Ref. 147). This is based on the behaviour predicted by the Modular Accident Analysis Program (MAAP) code, as discussed in Ref. 148. This is the so-called "*best estimate*" model. It considers both the in-vessel and ex-vessel phases, with the former further split into the in-core and lower plenum phases. These three phrases therefore represent the entire flow path available to the corium during a SA, starting as solid fuel within the fuel rods and ending with molten corium on the lower drywell floor.

In-Core Behaviour

881. The first step in the behaviour is oxidation of the fuel cladding due to the chemical reaction with steam. This produces hydrogen and is the main source of flammable gases, as assessed in detail in Section 4.7.3 of my report.

882. Above around 900 °C the cladding may begin to melt. Next, the control rods and fuel may melt. It is predicted that this will start before their respective melting points are reached due to eutectic reactions between boron carbide and SS (at around 1200 °C) and uranium dioxide and zirconium (at around 2200 °C). As the control rods may melt

before the fuel, there is the possibility of re-criticality, especially if the degraded core is re-flooded with water at this time. This is specifically considered by Hitachi-GE, and assessed by ONR in Ref. 138. The molten corium will relocate to the lower parts of the core structures, mainly above the large core support plate. Ultimately the core support plate may fail and corium passes to the lower plenum within the RPV. This is likely to remain water filled and further interaction between the corium and water is possible. This will release further hydrogen. It is also possible that this reaction may be energetic enough to break the RPV, by an energetic Fuel-coolant Interaction (FCI) or what is sometime referred to as a “*steam explosion*”. Hitachi-GE argues that this is very unlikely in UK ABWR and the consequences of this would be tolerable in any case, which is assessed by ONR in Ref. 138.

883. These initial corium formation processes are amongst the most studied and best known of all SA phenomena. MAAP is an industry standard approach to modelling of these, and they are considered by the SA analysis in Ref. 147. While the chemistry related aspects would clearly be important to the extent and timing of these phenomena, I am content that Hitachi-GE has considered the impact of these adequately in their SA analysis in the context of corium behaviour; importantly I consider that the main impact of any change to chemistry would be bounded by the scope of uncertainty analysis conducted, as described in my assessment below.

Lower Plenum Behaviour

884. It is expected that the corium which accumulates within the lower plenum will stratify. Figure 39 shows the “*best estimate*” model of the corium in the lower plenum (from Ref. 147). This consists of a layer of particulate corium (particulate debris bed), an upper metal layer of lower density, and a heavier oxidic molten corium layer which is surrounded by a solidified crust. The heat transfer behaviour of each of these phases is different, which impacts on the potential failure mode of the RPV. There is much uncertainty in this, in particular how the various phases may distribute and how much of the FPs resides within each of these (and hence decay heat).

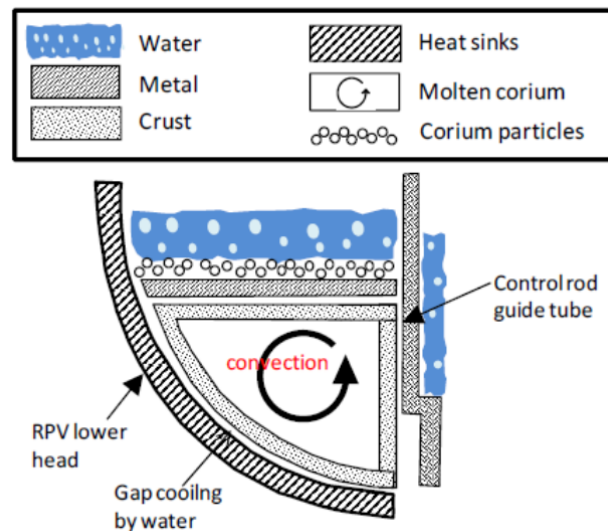


Figure 39: Best estimate lower plenum corium structure (Ref. 147)

885. There is no explicit chemistry modelling of this using MAAP, but there are assumptions and simplification applied. The approach taken in MAAP is to use interpolation to derive properties for four-component (uranium, zirconium, iron and oxygen) and higher-order mixtures. No attempt at detailed chemical modelling is made because the validation data did not exist at the time it the code was written and computing restrictions on complex properties modelling restrict what can be achieved. This leads

to inconsistencies in parameters such as component thermal capacities and simplifications such as solid and liquid mixtures having the same compositions. Nevertheless, the models do reflect the key features of the uranium-zirconium-oxygen system including the eutectic and extended mutual solubility at higher temperature. MAAP does not predict whether corium will form different layers but the user can specify whether the corium will be layered or mixed. In principle therefore, I consider MAAP is an appropriate tool from a chemistry perspective to model corium behaviour in the lower plenum, although this will always carry a large degree of uncertainty.

886. Initially cooling of the corium is provided by water remaining in the lower plenum, but once this is depleted the corium is no longer effectively cooled. The structures in the lower plenum, such as the RPV lower head, CRD guide tube, CRD housing tube, Instrumentation tube and housing, are over-heated and their temperature increases. Several failure modes are considered in the UK ABWR analysis including:
- failure due to local side wall overheated caused by the high heat conductivity of the metal layer;
 - creep failure of the RPV lower head caused by high pressure insider the RPV or the weight of corium;
 - jet ablation of the vessel wall;
 - penetration failure due to melt through; and
 - penetration ejection due to stress or melting.
887. Any changes to the chemistry assumed for the corium would impact on which of these failure modes is likely, and also the timing of when it may occur. For example, changing the mass of the upper metal layer or the distribution of FPs between phases may result in quicker rates of RPV ablation. The uncertainty associated with these failure modes is specifically examined in Appendix I of Ref. 147. While it is argued by Hitachi-GE that the dominant failure mode is analysed to be penetration ejection due to stress or melting (specifically the CRD tubes), consideration is given to how other failure modes may impact on FCI or Molten Core Concrete Interaction (MCCI) (and hence consequences). Ref. 149 also considers the impact of core melt progression and timing of vessel failure. It is concluded that this sensitivity analysis shows that the “CRD ejection” failure mode is bounding. Given the nature of this analysis, I am content that it is unnecessary to examine the impact of detailed chemistry on the corium behaviour in the lower plenum. The mode and timing of RPV failure, the main consequence of corium chemistry, has been examined parametrically. Provided this sensitivity analysis is considered reasonable in the wider SA topic, Ref. 138, I am content from a chemistry perspective.

Ex-vessel Behaviour

888. Irrespective of the RPV failure mode, corium released from the vessel will transfer to the lower drywell. Various phenomena could occur at this point, such as Direct Containment Heating or further FCI. These are not chemistry related, but are considered further by Hitachi-GE, and assessed by ONR in Ref. 138.
889. The lower drywell in UK ABWR is made of concrete and hence corium contact would lead to MCCI. Hitachi-GE has adopted water injection into the lower drywell, both before and after RPV failure as a means to minimise and arrest MCCI. This is discussed further in, *An ALARP Evaluation of Methods/Techniques for the Mitigation of Molten Core Concrete Interactions for the UK ABWR* (Ref. 146). This report considers the UK ABWR design features and a range of experimental tests. The conclusion is that the design includes all reasonably practicable measures to minimise MCCI, and importantly that the means taken to inject water into the lower drywell indicate that this phenomena is unlikely.

890. The main generic design feature for MCCI with chemistry relevance is the choice of concrete. Hitachi-GE proposes the use of a sacrificial concrete layer selected to have low condensable and flammable gas generation, and have identified that refractory concrete may also offer advantages. This does not impact on MCCI itself, but does have benefits for pressurisation of the containment and generation of flammable gases. Flammable gases are discussed in Section 4.7.3 of my report. I consider the use of such concrete to be a reasonable approach. The impact of MCCI on FP behaviour is assessed in Section 4.7.4 of my report.
891. The key safety driver during MCCI is to ensure that this does not lead to failure of the PCV. The underlying chemistry of the corium will affect this, but I would consider this to be within the overall boundaries of the uncertainties in the SA analysis. Various uncertainty analyses are reported for MCCI in Ref. 149. Other, more sophisticated codes than MAAP (as is used for UK ABWR) could be used to model MCCI in much more detail, including the chemistry effects. However, on the basis of what has been presented by Hitachi-GE, I do not judge that this would change the conclusions, nor undermine the basis of the safety case presented.

Corium Behaviour Summary

892. The UK ABWR generic safety case considers the relevant phenomena associated with the chemical behaviour of corium. These are important effects that could impact on accident progression and consequences; however they are subject to much uncertainty. As such the chemistry of this is not considered in detail, and the analysis code does not attempt to do so. It has been demonstrated that the sensitivity of the conclusions to this is comparable to the inherent uncertainty in such analysis, and no cliff-edge type effects were identified. It may well be necessary to revisit these aspects should more mechanistic tools be developed and applied in the future, but I would consider this to be normal business for a future licensee. I am therefore content with this aspect of the safety case from a chemistry perspective for GDA.

4.7.3 Flammable Gases

893. During an accident there are several potential sources of flammable gases, generated by a plethora of chemical or radiolysis reactions. In the presence of oxygen, potential detonation and/or deflagration of these gases may damage the PCV and result in increases in the quantity of radioactivity released to the environment.
894. Hitachi-GE's strategy for flammable gas control in UK ABWR is described in, *Flammable Gas Control and Supporting Analysis in UK ABWR* (Ref. 150). This report considers the overall strategy, engineered safety features incorporated into the generic design and presents supporting analyses. The scope of Ref. 150 covers DB, BDB and SA faults and includes the PCV and secondary containment. Flammable gases generated from the reactor and SFP are considered. A number of measures are implemented for UK ABWR to control flammable gases, should they arise. In summary they are:
- PCV – inerting with nitrogen, using PARs and the capability to vent.
 - Secondary containment – using the SGTS, PARs and the capability to vent. (including “blow out panels” in the secondary containment).
895. Not all of these features are therefore relevant to my chemistry assessment. I have focussed my assessment on the quantity and timings of flammable gas production, and any chemistry effects which are considered in the supporting analysis. This latter part therefore concentrates on the PARs. Hitachi-GE's full assessment of flammable gases and the resultant combustion risks during accidents in UK ABWR are considered in Refs. 120 and 138.

4.7.3.1 Quantities of Flammable Gases

896. Ref. 150 provides information on the sources of flammable gases considered by the UK ABWR generic safety case. This is consistent with the information presented in the PCSR and elsewhere.
897. For reactor DB faults the main source of hydrogen is from water radiolysis. To calculate the amount of hydrogen and oxygen produced, it is assumed that 100 % of the beta and gamma energy from the FPs in the coolant, and 10% of the gamma energy from the fuel is absorbed. G-values of 0.2 / 100eV and 0.1 / 100eV are assumed for hydrogen and oxygen, respectively. The basis for these values is given as a conference paper from 1988 (Ref. 151). This compares to higher G-values cited elsewhere, for example US NRC Regulatory Guide 1.7 (Ref. 152) gives G -values 2.5 times larger for both hydrogen and oxygen. The difference between these two sets of data is due to the impact of radioactive iodine. However, Ref. 151 appears to consider zero iodine, which even for a reactor with no damaged fuel is not a realistic estimate. Conversely, the larger values may be too conservative. No sensitivity analysis is provided by Hitachi-GE to examine the impact of these parameters, although clearly increasing the G-values will have a proportionate impact on the concentrations of flammable gases produced.
898. The importance of this assumption depends on the results of the analysis; this model directly supports the case made regarding the adequacy of the mitigation measures for flammable gases within the PCV (the PARs). Ref. 150 shows that without PARs a flammable gas atmosphere (oxygen > 5 vol. % and hydrogen > 4 vol. %) is reached within 104 hours of a LOCA occurring. The PARs specified for UK ABWR are shown to stop such a flammable atmosphere developing, limiting the hydrogen concentration to around 1.5 vol. %. However, the PARs only have a finite capacity and therefore it is not clear if a cliff-edge effect would occur if more hydrogen were generated, and at a faster rate. In effect, this would require a higher PAR removal capacity. Given the importance of this in determining if the measures adopted for UK ABWR are adequate, I would expect a future licensee to provide further justification for this aspect. I would not expect this to undermine the concept of using PARs in the PCV, but may well alter the specific design requirements. **I consider this to be an Assessment Finding:**

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.

899. For SA faults the amount of hydrogen is calculated using the MAAP code, as described in Ref. 148. For hydrogen production from zirconium oxidation, the MAAP models have been benchmarked against relevant experimental tests and the predicted hydrogen production should be similar to that obtained via other codes. The kinetic oxidation models in the MAAP analysis are believed to be the most consistent with knowledge in this area, so can be considered adequate. Several parameters during a SA could impact on the timing and extent of flammable gases produced, including the treatment of the boron carbide control rod material and the surface area available for oxidation during core relocation. There is therefore uncertainty in the rates and timings for hydrogen generation. However, in a SA it is assumed that the PARs within the PCV do not function (which is conservative) and the impact of hydrogen relates to how much leaks from the PCV to other areas of the secondary containment, which unlike

the PCV are not inerted with nitrogen. These aspects therefore need to be considered within the overall uncertainties in the SA analysis. In particular, Hitachi-GE has also undertaken a sensitivity analysis to examine the impact of increased hydrogen generation by doubling the fuel cladding surface area oxidised (Ref. 150). The net impact on the analysis is modest, suggesting that the results are not overly sensitive to this parameter. I consider that this is due to the other assumptions made in the analysis, such as the conservative PCV leak rate. I am therefore content with the predictions made by MAAP from a chemistry perspective.

900. For ex-vessel flammable gas production, in particular during MCCI, MAAP is again used to predict the concrete erosion, and therefore the amount of gases produced. However, it is important to note that the safety case presented by Hitachi-GE claim that MCCI is extremely unlikely, given the measures in place to mitigate this. Most of the analysis therefore does not predict any meaningful MCCI, even if the RPV fails.
901. The treatment of chemical effects during MCCI is described in Ref. 148. The approach to chemistry within MAAP is simplified, and assumes instantaneous chemical equilibrium, which is not realistic. The key variable in determining the amount of combustible gases produced during MCCI is the concrete ablation rate, which is essentially a heat transfer problem and is a function of the time between the start of the accident and contact of the corium with the concrete. Faster accidents generally have higher decay heats and metal content within the corium, which will increase ablation and flammable gas generation respectively. An important parameter is how much gas passes through the corium, rather than by-passing, which can be set by the user in MAAP. Ref. 148 contains details of validation of the MAAP code against MCCI experiments, which concentrate on predicting the ablation rate (i.e. the physical (heat transfer) process). These show reasonable agreement, which suggest that MAAP should provide reasonable estimates of this key parameter.
902. Ref. 150 contains an analysis to examine the effect of MCCI on flammable gas production. This considers a SA in the reactor with the RPV head removed. Hitachi-GE claims that this scenario is chosen to maximise the amount of MCCI as in this case the addition of water to the lower drywell fails. In this case the RPV is predicted to fail after 47 hours, hence the decay heat will be significantly reduced and the metal content may be lower (in fact the in-vessel phase has stopped before vessel failure). Nonetheless the results show that the amount of flammable gases produced during the MCCI phase is five times larger than that from the in-vessel period, but that carbon monoxide accounts for only about 10% of the total. This is clearly a very pessimistic scenario, as the MCCI proceeds to such an extent that the lower drywell pedestal fails.
903. Overall, I am satisfied Hitachi-GE has provided suitable and sufficient analysis for GDA regarding the production of flammable gases during MCCI, especially considering the low likelihood of such events.

4.7.3.2 Chemistry Effects Considered in Hitachi-GE's Supporting Analysis

904. PARs have been installed in many nuclear plants worldwide; see Ref. 153. These devices use catalytic material to recombine hydrogen and oxygen to water. They are passive devices, which operate once the temperature and gas concentrations are sufficient to allow the recombination reaction to proceed. The heat generated by the exothermic recombination further drives the process, as well as drawing fresh gas into the device as the hot exhaust fumes rise. How the PARs are modelled within the supporting analysis may therefore be important, as would consideration of effects which may interfere or degrade their function.
905. Before doing this it is important to place the PARs in UK ABWR in the context of the overall safety case. The approach to flammable gas management in UK ABWR is described in Ref. 154, which identifies where the different PARs are claimed:

- The PCV PARs are the primary means of removing hydrogen and oxygen during a DB fault. They are not claimed for SA conditions. It is also important to bear in mind that the containment would be inerted under most conditions;
 - The secondary containment PARs are one means available to remove hydrogen that has leaked from the PCV or generated from the SFP. Other means include the SGTS or venting.
906. The relevance of the various PARs (PCV or secondary containment) to the different accidents (DB or SA) is therefore a key aspect of the safety case made by Hitachi-GE. This therefore affects the importance, or otherwise, of accounting for the various chemistry effects.
907. The first chemistry effect which needs to be considered is the recombination reaction itself. Ref. 150 provides details of the PAR model used by Hitachi-GE, which assumes that the NIS-22 type PAR will be used. The correlation used to model the recombination reaction is based on analysis conducted in 1995. It is notable that this uses a “*degradation factor*” of 0.5, meaning that the hydrogen depletion rate is 50% of the maximum under any given condition. This is an important conservatism. Conversely the “*efficiency*” suggested seems quite high. However, it is well known that experimental tests conducted since then, in particular the OECD Thermal-hydraulics, Aerosols and Iodine (THAI) Programme, have improved the PAR models used in such codes. In particular to account for operation under initially starved conditions, as would be expected in the PCV of UK ABWR, the model used also restricts when the PAR operates to conditions where the concentration of hydrogen and oxygen are greater than 1.5 vol. % and 2.5 vol. % respectively. The rationale for this is given as “*experimental data*”, but no further evidence is provided. I also queried aspects of this correlation in RQ-ABWR-1337 (Ref. 49). The response does not provide detailed evidence, but instead argues that the analysis provided in Ref. 150 is sufficient to demonstrate the use of PARs.
908. The PARs may also be subject to degradation during an accident, in particular from FP deposition, acidic vapours or carbon monoxide. Engineered safety features may also interfere with their operation, such as sprays. Such effects will only occur in a SA. As such the PCV PARs do not need to explicitly consider this. However, the secondary containment PARs may well be subject to such effects, to some degree, albeit diluted by the secondary containment atmosphere. In response to RQ-ABWR-1337 (Ref. 49) Hitachi-GE acknowledges this, but notes that with the much reduced reliance on the PARs and the other means available to control flammable gases (notably the SGTS and venting) it is considered that even with no PARs the fundamental safety case claims can still be met. I expect a future licensee to adequately justify this argument. **This has been captured as part of Assessment Finding, AF-ABWR-RC-23, below.**
909. In Ref. 150 Hitachi-GE consider the impact of carbon monoxide, but only in terms of its impact on the flammability of mixtures with hydrogen. This concludes that carbon monoxide will increase the flammability limits and combustion pressure compared to pure hydrogen mixtures. The potential impact on PAR performance is acknowledged, but notes that the model does not include this. It is argued that this effect is negligible. However, the energy and flows produced from carbon monoxide recombination would be different than hydrogen alone. Given the safety case presented by Hitachi-GE, I am satisfied that this simplification may be adequate, but is not necessarily conservative and does add to the uncertainty in the supporting analysis. I would expect a future licensee to justify this argument. **This has been captured as part of Assessment Finding, AF-ABWR-RC-23, below.**
910. Overall, Hitachi-GE has used a simplified recombiner model, as is common practice, but it appears to not be the most up to date, nor consider a number of chemistry related effects. However it can be considered sufficient for GDA and further refinements to this would not be practicable given the current status of the design. I

would expect that the PAR model is more important in DB faults, given the reliance placed on the PARs and the inerted conditions. The PARs in the secondary containment would see less challenging conditions, and the reliance on the PARs is less. I would therefore consider that the analysis has shown that PARs may be a suitable design feature, but further analysis would be necessary to confirm this at the site specific stage once details of the PARs to be used are known. As part of this I expect the future licensee to confirm that the PAR correlation used is adequate, as this will directly impact on the design of the system. **I consider this to be an Assessment Finding:**

Finding:

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.

911. A further important assumption made in the supporting analysis reported in Ref. 150 is that *“the total capacity (hydrogen recombination rate) of PARs in the [PCV or secondary containment] is the summation of that of all PARs units”*. This simplification means that no account is taken of the environmental conditions around a given PAR, which would affect the recombination rate, and that all PARs work fully as intended. For the PCV it is argued that the gases and heat released during a DB or SA are sufficient to ensure that the gases are well mixed and therefore all PARs effectively see the same conditions, citing Ref. 155 which contains a summary of supporting experiments. I am content that this is a reasonable assumption to make for DB faults in the PCV. For the PARs within the secondary containment, which are distributed in many different physically separated locations; this is potentially a more important effect to capture. The analysis in Ref. 150 models the secondary containment using the GOTHIC code. Different schemes are applied to the nodalisation in this model depending on the purpose of the analysis, with the most detailed considering the refuelling floor split into 144 nodes. The PARs are located within two of these. The performance of the PARs is therefore a function of the gas conditions that are calculated for that node, and the combined performance of all the PARs assumed to be within that node. It is likely that this will capture the most relevant behaviour, to an extent that is adequate to demonstrate the applicability of PARs within the design for GDA. The on-going adequacy of this approach would need to be reviewed by the future licensee, depending on the outcome of Assessment Finding, **AF-UK ABWR-RC-23**.
912. One aspect that is not considered within the analysis is the treatment of hot gases which exit the PAR. As the PARs are not 100% efficient this will consist of hot gases which still contain flammable gases, but at a much lower concentration than at the inlet. In response to RQ-ABWR-1337 (Ref. 49) Hitachi-GE claim that, *“stagnant or accumulated area around outlet of a PAR hardly forms”*. This is true, but does not consider how such gases may lead to accumulations within the wider secondary containment, for example within the dome, which unlike the PCV are not expected to see conditions which would promote rapid mixing. Other effects may also need to be considered by the analysis, such as thermal loads from the PAR exhaust gases on SSCs in their vicinity. Proper modelling of these effects needs details of the design that are not yet available. It is not clear to me therefore, whether the analysis presented in GDA is sufficient to capture such effects, or whether they are relevant. Again, there are limitations on what can be achieved in GDA without a detailed system design. I

therefore expect further justification to be provided by the future licensee. **This has been captured as part of Assessment Finding, AF-ABWR-RC-23.**

913. The assessment of combustion risks is outside of the scope of my assessment, and is presented in Ref. 138. I do note that Hitachi-GE acknowledge that the PARs may act as an ignition source and have considered this within their analysis (although use a more conservative assumption of 10% hydrogen as the concentration above which ignition is possible). The resulting pressure is calculated using the Adiabatic Isochoric Complete Combustion (AICC) approach, which is bounding. The temperature loads from combustion are not considered in Ref. 150, which may require consideration of a different accident scenario but again need further system design information which is not yet available.

4.7.3.3 Flammable Gases Summary

914. The generic safety case for flammable gas mitigation in UK ABWR is much wider than my assessment of the relevant chemistry-related aspects above; this therefore only forms part of the ONR assessment on this topic. Following my assessment, it is clear what the mitigation measures are trying to achieve, and how the PARs form part of the overall case. On the basis of my assessment, I am content that Hitachi-GE has demonstrated that installing PARs in both the PCV and secondary containment are, in principle, an appropriate means to remove flammable gases, particularly when combined with the other measures in place.
915. However, it is not possible to fully substantiate the required performance of the PARs without a detailed design. As such, Hitachi-GE has used assumptions which may be appropriate, but will need to be demonstrated as such, by the future licensee. I consider it likely that the final design may need to be further refined, but that any such changes will not undermine the UK ABWR generic design or safety case presented during GDA. During my assessment of this topic I have raised two Assessment Findings, **AF-ABWR-RC-22 and -23**, to capture the specific matters I have identified which require further licensee input.

4.7.4 Fission Product Behaviour

916. The TRs on DB, BDB and SA analysis (Refs 143 – 147) present detailed descriptions of the performance of the UK ABWR systems under fault conditions. During GDA, Hitachi-GE undertook a systematic review of the chemistry effects that have a direct impact on the release of radiologically significant nuclides to the environment considered by this fault analysis. This was undertaken in two parts, leading to two main reports, *Topic Report on a Strategy for Consideration of Chemistry Effects in UK ABWR Fault Studies* (Ref. 156) and, *A Study of Chemistry Effects in UK ABWR Fault Studies* (Ref. 157). These TRs consider many relevant aspects of FP behaviour, identified under a number of underlying “mechanisms”.
917. One significant aspect of FP behaviour relates to the retention of iodine within the S/P. This was the subject of RO-ABWR-0043 (Ref. 25). In response, Hitachi-GE provided *Suppression Pool pH Model during Design Basis Accident and Severe Accident* (Ref. 158). Work in this report is also incorporated into Ref. 157, because it is one of the mechanisms considered by Hitachi-GE. This therefore falls under the scope of my assessment presented in this part of my report.

4.7.4.1 Hitachi-GE’s Approach and Scope

Identification of Applicable Mechanisms

918. Ref. 156 identifies the chemistry effects considered in the fault analysis, in particular those that directly support the claims against SSCs made in the safety case. The

approach taken by Hitachi-GE to identify the mechanisms for further consideration is comprehensive. It includes the review of the relevant DB, BDB and SA analysis reports to identify relevant scenarios, production of “flow diagrams” to illustrate the key phenomena and important SSCs, and determination of the important nuclides and effects. This considers 11 DB scenarios, representative of five main release pathways, one BDB scenario and 37 SA scenarios, grouped as five release categories. These include events at power, during different operating modes and from facilities outside the reactor and are consistent with the main fault analysis reports. The report does not include “consequential” effects, namely those which may have an indirect impact on radioactivity release such as combustion. Such effects are considered in the wider safety case but are not the main driver for this report. I consider this scope to be reasonable for GDA.

919. Based on this analysis, Ref. 156 identifies 17 mechanisms. Of the 17, five are identified as primarily being physical rather than chemical in nature, and are not considered any further. A number apply solely to DB or SA scenarios, but many are applicable to both, to varying degrees. Collectively, I am satisfied Hitachi-GE has identified the most important chemistry-related effects regarding FP retention and behaviour. The mechanisms identified in Ref. 156 are subjected by Hitachi-GE to further detailed evaluation in Ref. 157, to explore the uncertainty in the understanding and treatment of those mechanisms, including sensitivity analysis, where appropriate.
920. I chose to sample a number of these mechanisms in greater detail as part of my assessment. My sample largely aligns with the mechanisms defined by Hitachi-GE, but in some cases there is some overlap, or my sample is broader. My assessment approach was informed by my own understanding of the effects which may be the most significant, uncertain, or sensitive and also independent work conducted by my TSC (Ref. 88). I have also based my sample on Ref. 159, which identifies a number of important aspects of iodine behaviour. Whilst this means that my sample considers some mechanisms in much greater detail than other, my assessment did consider the overall adequacy of the generic safety case provided by Hitachi-GE.

4.7.4.2 Quantity and Speciation of Radioactivity Released to the Containment

921. This represents the starting point for all the transient analysis that follows and applies for all accidents considered. This is identified in the Nuclear Energy Agency (NEA) iodine report (Ref. 159) as an area of particular safety significance. Clearly, it is important that the starting point for any analysis needs a good understanding of the quantity and speciation of any radioactive release to the containment. This largely corresponds to “*Quantity and Speciation of Radioactive Release from Fuel (M1)*” from Ref. 157.

Basis for Assumptions

922. In summary, the radioactive inventories derived for DB and BDB scenarios are, in most cases, based on DB radionuclide concentrations calculated as described in the PST and PrST reports (Refs. 73 and 74), as no (additional) fuel damage occurs. I have previously assessed the derivation of these values as appropriate; see Ref. 79, as discussed in Section 4.3.1.4. The exception is the fuel handling accident, where the inventory is calculated using the ORIGEN code based on a number of irradiated fuel assemblies. The SA scenarios also use ORIGEN, but include the entire core inventory of 872 assemblies. Subsequent releases to the containment in a SA are calculated using the MAAP code, as described in Refs. 147, 148 and 160. It is beyond the scope of my assessment to consider the suitability of the MAAP code in detail, but I do note that it is an industry standard tool for such calculations and has undergone extensive verification and validation. I therefore consider it an appropriate tool to determine the quantity of radioactivity released into the containment during a SA. I do assess some

specific aspects of this code, but only in so far as they refer to chemistry related effects of significance within the analysis for UK ABWR.

923. One case worthy of mention is the BDB scenario. Hitachi-GE use the BE values for this analysis, as opposed to the DB values, used within the DB faults. This appears to be consistent with the expectation that DB analysis is conservative and BDB is best estimate (noting that this refers to the overall analysis, not necessarily each component of it). However, this does not consider the underlying derivation methodology for these values, which means that the DB analysis therefore includes the contribution damaged fuel makes to the coolant radioactivity levels, but the BDB does not. As Hitachi-GE do not present evidence to demonstrate the sensitivity to this assumption, it becomes very difficult to judge whether this is significant or not. Considering the radiological consequences for such faults in Ref. 146, I judge that while this would proportionately impact on the quantitative results, it should not threaten the overall conclusions. However, I expect a future licensee to review this aspect of the generic safety case. **I consider this to be a Minor Shortfall:**

MS-ABWR-RC-08: The quantities and speciation of radioactivity postulated to be released from fuel during accident conditions, derived in the UK ABWR generic safety case for the beyond design basis analysis, assumes Best Estimate radionuclide inventory values. This means the contribution damaged fuel makes to the coolant radioactivity levels is not included. The licensee should review and update the UK ABWR safety case to provide evidence to demonstrate the sensitivity of the consequences of such faults, to the use of the more conservative, Design Basis radionuclide inventory values.

924. As well as the radioactive inventory, information regarding the chemical form of releases is also provided in Ref. 157. It is assumed that iodine releases are present in water in ionic forms, transported in steam as hypoiodous acid and released during SA as caesium iodide aerosol. There are technical reasons to question some of these assumptions, in particular:
- The speciation of iodine in steam as hypoiodous acid is based on plant data under normal operating conditions, which are notably different to those during a fault.
 - Experimental tests have questioned whether iodine would be released as predominantly caesium iodide, in particular the Phebus FPT3 test (Ref. 159). This test may be of particular relevance to UK ABWR given the use of boron carbide and hafnium-based control rods. This test showed that a significant proportion of volatile iodine could be produced.
925. Ref. 147 considers how FPs are treated by MAAP in the SA analysis, including consideration of the release of inorganic iodine and organic iodine. This refers to the Level 3 PSA assessment report, Ref. 160. It is notable that Ref. 160 was updated several times since Ref. 147 was produced. In this latter report the “Base Cases” all assume all iodine is released in the form of caesium iodide. A number of sensitivity studies are reported, whereby the MAAP results are post-processing, to look at the impact of other iodine chemical forms. In these cases the NUREG-1465 assumptions are used, namely iodine speciation of 95% caesium iodide, 4.85% elemental iodine and 0.15% organic iodine (Ref.161). In addition, cases with much higher levels of volatile iodine were considered (up to 19.2% elemental iodine and 0.8% organic iodine). An important conservatism in all these cases is that these levels are established instantaneously. The results are compared against Targets 7, 8 and 9 of the SAPs (Ref. 2); the assessment of these results is provided in Ref. 162. From a chemistry perspective what these results demonstrate is that the form of iodine (whether as released or from in-containment effects) does have an important impact on the quantities of iodine subsequently released to the environment, particularly for

organic iodine given the difficulty in removing it. This is an important factor in my consideration of suppression pool pH control and the FCVS, discussed later.

926. Regardless of its chemical and physical form, iodine in the reactor containment will undergo further reaction within the containment, including during deposition onto surfaces and/or being washed into the S/P by condensing steam or containment sprays. Any effect of iodine being released in a different form will therefore depend on how these latter effects are considered within the analysis. I would therefore consider adequately representing these latter mechanisms to be more significant, given that the overall quantities of releases are relatively well understood, and the speciation at release only matters if it significantly impacts subsequent behaviours. In particular, provided suitable assumptions are made regarding the behaviour of any volatile iodine produced.
927. Similarly, because there is 10 times more caesium than iodine in a BWR core, caesium can be released in a number of forms during a SA, including: caesium iodide, caesium hydroxide and caesium molybdate. In the UK ABWR analysis, Hitachi-GE's assumption is that it is not released as caesium molybdate. A significant amount of recent international study has centred on the potential impact of the molybdate form of caesium on accident progression and consequences (for example, Ref. 163). I queried the impact of this assumption in RQ-ABWR-1343 and 1459 (Ref. 49). In response, Hitachi-GE explained the assumed forms (caesium iodide and caesium hydroxide) are considered to be conservative. It is noted that the MAAP version available to Hitachi-GE cannot study this effect, but reference is made to a MELCOR analysis performed by Sandia National Laboratories (SNL) which demonstrates such effects. I am satisfied this is a reasonable assumption to make, although this clearly depends on the impact that this may have on down-stream effects (in particular pH control in the S/P, as considered later in my report).

4.7.4.3 Radioactivity Scrubbing by the Suppression Pool

928. For a number of DB and SA scenarios, radioactivity released into the containment is first directed through the S/P water. The primary purpose of this is to condense steam and minimise the pressure within the containment, but this also reduces radioactivity by scrubbing. This is "*Dissolution and Scrubbing of Radionuclides into the Suppression Pool (M4)*", considered in Ref. 157.

Basis for Assumptions

929. In UK ABWR there are two possible routes for this scrubbing; either steam is directly released to the S/P via the Safety Relief Valve(s) (SRV(s)) and associated quenchers, or steam is released to the S/P from the upper drywell via the vent pipes/vertical downcomers. The S/P, including the vertical vent pipes are shown in Figure 36, in Section 4.4. The approach for considering these routes within DB and SA scenarios differs. The key assumptions for DB sequences are:

- DF = [redacted] or noble gases or organic iodine;
- DF = [redacted] for release via the SRVs; and
- DF = [redacted] for release via the downcomers.

930. Assuming no retention of noble gases or organic iodine is reasonable, and conservative for the iodine component as some small fraction may be retained in the water (through reactions rather than direct retention). The implicit assumption made by Hitachi-GE is that all other species, be they particulate aerosols or other forms of iodine, are removed with the same efficiency. The efficiency of the scrubbing process is a function of the more fundamental heat and mass transfer processes that take place when steam is sparged through water. There are many factors which influence this. Further information, including a useful summary of the basis, status, experiments,

insights and modelling of pool scrubbing was undertaken as part of the Passive and Active Systems on Severe Accident source term Mitigation (PASSAM) Project (Ref. 164); I considered this report as part of my assessment.

931. The basis for the DFs of [REDACTED] and [REDACTED] assumed by Hitachi-GE is presented in Ref. 157. For SRV releases this is based upon modelling using the BUSCA and SPARC codes, which are both codes specifically developed and validated for this purpose. Ref. 157 provides information on the testing and development of these codes and identifies the specific basis for the adoption of the DF [REDACTED] assumption, which is based upon United Kingdom Atomic Energy Authority (UKAEA), Advanced Containment Experiment and the EPRI tests. Hitachi-GE also considers the form of release, considering soluble and insoluble aerosols and iodine. Arguments are presented to suggest that the conditions of the various tests are, in most cases, conservative. Ref. 157 also contains a sensitivity analysis to assess the effect of reducing the DF in the long-term SBO fault (BDB). Various cases are considered with reduced DF claimed for aerosols or elemental iodine. It is concluded that the BSO for this fault could still be met even with a 10-fold reduction in the DF (from [REDACTED] to [REDACTED]).
932. On the basis of the evidence presented I am content that sufficient has been provided to support the use of a DF of [REDACTED] irrespective of the chemical form of the releases. The basis and evidence is largely consistent with Ref. 164. It is clear that, although there is potentially a wide variation in the actual DF that could be achieved, the use of [REDACTED] is conservative. DFs greater than [REDACTED] would have negligible impact on other aspects of the analysis in DB faults (for example by increasing the mass of iodine within the S/P), given the overall small concentrations and negligible impact of S/P pH (see later in my assessment for further details).
933. The much lower DF of [REDACTED] for downcomer releases is based on two NUREG reports, which again use the SPARC code that is integrated into MELCOR. Claims are also made that this is a further conservatism, given that not all radioactivity released during the fault (e.g. during a LOCA) would flash to steam and be subject to scrubbing in the S/P. This effect was not taken into account in the corresponding SPARC calculations which adds to the conservatism in the claimed DF. I am content with this assumption.
934. For the SA cases, scrubbing of aerosols in the S/P is treated within the MAAP code, using DFs calculated by the SUPRA code, originally developed by the EPRI. In effect SUPRA pre-calculates a series of DFs, which are applied within MAAP depending on the accident conditions. While Ref. 156 provides information on the approach, no DF values are given in the report, but it is noted that the uncertainty in the DF has been studied separately and found to be relatively small. Further information is also given in Refs. 147 and 148. I asked for further information in RQ-ABWR-1341 (Ref. 49). In response Hitachi-GE provided a series of parametric calculations examining the calculated DF with variations in aerosol particle size, steam fraction, sub-cooling and water depth. A comparison to experimental data is cited from an academic paper, which suggests reasonable agreement (considering uncertainties). For conditions which could be credible for UK ABWR, the calculated DF exceeds the value of [REDACTED] used within the DB analysis.
935. However, what is not clear is how the SA analysis for UK ABWR deals with iodine scrubbing within the S/P when it is not in aerosol form. SUPRA seems to consider only aerosols. Ref. 156 seems to suggest that scrubbing of fission product vapours is not treated in the MAAP calculations. This would be conservative, if volatile forms of iodine were treated in MAAP, but as this is not the case then it is equivalent to assuming the same DF for vapours and aerosols. This is unlikely to be realistic, particularly for organic iodine. The scrubbing efficiency of iodine will depend on the pH of the pool water, which is again not calculated by MAAP. It is not clear therefore how these effects are considered by Hitachi-GE, nor the impact they might have on the radiological consequences. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-24: The UK ABWR generic safety case makes important assumptions about the claimed efficiency of the scrubbing effect which the suppression pool provides for the radioactivity released during a severe accident. The approach effectively assumes that radioactive iodine is all released in the physical form of an aerosol. This overestimates the removal of volatile forms of radioactive iodine, and therefore potentially underestimates the potential radiological consequences. The licensee shall review and update the UK ABWR safety case and provide a proportionate justification to show that these aspects of the analysis adequately represent the physical and chemical processes taking place. This shall include considering the sensitivity and uncertainty in such events, and an examination of the overall impact on the postulated radiological consequences, to demonstrate relevant risks are reduced SFAIRP.

Radioactivity Scrubbing by the Suppression Pool Summary

936. With the exception of Assessment Finding, **AF-ABWR-RC-24**, identified above, overall I am satisfied with the chemistry related aspects of S/P scrubbing assumed by assumed in the generic safety case. I do note that these particular aspects of the case itself are often difficult to follow, and relate to the safety analysis performed for UK ABWR. During GDA, this was partially addressed in response to some my RQs, to such an extent that enabled me to undertake a meaningful assessment. However, as part of their on-going safety case development process, I expect a future licensee to address this aspect of the UK ABWR generic safety case, to arrive at a more structured, consistent and traceable output. **I consider this to be a Minor Shortfall:**

MS-ABWR-RC-09: The licensee should review and update the relevant aspects of the UK ABWR safety case which deal with the claimed impact of suppression pool scrubbing on radioactivity. This update should provide a clear, consistent and transparent demonstration that the assumptions used within the safety analysis are adequate.

4.7.4.4 Retention of Radioactivity within the Suppression Pool

937. Once radioactivity is removed by S/P scrubbing, it is important that it is retained for the longer-term within the water. Various chemical and radiochemical reactions can take place within the S/P water, due to the chemical, radiological and physical changes that occur during accident progression. This is particularly true during a SA, where the S/P will contain a large proportion of the entire radioactivity released to the containment during the course of the accident and radiolysis reactions become important. Other means of transferring radioactivity to the S/P may also come into effect following the initial steam scrubbing phase, including condensation of steam from surfaces and gas phase reactions. This is considered in Ref. 157 under the mechanism of "*Retention of Radionuclides in the Suppression Pool (M5)*".
938. The main nuclides that dominate the radiological consequences are noble gases, caesium and iodine. Most of the radioactivity scrubbed into the S/P will be aerosols and iodine, given noble gases are not retained. Thus the main focus of this mechanism relates to the retention of iodine within the water. Iodine has particularly complex chemistry as it is able to exist in multiple oxidation states and forms, some of which can be significantly volatile. The extent of this volatility is affected by many parameters of relevance to accidents, including the pH, temperature and dose rate. In general, high pH, higher temperature and low dose rate minimises volatility.

Suppression Pool pH Modelling

939. An important claim made by Hitachi-GE in this area is that, "*Alkali injection into suppression pool (S/P) is designed to control the pH of the S/P water and mitigate re- evaporation into containment gas phase of radioactive iodine retained in the S/P*", from

PCSR Chapter 26 (Ref. 17). Substantiation of this claim led to RO-ABWR-0043 (Ref. 25) being raised during Step 3 to request supporting evidence. The main response to RO-ABWR-0043 was *Suppression Pool pH Model during Design Basis Accident and Severe Accident* (Ref. 158). It also incorporates information previously included in Ref. 165. This document focuses on the S/P pH evaluation model for SAs and calculates results for UK ABWR, in order to assess the effectiveness of this mitigation measure. It presents how the pH changes during a single DB and five SA cases, and whether control of this can reduce the amount of iodine volatilised to the containment atmosphere from the S/P. Most of the content of Ref. 157 on this topic is a summary of information from Ref. 158.

940. The basic methodology used in Ref. 158 for the various accident scenarios considered is logically straightforward, and consists of:
- Industry standard calculation codes are used to calculate the FP inventories (ORIGEN) and releases (MAAP).
 - The MAAP data is used to calculate the dose rates to the water and containment atmosphere. Temperature data is also obtained from MAAP.
 - The pH in the S/P is calculated as a function of time using a bespoke method which is assessed in greater detail below, and;
 - The iodine volatility is then calculated from (i) the oxidised fraction at the S/P temperature, (ii) the relative concentrations of iodine and hypoiodous acid from the hydrolysis equilibrium constant, and (iii) the partition coefficients of iodine and hypoiodous acid at the S/P temperature. The organic iodide concentration is calculated assuming a formation rate of 0.01% per day. The partition ratio of iodine is then calculated, which is the ratio of iodine inventory in the water to that in the atmosphere. This iodine chemistry is therefore important, and is considered further below.
941. There are many substances produced during an accident that can impact upon the pH of the S/P water, to either acidic or alkaline conditions. Ref. 158 considers a comprehensive list of other potential pH influencing substances. The most important are FP aerosols, hydrochloric acid and nitric acid. With the exception of caesium, the amounts of FP are too small to affect the pH, but they are considered in the calculation of the dose rate. The speciation of caesium is important, with caesium hydroxide being basic and caesium molybdate, neutral. The most important reactions in determining the evolution of pH are those determining the production of the acids, hydrochloric acid and nitric acid, from radiolysis of cable materials and nitrogen, respectively. To ensure these effects are bounded the calculations consider the pH change assuming 100% of either caesium hydroxide or caesium molybdate, both with 100% and 20% of halogenated cables (noting the J-ABWR reference design is the former).
942. Hydrochloric acid production rate is a function of the energy deposition rate in the cable insulation, the mass of cable and the G-value. In Ref. 158 this is calculated using the gamma and beta dose rates in the containment atmosphere (from MAAP outputs), taking account of the attenuation by the air and the cable material. A G-value of 2.1 / 100eV is assumed, based upon Hypalon from Ref. 166. It is not clear if this is appropriate for the materials proposed for UK ABWR, however given the uncertainties in cable lengths, material and percentage of halogenated cabling to be used; I consider it a reasonable assumption to make at this GDA stage.
943. Similarly, the production of nitric acid is calculated assuming a G-value of 2.2 / 100eV in air. This is a well-founded value, based on many published experiments. However, the containment atmosphere will not be air, but a steam and nitrogen mixture. Hitachi-GE account for this difference using the electron density. In response to RQ-ABWR-1338 (Ref. 49) Hitachi-GE confirmed that they had not used the correct electron fraction for nitrogen, nor properly corrected the G-(nitric acid) value used for UK ABWR conditions. They subsequently revised their values. In response to RQ-ABWR-1456

(Ref. 49) an example calculation showed that making these changes increased the nitric acid production by around 10%. While, not material to the conclusions of the analysis conducted here, it does demonstrate the uncertainty in such calculations.

Iodine Chemistry Considerations

944. A number of important assumptions are inherent within the approach adopted by Hitachi-GE in their consideration of iodine chemistry:
- Hypoiodous acid is the only volatile species formed at meaningful concentrations when the iodine concentration is low; it has been shown that the volatilisation rate of iodine from irradiated solutions does not fall significantly when the iodide concentration is reduced from 10^{-5} to 10^{-9} mol. dm⁻³, as would be expected if the only volatile species were iodine and hypoiodous acid (Ref. 167).
 - Calculations assume an organic iodine formation rate of [REDACTED] % per day of the total iodine in the containment, based on measurements taken 14 months after the accident at Three Mile Island 2 (TMI-2); there are reasons to question this value, in particular the linear rate of production, and the claim made by Hitachi-GE that this is conservative. This is an important aspect of in-containment behaviour discussed in greater detail in the next section of my assessment.
 - As the models are based on the Sizewell 'B' safety case, the underlying data would be for iodine volatility in PWR conditions (aerated borate/phosphate solutions) not BWR conditions (inerted conditions, without boron). I asked for evidence to support this difference in RQ-ABWR-1130 and 1338 (Ref. 49). The response confirmed that no such tests had been performed, but provided evidence of comparisons of various codes on the production rate of organic iodine. This did not respond fully to the questions asked.
945. The net result of the above factors therefore further underline the nature of the analysis conducted by Hitachi-GE. Namely, it can be considered qualitatively reasonable in order to understand the potential impact of pH changes on the behaviour of iodine in the S/P. But the limitations on the methods and assumptions used should be borne in mind when attempting to apply the results in a more quantitative manner, for example when determining the amount of pH additive that may neutralise the acid produced or when determining the radiological consequence of such events.

Results of the UK ABWR Analysis

946. The main results of the analysis in Ref. 158 are presented in a number of plots which show various data such as the S/P pH and iodine speciation in the water or gas phase for a 72 hour period following the accident. Separate data is presented with and without the addition of [REDACTED] ppm of sodium hydroxide to the S/P as a neutralising agent; the basis of this amount of alkali is to neutralise the calculated acid production at 72 hours. 72 hours was considered as a conservative time frame for the analysis, as it does not include any venting which would reduce the inventory of radioactivity and acid within the gas phase. For the purpose of the calculations, to understand the behaviour, these are reasonable assumptions to make.
947. The analysis considers one DB and five SA cases, in these cases with [REDACTED] halogenated cabling. The results vary depending on the scenario considered. For the SA cases the analysis shows:
- The dominant acid is nitric acid, at around 30 times more than the hydrochloric acid produced by cabling. This results in a pH decrease to around two over the course of the calculation if no sodium hydroxide is added.
 - When sodium hydroxide is added the pH is initially increased to around 11, and decreases over the course of the 72 hours considered to around nine.

- Where caesium is assumed to be released as caesium hydroxide, there is a transient increase in pH to around nine after one to two hours, but this is rapidly removed once acid production exceeds the caesium release.
 - The gas phase concentration of iodine at the end of the calculations is typically about an order of magnitude lower for the pH controlled cases.
 - The amount of organic iodine was the same in each case, determined by the formation rate of [REDACTED] per day. In the pH controlled cases, this species was generally the dominant gaseous form at the end of the calculation whereas in the non-pH controlled calculations, inorganic iodine was dominant.
948. The results for the SA cases are qualitatively consistent with what might be expected, principally in relation to the difference in volatility between the pH-controlled and uncontrolled cases. However, the quantification of these differences is much more uncertain. For the uncontrolled pH cases, the final pH of two is significantly lower than in the experiments from which the oxidised fraction versus temperature expression was derived. It would be expected that the resulting oxidised fraction would be higher at these lower pH values predicted. Similarly, in the pH controlled cases the gaseous iodine concentration is dominated by organic iodine. This is entirely dominated by the assumed formation rate of [REDACTED] per day. These two factors mean that the difference between the controlled and uncontrolled pH cases may be larger than shown by the current analysis, but this needs to be considered in terms of overall uncertainty.
949. The assumption of [REDACTED] halogenated cabling is important. J-ABWR plants have considerably more, and there is uncertainty at present regarding what level of reduction can be achieved for UK ABWR, as this is a function of the detailed design. I expect the licensee to factor this into their future development. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-25: During GDA, ONR's chemistry assessment of the UK ABWR generic safety case established the postulated radiological consequences of accidents are sensitive to several assumptions. These include the inventory of halogenated cables used within the Primary Containment Vessel, and the type and specification of coatings applied to the Suppression Pool. Both of these can directly influence the inventory of radioactive iodine which may be released during postulated accident scenarios. The generic safety case recognises that detailed design information, licensee choices and other further work is needed before the overall importance of these assumptions is clear. The licensee shall review and update the UK ABWR safety case to provide a proportionate demonstration that all reasonably practicable measures to minimise the inventory of halogenated cabling, and specifying a suitable Suppression Pool coating, have been taken, to be able to show that relevant risks are reduced SFAIRP.

950. For the DB case considered by Hitachi-GE, their calculated dose rates are too low to produce significant amounts of acids, so there is no change to the S/P pH. Adding sodium hydroxide therefore has no effect on the calculated iodine behaviour. However, this arises from the particular model used, in which hypoiodous acid is the only volatile iodine species that can form under these conditions. Ref. 167 suggests that a pH effect would be seen at concentrations of iodine down to 10^{-9} mol. dm⁻³, which is within the range of the DB case considered by Hitachi-GE. This means the conclusion that pH control is not required for DB faults is not supported. However, recognising the much lower concentrations, and therefore radiological consequences of such faults, I agree with the outcome of this conclusion, namely that pH control in DB faults is unlikely to be necessary. However, I expect the UK ABWR safety case to provide a more evidentiary basis for this conclusion. **This has been captured as part of Assessment Finding, AF-ABWR-RC-26, below.**

951. Another specific aspect considered in Ref. 158 is the impact of water within the lower drywell area. During a SA there may be some accumulation of water within this area, which may not be pH controlled. Rather than claiming the lower drywell can be pH controlled, as with the S/P, arguments are made regarding the timing of iodine releases compared to these events occurring; in particular regarding the prior scrubbing of most of the iodine within the S/P. A sensitivity study was undertaken, considering venting of the wetwell with a flooded lower drywell both before and after RPV failure, and compared to the analysis undertaken in Ref. 158. This showed that the amount of iodine released in these cases increased by 37 and 43%, although this is in comparison to a pH controlled case. In comparison to a case of not controlling the S/P pH these increases would be much smaller. I am content that this demonstrates the relative importance of water within the lower drywell. It would be useful for the future licensee to see if any reasonably practicable measures can be taken to control the pH in this area, as part of the overall final design for the pH control system, given this could potentially lead to non-trivial reductions in volatile iodine. **This has been captured as part of Assessment Finding, AF-ABWR-RC-26, below.**

Impact of Suppression Pool pH Control

952. While Ref. 158 provides estimates of iodine behaviour, these are not used directly within the safety case to determine the impact of pH control in the S/P on the radiological consequences or risk targets. The impact on the L3 PSA results is described in both Ref. 157 and 165. The severe accidents topic report (Ref.147) also cites Ref. 165 as demonstrating the adequacy of S/P pH control. There is some confusion between these various reports, given they are produced at different times and, importantly, use earlier version of the calculations which have since been superseded. I therefore focus on the analysis provided in Ref. 157, given this is the most up to date report. This is not presented under the mechanism of “*Retention of Radionuclides in the Suppression Pool (M5)*”, but rather in a separate appendix which deals with “*Optioneering for additional iodine filtration in FCVS &/or active pH control in S/P in accidents*”.
953. The Level 3 PSA, Ref. 168, considers a range of sensitivity cases examining the effect of different assumed iodine speciation within the containment atmosphere. The results are compared against Targets 7, 8 and 9 of the SAPs (Ref. 2); the assessment of these results is provided in Ref. 162. The cases considered include:
- “Base Case” – 100% caesium iodide (as per MAAP outputs)
 - “Modified Base Case” – which uses NUREG-1465 assumptions on iodine speciation; 95% caesium iodide, 4.85% elemental iodine and 0.15% organic iodine (Ref. 161)
 - “Worst Case” – assuming 20% elemental and 0.8% organic iodine (and with no filtration provided by the FCVS, so effectively the speciation is not important for this study)
954. The argument presented in Ref. 157 is that the “modified base case”, is bounding for all the results calculated within Ref. 158. The effect of implementing pH control would be to move the analysis more towards the “base case”, where it is assumed all iodine is released in the form of caesium iodide. I do not agree, but consider neither of these cases are a reflection of the likely iodine behaviour, but are a reasonable assumption on which to understand the potential impact. Hitachi-GE report that irrespective of the iodine assumptions the relevant targets can be met by UK ABWR (aside from when using the “worst case” assumptions). I agree with Hitachi-GE that the “worst case” is very unlikely to be realised. It is noted that, “*Active control of S/P pH at SA would limit the formation of elemental and organic iodine and release to the PCV in a number of SA sequences. In these cases, the conditional risk benefit from pH control in the suppression pool could be significant and would add to ALARP*”, which I agree with.

955. On the basis of the L3 PSA it is concluded that the installation of a pH control system is not significant for demonstrating UK ABWR plant safety. This is based purely on meeting the SAPs numerical targets and does not consider ALARP. However, Hitachi-GE do conclude that a pH control system will be included in the UK ABWR design due to offering defence in depth in the design and because there are plants around the world that have implemented this, suggesting S/P pH control for SA represents RGP. Hitachi-GE contends that given the risk reduction this offers is limited, the safety classification of the system/s required to achieve pH control during a SA is unlikely to be onerous. The engineering of this system is considered in Section 4.7.5 of my assessment.
956. Considering my earlier observations regarding some of the assumptions and approaches used within the analysis, I am not convinced that the significance of S/P pH control is as small as suggested by Hitachi-GE. However, I do agree that the inclusion of a means to achieve this in the design is important. From a chemistry perspective effective, control of the S/P pH will undoubtedly result in a lower amount of volatile iodine which is available for subsequent release. The major benefit of this system would be in minimising organic iodine, which is difficult to remove once formed.
957. While not the main focus of my assessment, I also note that Ref. 165 considers a different set of L3 PSA analysis, using different iodine assumptions. Again, these do not make use of the volatility calculations described in Ref. 158. The results are qualitatively similar, with pH controlled cases decreasing the iodine release by between 3 and 96%. The importance of organic iodine is also clear from these results, but as they are different, and not cited in Ref. 157 I do not consider them further.
958. Ref. 159 also considers the DB scenario (LOCA), examining the effect of pH by changing the scrubbing effect of elemental iodine (from a DF of five if alkaline, to a DF of one if acidic). There is no change to the resulting dose. However, according to Ref. 157 iodine in steam is present as hypoiodous acid, not elemental iodine, and a DF of [REDACTED] for scrubbing of inorganic iodine is assumed. Neither of these factors is consistent with the above. It is therefore difficult to conclude whether S/P pH has no impact on releases from a DB LOCA, or whether this mainly arises from other assumptions in the calculation. **This has been captured as part of Assessment Finding, AF-ABWR-RC-26, below.**
959. Considering the inconsistencies present across the various submissions, and the technical questions I have on the analysis, I expect a future licensee to update the aspects of the UK ABWR safety case dealing with pH control within the S/P. I do not expect this to change the conclusions, but it would provide a clearer rationale for the benefits and impacts. This would be beneficial for both to the usability and clarity of the UK ABWR safety case, but also as an input into the detailed system engineering that will need to be progressed. **I consider this to be an Assessment Finding:**

AF-ABWR-RC-26: The UK ABWR generic safety case captures the requirement for the design to include a Suppression Pool pH control system, to be used during, or following, some postulated accident scenarios. The aim of the system is to retain radioactive iodine and potentially other forms of radioactivity, within the suppression pool, to ensure it is unavailable for potential release to the environment during or following an accident. The final design of this system will be significantly influenced by licensee decisions on how this technology is implemented in the design. The licensee shall review and update the UK ABWR safety case to provide a proportionate demonstration to justify the final design of the UK ABWR suppression pool pH control system reduces relevant risks SFAIRP. This shall include a suitable suite of supporting analyses which adequately considers all relevant physical and chemical processes taking place within the reactor containment.

Suppression Pool Boiling

960. An effect not originally considered by Hitachi-GE in Ref. 157 relates to boiling of the suppression pool, and the impact this has on the retention, or release of otherwise retained radioactivity. This relates to scenarios where a saturated pool boils as a result of pressure reduction in the containment, due to venting. Hitachi-GE claim that in most instances sufficient cooling of the S/P is expected to be maintained. The exceptions would be during a SBO or SA. It is acknowledged that such states could impact on radioactivity behaviour, however current analysis codes, including MAAP, do not adequately account for such effects and this is an on-going area of international research. For these reasons, Hitachi-GE is not able to fully quantify such effects for UK ABWR. Hitachi-GE's main arguments put forward to suggest that the analysis is suitably conservative to bound such effects, includes:

- The DF calculated (from experiments) for pool scrubbing does not differentiate between entrainment and other means of radioactivity release from the S/P, and therefore already includes this effect for saturated conditions.
- A simple calculation is performed to demonstrate the impact on entrainment of radioactivity already contained within the pool. This shows that the net impact is likely to be small, mainly due to the low concentrations expected in the S/P.
- The assumed equilibrium between the water and gas phase iodine concentrations is cited as bounding for the impact on iodine release. While this is indeed conservative, the assumption here is that the equilibrium under both conditions is the same.

961. I accept these arguments are reasonable, when weighted against the other uncertainties and conservatisms contained in these aspects of the UK ABWR generic safety case. However, given the on-going international developments in this topic, I expect the future licensee to consider this matter further beyond GDA. **This has been captured as part of Assessment Finding, AF-ABWR-RC-26, above.**

Other Fission Products

962. Ref. 157 also considers the volatility of other FP species. It is argued that, as the S/P will not see significant temperature increases (i.e. > 200 °C) the release of other FPs is not significant enough to have an impact on the radiological consequences. Ruthenium is noted as being a possible exception to this, but it is argued that the containment environment will inhibit the formation of volatile ruthenium tetroxide, and it would rapidly deposit on surfaces even if formed. I agree that the retention of other FP species within the S/P should not be a significant concern for UK ABWR.

Retention of Radioactivity within the Suppression Pool Summary

963. I am satisfied Hitachi-GE has given suitable and sufficient consideration to the important claim regarding the impact of retaining radioactivity within the S/P. The analysis conducted in the UK ABWR generic safety case is adequate to demonstrate the impact of important factors such as the cabling material and pH control in the S/P. This has identified several mitigations to be put in place that should maximise retention, including minimising the quantity of halogenated cabling and including a system to control the S/P pH. On the basis of my assessment completed and presented above, I agree with Hitachi-GE that these changes are necessary for UK ABWR.

964. Although I am broadly satisfied with the position reached on this topic during GDA, further important work and decisions will be required from a future licensee to develop this topic further. This has been captured as one of the outputs of my assessment as Assessment Findings, **AF-ABWR-RC-25 and -26.**

4.7.4.5 In-containment Behaviour

965. Once radioactivity is released to the containment, either directly from the RCS or after passing through the S/P, it will interact with the containment environment in multiple ways, including depositing onto surfaces, dissolving into water volumes, re-volatilising and reacting within the gas phase. The extent and nature of these reactions is strongly influenced by the environmental and radiological conditions during the accident, as well as being brought about by the actions of engineered systems, such as containment sprays. Several mechanisms considered in Ref. 157 describe the treatment of different aspects of this behaviour, including “*Containment Spray Operation (M2)*”, “*Retention of radionuclides on containment wall surfaces (M3)*” and “*Retention of radioactivity in Primary Containment (M6)*”. These mechanisms apply to a range of DB and SA scenarios.

Aerosol Behaviour

966. For FP aerosols processes such as thermophoresis, diffusiophoresis, gravitational sedimentation and inertial impaction will tend to remove these species from the atmosphere, where they deposit on surfaces or in the water volumes. For UK ABWR, a simple pressure driven leakage model is used for DB analysis which takes no credit for aerosol removal processes. For SA, additional processes that remove aerosol are calculated using the MAAP code. This is described under the mechanism of “*Retention of Radioactivity in Primary Containment (M6)*” in Ref. 157, although this only considers aerosol removal mechanisms, not re-suspension or volatilisation. These are physical, not chemical processes. The MAAP code, as described within Refs 147 and 148, is a standard tool used within the industry to do this, and has been validated against large scale experiments. On this basis I chose not to sample these aspects further.
967. In SA analysis the containment sprays in UK ABWR may be activated. While this is primarily to condense steam and reduce containment pressure it also has the effect of washing FP aerosols into the S/P, either from the atmosphere or containment surfaces. Ref. 157 mainly describes the treatment of these effects within the MAAP code and the underlying validation under “*Containment Spray Operation (M2)*”. These aspects are primarily physical in nature, rather than chemical, relying on effects such as Brownian motion or impaction to determine the removal efficiency. Hitachi-GE claim that their treatment of these effects is conservative, which I would agree with. However, operation of the spray may also have some chemical effects in removing volatile iodine forms from the atmosphere or increasing transfer from the S/P to the atmosphere, depending upon when it is used. These latter chemical aspects are not explicitly considered by Hitachi-GE in Ref. 157, nor are they modelled in MAAP, but again the importance of this relates to the consideration and assumptions for other in-containment behaviour, as I have described and assessed below.

Iodine Chemistry

968. The behaviour of volatile species themselves, in particular iodine, is much more complex than aerosols. This is exemplified by Figure 40 which, although it does not include all possible reactions, demonstrates how some of the important iodine species can form or change. This includes species both within the water and gas phases. While this diagram suggests that the behaviour is cyclic, this is not the case, with all of the reactions operating in both directions such that the observed effect is the product of production and removal mechanisms acting simultaneously. Which reaction(s) dominate depends on many variables and may change during the course of an accident.

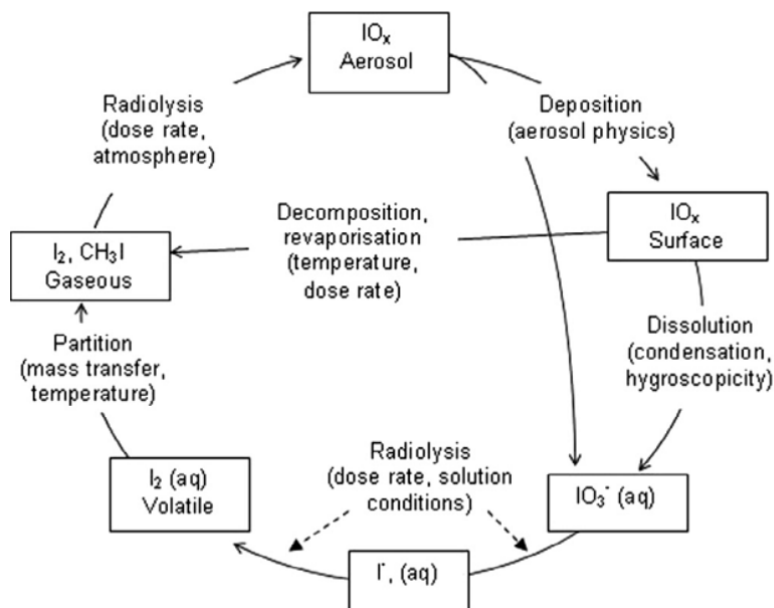


Figure 40: Typical iodine chemistry in containment (Ref. 139)

969. One of the most important aspects of this behaviour, retention within the S/P, has already been considered in my assessment above. However, what happens to iodine on surfaces and within the containment atmosphere are also relevant factors that need consideration. In particular the importance of organic iodine on the consequences of such events means that any assumptions on its behaviour should be reasonable.
970. Hitachi-GE treat FP aerosols that deposit on containment surfaces in a manner which assumes they remain on the surface unless they are washed into water pools or the S/P by containment sprays or steam condensation, otherwise they are treated as chemically inert and make no contribution to iodine volatility. Hitachi-GE state that they have assumed that all of the radioactivity deposited on walls flows into S/P and is then available for re-evaporation and re-suspension, which is claimed to be a conservative assumption. This is unlikely to be a realistic representation since deposited aerosol will be dissolved in surface water from steam condensation and/or sprays, and dissolved iodine will be subject to the same radiolysis reactions as would occur in the bulk S/P. The same factors would have a bearing on this, such as temperature, dose rate, concentration and pH, but these may be markedly different in relatively thin water films on surfaces as opposed to a large water volume such as the S/P. These reactions could therefore lead to:
- Release of volatile iodine (e.g. iodide, iodine, hypoiodous acid) directly from the water film.
 - Interaction of iodine with the underlying surface, resulting in retention by the surface rather than washing into the pool.
 - Release of organic iodide formed by interaction of iodine with organic-based surface coatings.
971. To justify their approach to this, Hitachi-GE produced a sensitivity analysis, which was then reviewed against the L3 PSA results. The response provides the rationale for selecting a suitable SA sequence for analysis. This is based on identifying the scenarios which would promote wall effects (e.g. early core damage with an intact containment) with further selection based on the contribution of that scenario to Target 9. Unfortunately this means that the selected scenarios are only calculated to have a low amount of wall deposited caesium iodide, up to 0.26% of the total iodine release. It is not clear what this might be in absolute terms (i.e. by mass), how this might compare

to other sequences, or whether this is the “instantaneous” behaviour or that after a period of time. Other sequences may be considerably higher. This is a product of Hitachi-GE basing their comparisons firstly against Target 9, rather than showing the impact on iodine behaviour before relating the results to Target 9. This makes judging the conclusion more difficult.

972. Nevertheless, Hitachi-GE analyse the resulting impact. The iodine behaviour is the same as that described in Ref. 157 and 158, and my preceding assessment. Many assumptions are necessary in this analysis, such as the water film volume, pH and dose rates. Organic iodine formation is assumed to be the same as in Ref. 158; therefore no account is taken of any surface formation mechanisms. The analysis is therefore subject to much uncertainty. Hitachi-GE calculate that in this case, around 26 % of the volatile iodine comes from the wall films, but this is small compared to the “modified base case” assumed within the L3 PSA. I am not convinced this is a fair comparison, given Hitachi-GE argue that for a properly controlled S/P pH the “base case” is more representative. This suggests the impact of wall films may be significant, in absolute terms. Hitachi-GE has not been able to demonstrate why this effect does not merit any further consideration though. Based on the information presented during GDA, Hitachi-GE was unable to demonstrate why their assumption of full transfer of deposited aerosols to the S/P, is conservative. Despite this, considering the conservatism used elsewhere in the analysis, I am satisfied this does not undermine the UK ABWR generic safety case, but I expect a future licensee to resolve this matter. **This has been captured as part of Assessment Finding AF-ABWR-RC-26, above.**
973. The PCV in UK ABWR is equipped with a number of PARs for hydrogen control. While these are not claimed in the safety case to be available for hydrogen removal during a SA, it has been shown that they can also have an impact on iodine chemistry, producing volatile forms of iodine (for example, Ref. 169). This is not considered by Hitachi-GE. I consider it is likely that it may be shown that such effects are not significant to the overall behaviour, but this should be demonstrated, considering the UK ABWR’s relatively small containment volume. **This has been captured as part of Assessment Finding, AF-ABWR-RC-26, above.**
974. Ref. 157 does consider the reactions of iodine with the containment surfaces, under “Retention of radioactivity in Primary Containment (M6)”. The reactions of molecular iodine with steel and zinc primed surfaces are mentioned, but no comment is made about the relevance of these to the UK ABWR design. The formation of organic iodide is discussed at some length, but again there is little presented to link the information to the design, in particular whether organic paints are used within the containment. RQ-ABWR-0409 (Ref. 49) refers to the paint specifications for UK ABWR, where it is confirmed that they are not yet specified. It would appear that J-ABWRs have not considered such paint effects when selecting the final coating. I expect the licensee to factor this into their final choices. **This has been captured as part of Assessment Finding AF-ABWR-RC-25, above.**
975. The discussion on organic iodine cites the main formation mechanism as being a heterogeneous reaction between adsorbed iodine and the organic substrate. This is inconsistent with the approach taken in examining the effect of iodine containing water films discussed earlier. In reality, it is not clear whether organic iodine formation is instead dominated by homogeneous reactions between reactive iodine species (molecular iodine, iodide) and organic materials released from surfaces. Hitachi-GE claims that gas-phase reactions are considered to be of minor importance. However, as no attempt is made to quantify the different processes it is not possible to substantiate this conclusion. Removal mechanisms are only briefly discussed. I would not consider a mechanistic understanding of these effects important, provided suitable assumptions can be made about the formation rate and quantities.
976. According to Ref. 157, the assumptions regarding organic iodine are:

- In the DB analysis, 3.2% of the iodine is converted to organic form; for LOCA faults this fraction is applied to the total release; while for other faults this is applied to the iodine not retained by the S/P.
 - In the SA analysis, 0.15% of the core inventory of iodine is converted to organic iodine at a formation rate of [REDACTED] per day.
977. The basis for using a conversion fraction of 3.2% for the DB analysis is WASH-1233 (Ref. 170). This is a 1972 review of containment experiments in which only gas-phase formation was measured. Around 70% of this was assumed to come from radiolytic reactions. The SA assumption of 0.15% conversion is obtained by applying the 3.2% conversion from WASH-1233 to the 5% of the core inventory assumed to be released as molecular iodine (which itself is taken from NUREG-1465 (Ref. 161)). The underlying justification is therefore WASH-1233 in both cases. I queried numerous aspects of these assumptions in RQ-ABWR-1341 and 1459 (Ref. 49), including the applicability of WASH-1233 data to UK ABWR conditions.
978. Hitachi-GE themselves conclude that “*the data stated in WASH-1233 is mechanistically incorrect and unduly conservative*”. Whilst the former is now generally accepted to be true (due to the importance of gas phase reactions), the latter is not demonstrated in Ref. 157. In response to my RQs Hitachi-GE clarified that, while there is a range of conversion rates given in WASH-1233 (up to 10%), it is claimed that comparable conditions to UK ABWR show much lower conversion factors. Additional factors such as the test conditions (iodide rather than iodine) and radiolytic destruction of methyl iodine would further limit the net conversion factor. In fact, Hitachi-GE claim that the gas phase reactions in the inerted conditions of UK ABWR would positively remove organic iodine; an effect not included in WASH-1233. Indeed inerted conditions would be favourable to organic iodine destruction (see Ref. 171), but may also favour gaseous production reactions so the net effect would need to be quantified for this to be claimed as a significant conservatism (which Hitachi-GE does not do, as gas phase reactions are considered unimportant). It is not clear if such an effect has been claimed in safety analysis for other plants, but Hitachi-GE have provided experimental data that suggests it may be reasonable to do so.
979. An important part of the supporting evidence for application of this organic fraction to SA cases is data from Three Mile Island (TMI-2) which showed that 0.16% of the containment iodine inventory was present in the gas phase before venting 14 months after the accident. It is claimed that this TMI-2 data is conservative because:
- Not all the gaseous iodine in TMI-2 was organic; this is likely to be true, however, it is estimated that > 90% was in this form.
 - 0.16% is the equilibrium concentration after 20 days; however, this does not account for the much higher dose rates that would be seen immediately following the accident which would tend to increase organic iodine formation.
 - The ratio between the gas and liquid volumes is much smaller in UK ABWR, which would decrease the organic iodine formation proportionately; the assumption that the fraction scales directly with this ratio assumes that the total partition coefficient is the same. However, the preceding discussion highlights the probable importance of surface formation mechanisms for volatile organic iodides, so a comparison of surface areas would arguably be more valid. In RQ-ABWR-1341(Ref. 49) arguments are presented for why Hitachi-GE believes ABWR should be lower; however these are not quantified in any way.
 - Radiolytic destruction would be more favourable in the inerted containment conditions of UK ARWR; as above, this is a likely reaction, but it is not clear how this balances against other effects likely to occur.
980. I would therefore consider the responses have not demonstrated that the TMI-2 data is directly applicable to UK ABWR. The same is true regarding the [REDACTED] per day formation rate, which again is based on the TMI-2 information, but looking at the re-

establishment of the equilibrium levels 20 days after the venting took place. The impact of lower dose rates during this time would be significant. Experimental tests show much higher release rates, by up to an order of magnitude, under representative irradiation conditions.

981. The overall argument made by Hitachi-GE is therefore that the assumption of 3.2% conversion to organic iodide is conservative since the technical basis for this value (production by gas-phase reactions) is incorrect. I do not consider that this has been quantitatively demonstrated in Ref. 157. However, for the DB analysis I would not expect the consequences to be particularly sensitive to this value, irrespective of any reasonable degree of change that might be considered. For the SA analysis, it is shown that organic iodine is an important consideration and therefore, as confidence cannot be taken in this value, it is important to consider the impact of this further. This would therefore lend itself to sensitivity analysis, but this is not done in Ref. 157 and no use is made of the iodine volatility calculations reported in Ref. 158.
982. Some sensitivity studies are reported under "*Retention of Radioactivity within Filtered Containment Venting System (M9)*" in Ref. 157, which although not aimed at this purpose, do illustrate the potential impact of increasing the organic iodine fraction. In these a wide range of possible FCVS filter factors (combinations of different particulate, elemental iodine and organic iodine filter factors) are examined for their impact on individual and societal risk (Targets 7, 8 and 9). This study uses the "modified base case" chemistry assumptions (namely with 95% caesium iodide, 4.85% elemental iodine and 0.15% organic iodine). In effect, changing organic iodine retention factors can be used as a surrogate to assuming different fractions on release. While it is difficult to use these results in a quantitative manner, they clearly demonstrate the importance of the fraction of organic iodine. The conclusion drawn by Hitachi-GE is that "*If the elemental iodine and organic iodide fractions were greater than those assumed in the Modified Base Case (i.e. significantly more than 5% volatile iodine formation in NUREG 1465) the releases to the environment, whilst allowing for a DF for I₂ of 100 in the FCVS, would not be expected to be greater than those from Case 1 (that is with no filtration and with the 5% volatile iodine). The results show that the BSO for Target 7 and Target 8 would not be challenged, although the BSO for Target 9 might be. The limiting condition is the "worst case" in [Ref-M9-10] where all the core iodine accumulates in the PCV as volatile forms and Target 9 is exceeded*". Without a direct comparison of these two cases it is difficult to confirm this assertion.
983. Overall, I am reassured that significant changes to the current assumptions on organic iodine (fraction or rate) would be necessary to invalidate the analysis conducted for GDA. I would however expect a future licensee to quantify the impact of this assumption in a more quantitative manner, given the significance of this species. This may well impact on future accident management arrangements or procedures. This has been captured as part of Assessment Finding, **AF-ABWR-RC-26**, above.
984. There are also many more gas phase reactions of iodine that are possible within the containment atmosphere. These are not considered by Hitachi-GE, because they are considered unimportant and they are assumed to be bounded by the more simplistic assumptions made. As they are not covered in Ref. 157 it is difficult to support that view. However, the most important impact of these on the consequences would be in producing organic iodine given it is not removed; hence this is covered by the Assessment Findings identified above. The potential generation of iodine oxide aerosols in the gas phase may impact on the FCVS, as assessed further below.

In-containment Behaviour Summary

985. The treatment of in-containment FP behaviour in the UK ABWR generic safety case is non-mechanistic, with simplified assumptions used. As such, Hitachi-GE has not quantified these effects, but has provided some indication of the likely sensitivity of the

radiological consequences to these, mainly through the use of L3 PSA. There remains much uncertainty in these values, but I am content that sufficient evidence has been provided to indicate that relatively large changes in the assumptions would be required to change the conclusions. I am therefore content with what has been presented for GDA, but have identified further work for a future licensee in this topic.

4.7.4.6 Mitigation of Releases by the Filtered Containment Venting System

986. UK ABWR is fitted with a FCVS. The main purpose of the FCVS is to prevent damage to the containment due to overpressure and consequential release of large quantities of FPs, particularly following a SA. During such scenarios, the FCVS treats the gas steam that contains particulate and gaseous radioactivity, using a combination of wet scrubbing (to remove aerosols and offer some chemical removal) and a fine filtration step (to remove fine aerosols from the wet scrubbing stage) prior to discharging the treated flow to the main stack. This is considered in Ref. 157 under "*Retention of Radioactivity within the Filtered Containment Vent System (M9)*", which considers both the claimed efficiency of this system and optioneering of the design.

FCVS Efficiency

987. The assumed efficiency of the FCVS is:

- DF = [REDACTED] for aerosols, including caesium iodide;
- DF = [REDACTED] for elemental iodine; and
- DF = [REDACTED] for organic iodine and noble gases.

988. The basis and sensitivity of these values is described and justified in Ref. 157. As the design of the UK ABWR FCVS is not yet decided, this includes experimental and empirical studies from the vendors of FCVS in addition to theoretical discussions of the processes involved. The removal of elemental iodine is based on chemical processes, with the scrubber solution containing a mixture of sodium thiosulphate and sodium hydroxide to minimise iodine volatility. The basis for these DFs appears to be supported by the evidence presented, which includes a wide range of test conditions and species, including particular tests for elemental iodine removal. They are also consistent with the information presented in Ref. 172, which includes a more systematic review of FCVS performance.

989. To date there is no quantitative analysis of the performance of the UK ABWR FCVS system. This is reasonable, given the current status of the detailed design. I would expect this to be progressed by the future licensee, as an input to the detailed system design and operating procedures. This should include consideration of other relevant matters, such as:

- I note that the FCVS does not consider iodine oxide aerosols which may be produced inside the containment gas phase. This is because the assumptions made in the preceding in-containment chemistry effects do not include it. Although this could be considered as an aerosol, its particle size may be small enough to make efficient removal difficult.
- The longer term retention of iodine within the FCVS will be governed by the same chemical processes as described earlier for the S/P. Therefore, as long as the pH is maintained alkaline and sufficient reducing agent (thiosulphate) is present, volatilisation should not be significant. This would need to be demonstrated as part of the detailed design, including the impact of other species from the containment (e.g. acid vapours).

990. I am content that these matters are best resolved by the future licensee. **I consider this to be a Minor Shortfall:**

MS-ABWR-RC-10: The UK ABWR generic design includes a Filtered Containment Venting System. The detailed design of this system was not fixed during GDA. This meant the information contained in the generic safety case regarding the system's efficiency and the chemical and physical radioactivity removal processes taking place, were not specific to the UK ABWR design and safety case. The information was largely based on vendor data and theoretical studies. The licensee should review and update these aspects of the UK ABWR safety case to ensure the supporting analysis adequately represents the physical and chemical processes taking place, and that the information is specific to UK ABWR postulated conditions.

991. Ref. 157 does include some consideration of the possible impact of the in-containment chemistry effects on the L3 PSA results. These consider cases which represent the two extremes with regard to iodine retention; from all the iodine is released to the containment as caesium iodide aerosol (i.e. no production of volatile species) with the claimed DF of [REDACTED] in the FCVS to all the iodine is released as elemental iodine with no filtering in the FCVS. Further sensitivity studies are also presented by varying the FCVS DFs for different species, but in these cases assuming the NUREG-1465 (Ref. 161) iodine speciation. It is not clear how other assumptions have been applied to these cases (for example is the caesium iodide release reduced by S/P scrubbing prior to the FCVS), but irrespective of this the analysis demonstrates the importance of the FCVS of reducing releases to the environment. An important conclusion is that there is a need to reduce iodine releases to at least 5% of the core inventory prior to release in order to ensure the relevant numerical targets (particularly societal risk, target 9) are not challenged. The analysis demonstrates that, in principle, much greater reductions than this can be achieved through a combination of S/P pH control and use of the FCVS. Changes to the DFs applied to different iodine species in the FCVS have varying degrees of impact, but the importance of organic iodine is apparent, given it is not removed.

FCVS Optioneering

992. Recognising the significance of organic iodine, Ref. 157 provides an optioneering review of means to remove this species as part of the FCVS. These include the addition of molecular sieve absorbers or additional chemical additives to the wet scrubber. These are also compared to the benefits offered by S/P pH control. I would not consider this last part to be a suitable comparison; given the former are mitigation measures, whereas the latter is preventative (thus higher in the hierarchy of controls). Irrespective of this the conclusion of the optioneering is that S/P pH control is the preferred approach, with the other means considered having other detriments. I am satisfied that this is a reasonable conclusion for GDA. I would expect the future licensee to review this as part of detailed design, which I would consider to be normal business.

FCVS Summary

993. Hitachi-GE has identified the efficiency claimed for the FCVS, which is backed by an extensive foundation of large scale tests under representative SA conditions. The importance of the FCVS has been demonstrated by L3 PSA analysis, which includes consideration of potential chemistry related impacts. Given the development of the design at this stage of the project, I am satisfied that sufficient information has been provided during GDA.

4.7.4.7 Mitigation of Releases by the Standby Gas Treatment System

994. The SGTS is included within the UK ABWR design to filter airborne radioactivity which may be released to the secondary containment during some DB accidents. Such faults include those within the spent fuel pool. It may also function during SA faults, where it is not claimed to reduce radioactivity, but is included as this maximises activity

released to the plant stack. The SGTS features various filtration stages, including High Efficiency Particulate Air (HEPA) filters for particulate and charcoal beds for iodine removal. The efficiency of this system is considered in Ref. 157 under “Retention of radioactivity in the Stand-by Gas Treatment System (M12)”.

SGTS Efficiency

995. Hitachi-GE claim that during DB faults the SGTS can achieve a DF of [REDACTED] (i.e. [REDACTED] removal) for particulate, elemental or organic iodine. No retention is claimed for the noble gases (i.e. DF = 1). It is also further claimed that this efficiency is maintained for at least [REDACTED] hours of operation following a fault.
996. Ref. 157 mainly focuses on the retention of organic iodine, as this is taken to be the most difficult to remove under accident conditions. Particulate removal is mainly via the HEPA filters, which is a standard industrial process. Iodine retention on charcoal is therefore discussed in terms of the impacts of parameters such as charcoal impregnate, flow rate, temperature and humidity. A range of experimental data is presented which demonstrates the impact of various factors, and appears to support the claimed DF. Information on DFs for “aged” charcoal is also presented, which demonstrates the impact of longer term use. I requested clarity in the applicability of this data to conditions during a DB fault in RQ-ABWR-1341 (Ref. 49). The response is not clear about what these conditions might be, instead comparing the DF to the expected operating conditions of the SGTS.
997. No sensitivity is presented for the impact of these DFs during DB faults, although looking at the experimental data the main potential impact would appear to be from humidity and temperature. The SGTS has specific components to control these to less than [REDACTED] RH and [REDACTED] before the charcoal. Unless the conditions significantly depart from these, a significant reduction in efficiency seems unlikely. Information is presented for a sensitivity study conducted for SA conditions, assuming various SGTS efficiencies. The main conclusion of relevance is that, provided a reasonable DF is maintained, doses are dominated by noble gases; this demonstrates the importance of the charcoal beds in removing iodine.
998. I am satisfied that the claimed efficiency of the SGTS is therefore reasonable.

4.7.4.8 Other Mechanisms

999. A number of other mechanisms are considered in Ref. 157. Those I consider most important to my chemistry assessment are discussed above. I do not discuss the remainder in detail here, other than the following brief commentary below.

Molten Core Concrete Interaction

1000. Should MCCI occur it may impact the behaviour of FPs. This is not considered as a mechanism in Ref. 157, but was included as a separate appendix as a result of RQ-ABWR-1343 (Ref. 49). This response explains the measures which would make MCCI unlikely, the treatment of FP releases within the MAAP code (as per Ref. 148) and further explains how these might be mitigated by the UK ABWR design. A sensitivity study is provided to examine the impact of a higher rate of MCCI which shows that while more FPs are released, the largest impact is on minor species which are not dominant in the consequences of SA.
1001. Considering the large uncertainties inherent in this process, the limited scope for it to occur and the limited impact, I am satisfied that this has been considered adequately for GDA.

SFP Accidents

1002. The amount of radioactivity released during an accident in the SFP, either due to dropped fuel or loss of cooling, is considered in Ref. 157 under the mechanism of “Retention of radioactivity in SFP water (M13)”. For both accident scenarios the approach adopted is claimed to be conservative. However, the approach used for these mechanisms may not be conservative from a chemistry perspective. For example no account is taken of radiolysis reactions in the SFP, and the partition coefficient calculated for a boiling pool assumes steam is lost from the system (which it won’t be). As the doses for both these accident are dominated by other species (noble gases for fuel handling accidents, and entrained material for a boiling pool) these non-conservatisms in iodine chemistry are unlikely to change the overall conclusions.
1003. For the SFP boiling accident, in addition to volatile iodine, radioactivity is also assumed to be released in entrained aerosols as a result of bubble formation. The activity in the aerosol is therefore the same as that within the bulk SFP water. Hitachi-GE assume this is equivalent to the normal operation DB water concentrations, with additional contributions from any damaged fuel stored in the pond, plus some release of fuel deposits from the fuel rods (although the value used is unclear). This is reasonable. However, credit is taken for the anticipated low fuel deposit amounts expected in UK ABWR and the efficiency of the shutdown process in removing this. It is concluded that this source is therefore negligible. I do not agree this has been considered on a conservative basis, and therefore potentially underestimates the impact of spalled fuel deposits. I expect a future licensee to review this approach, as this may impact on the provision and/or design of potential mitigation measures. **I consider this to be a Minor Shortfall:**

MS-ABWR-RC-11: In the analysis of spent fuel pool accidents, the UK ABWR generic safety case assumes, and takes credit for, the spent fuel having small amounts of fuel deposits. This approach may underestimate the impact spalled fuel deposits have on the radiological consequences following postulated spent fuel pool accidents. The licensee should review and update the UK ABWR safety case to justify that the treatment of spalled fuel deposits in a loss of spent fuel pool cooling fault has been considered in a suitably conservative manner.

4.7.4.9 Summary for Fission Product Behaviour

1004. Throughout the course of GDA Hitachi-GE worked very hard on this topic and provided a much improved generic safety case addressing the behaviour of FPs during DB, BDB and SA in UK ABWR. This has served to highlight the important considerations. While further work is needed on some of these aspects, the UK ABWR generic safety case now makes it clear to the future licensee what factors are important and should be the focus for further development beyond GDA.
1005. Much of the chemistry behaviour used within the safety analysis is simplified and not mechanistic, or uses industry standard computer codes to model the effects. This is a common approach, and is acceptable provided the assumptions made can be demonstrated to be reasonable. I have assessed some of the most important assumptions made by Hitachi-GE. Based on my assessment, I am satisfied a satisfactory justification has been provided for GDA. I have also identified three Assessment Findings, **AF-ABWR-RC-24, -26** for the future licensee to resolve, in addition to four Minor Shortfalls. **MS-ABWR-RC-08 – 11.**
1006. An important change proposed for the UK ABWR generic design is adopting pH control within the S/P during some postulated accident scenarios. It will directly impact (reduce) the quantities of radioactivity available for release during or following these scenarios. Similarly, Hitachi-GE’s analysis highlighted the benefit of minimising the quantity of halogenated cables within the UK ABWR containment. Both of these are improvements on the J-ABWR reference design. I agree with Hitachi-GE’s conclusion

that both of these are reasonably practicable improvements to implement for the UK ABWR design.

4.7.5 Engineered Safety Systems and Severe Accident Systems

1007. Chapter 13 of the PCSR (Ref. 173) provides information on the engineered safety systems included within the UK ABWR. Chapter 16 of the PCSR (Ref. 98) also includes a description of the severe accident systems. These chapters include details of the system design, functionality in normal operation and during faults, safety categorisation and classification and assumptions, amongst others. Most of the information in the PCSR is a summary of more detailed information contained with the underlying BoSC documents covering individual systems. For the most part, the designs are based on proven technology from the J-ABWR reference design. Additional risk reduction measures have been introduced for UK ABWR, including a re-design of the FCS, to replace active systems with PARs, and introduction of PARs into the R/B.
1008. In this part of my assessment I consider whether the relevant systems can deliver the chemistry-related functions claimed by Hitachi-GE within the UK ABWR generic safety case.

4.7.5.1 Chloride Ingress Protection System

1009. As described earlier in my assessment (Section 4.3.1.3), seawater ingress into the RCS can lead to corrosion, in particular SCC. While this can be controlled during normal operations using the CPS and CUW system (see Section 4.3.1.2), the potential exists for large and rapid increases in seawater leaks in the case of multiple main condenser tube ruptures. Such an event occurred at the Hamaoka-5 J-ABWR in Japan in 2011 (Ref. 174). Recovering from this event has required extensive inspection and remediation work to be able to demonstrate adequate levels of safety, prior to any plant re-start. While the UK ABWR CPS is designed with a 15 minute margin to handle a complete guillotine break in one condenser tube, a CIPS has also been added to the generic design by Hitachi-GE.
1010. The CIPS uses conductivity sensors and meters linked to an automatic interlock system which closes HPCP downstream of the CF and CD to prevent contaminated condensate being fed forward to the reactor. In the generic safety case, the main details on the CIPS are provided in two TRs, *Topic Report on Impurity Ingress* (Ref. 30) and, *Design Justification For Primary Water Systems* (Ref.18). The CIPS has yet to be fully designed for UK ABWR. However, the design presented for GDA appears reasonable and provides a suitable basis for development by the future licensee, as part of normal business. At present the CIPS interlock is identified as Category B, Class 3. Considering the overall safety case presented by Hitachi-GE for safely managing impurities, from a chemistry perspective, I am content with this classification. I welcome the introduction of the CIPS into the UK ABWR generic design, as a means to mitigate large scale condenser leaks. Although these events are rare, they have the potential to significantly impact the safe operation of the plant. Considering UK ABWR, by its very nature is a direct cycle plant, this is all the more important.

4.7.5.2 Filtered Containment Venting System

1011. The safety functions provided by the FCVS are described in PCSR Chapter 16 (Ref. 98) and in more detail in, *Basis of Safety Case on Severe Accident Mechanical Systems* (Ref. 175). While the FCVS fulfils a number of safety functions, relating to pressure control and containment isolation, the main one of relevance to chemistry is the filtering function during BDB and SA, which is identified as Category B, Class 3. An outline of the FCVS is shown in Figure 41. The main component is the vent filter itself, which is housed within a specific building separate from the reactor building. This is

connected to both the wetwell and drywell of the PCV. During normal operations the FCVS is maintained in an inert condition by nitrogen blanketing, to minimise the risk of hydrogen combustion caused by the inflow of combustible gases at the beginning of venting.

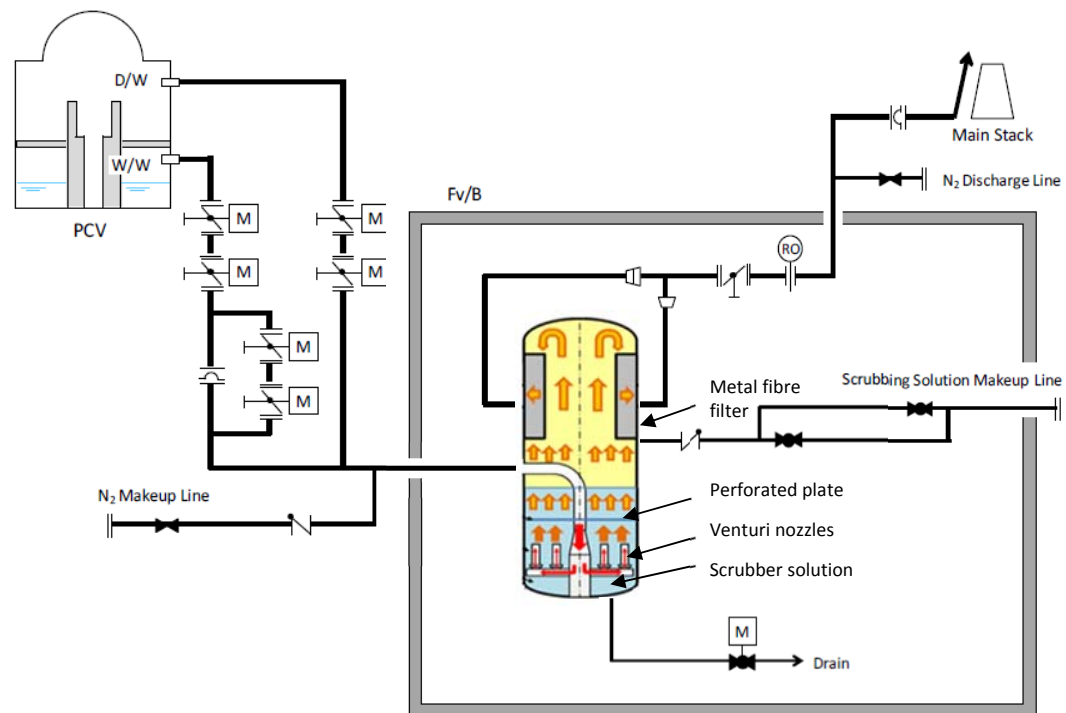


Figure 41: Outline of the UK ABWR FCVS

1012. The vent filter itself consists of a vertical, cylindrical vessel made of corrosion resistant SS, mainly to account for the impact of the scrubbing solution. A nest of venturi nozzles and perforated plate are installed in the bottom part and a secondary metal fibre filter is installed in the top part. The nozzles are fully submerged in a scrubbing solution of sodium hydroxide and sodium thiosulphate, whose purpose is to remove and retain iodine. This is a standard design concept, offered by many vendors.
1013. As described previously there are a number of design parameters that are important in ensuring that the claimed retention efficiencies can be achieved by the filter (██████ for aerosols and ██████ of elemental iodine), such as temperature or flow rate. These are consistently represented throughout the safety case (Refs. 172, 173, 147 and 157) and align with the basis assessed earlier. The exception to this is the requirements placed on chemistry of the scrubber solution. This is consistently stated to be:
- sodium hydroxide concentration of ██████ wt.%;
 - sodium thiosulphate concentration of ██████ wt.%; and
 - in a minimum solution volume of ██████ m³.
1014. These are captured in Ref. 20 under the “other ORs” section. There are no other chemistry related specifications for the scrubber solution. There is no basis for these values, which seem to be the standard requirements specified by the filter vendor. The only link is in Ref. 157, where the effect of pH on iodine retention is plotted and the conclusion is that the pH must be maintained above 13 (in fact the data suggests that much lower pH values, as low as eight, may be sufficient to maintain the ██████ retention). To achieve pH 13 would require only 0.4 wt. % sodium hydroxide, although some excess hydroxide is also required for the iodine reduction reaction and to

account for the thiosulphate. The basis for the concentration of sodium thiosulphate is also not given anywhere in the generic safety case, nor is the volume of solution required (although again a large excess in concentration and volume seems to have been specified). These may therefore not represent suitable safety related ORs, which may in fact be much lower. The concentrations required would be a function of the accident scenario considered to be bounding, with the amounts of iodine and other species absorbed in the scrubber solution having an impact (for example acid vapours). I also note that:

- The ORs as specified may not be the most appropriate, if the parameters of importance are the pH and thiosulphate ion concentration.
- Other parameters may need to be specified for the scrubbing solution, such as aggressive ions or impurities.

1015. I note that the FCVS does include a make-up line to refresh the scrubber solution, although no details of how this might be achieved are provided. No consideration has been given to ensuring this chemistry is controlled, such as sampling arrangements. Similarly a drain line is included, but it is not clear where this would be drained to, in particular following use of the filter when it would be highly radioactive. In some instances it is suggested that this may be returned to the S/P. Hitachi-GE indicated in response to my RQs that using sodium thiosulphate in the S/P would pose additional risks, hence it is not clear if returning it there is feasible. Such practices should be considered by the future licensee as part of normal business.

1016. While I am satisfied that the FCVS design should be capable of meeting the claimed chemistry-related safety functions, there are still gaps in the generic safety case at present regarding the definition of ORs and how these will be ensured during operation. I expect a future licensee to consider these matters further during the site specific stage as the detailed design develops, but I am content with what has been provided during GDA. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**

4.7.5.3 Suppression Pool pH Control System

1017. At the GDA stage there was no detailed design for the S/P pH control system. However, Hitachi-GE has specified the functional requirements for the prospective system and has performed optioneering for how this might be incorporated in to the UK ABWR design by a future licensee.

1018. *Basic Requirement Specification of Severe Accident Management Measures* (Ref. 176), defines the outline requirements for the system. Further information is also included in Ref. 157. From a chemistry perspective, these include the requirements to add \blacksquare m³ of \blacksquare wt.% sodium hydroxide to the S/P at a minimum injection rate of 100 m³ per hour. It is expected that this will need to be achieved within the first \blacksquare hours following the accident. This level of dosing will be sufficient to achieve a sodium hydroxide concentration in the S/P of \blacksquare ppm, which is demonstrated within Ref. 158 to maintain the pH at > 7 for at least \blacksquare hours. I note that there remains some uncertainty in these values, but these are consistent with those derived from the latest safety case analysis and are therefore appropriate at this stage. Hitachi-GE has currently identified the system as Category C, Class 3. I would expect the future licensee to keep this under review as the detailed design develops. However, for GDA, I am content this is reasonable.

1019. The optioneering exercise reported in Ref. 157 examines a number of potential options for how these requirements could be implemented. This also includes further information on other relevant considerations, such as availability of support services (e.g. C&I, electrical or gas supplies), impact on other SSCs and space requirements. The options considered are comprehensive and include modifications to existing

systems, new systems and both passive and active means of injection. While no decision is reached over which option should be selected, the optioneering shows that several are possible, and highlights some of the benefits and detriments of each. It is confirmed that sufficient space is available within the current plant layout to include two 3 m³ storage tanks with some contingency for associated pipework. I am comfortable that alkali injection as a means of pH control in the S/P has been demonstrated to be feasible.

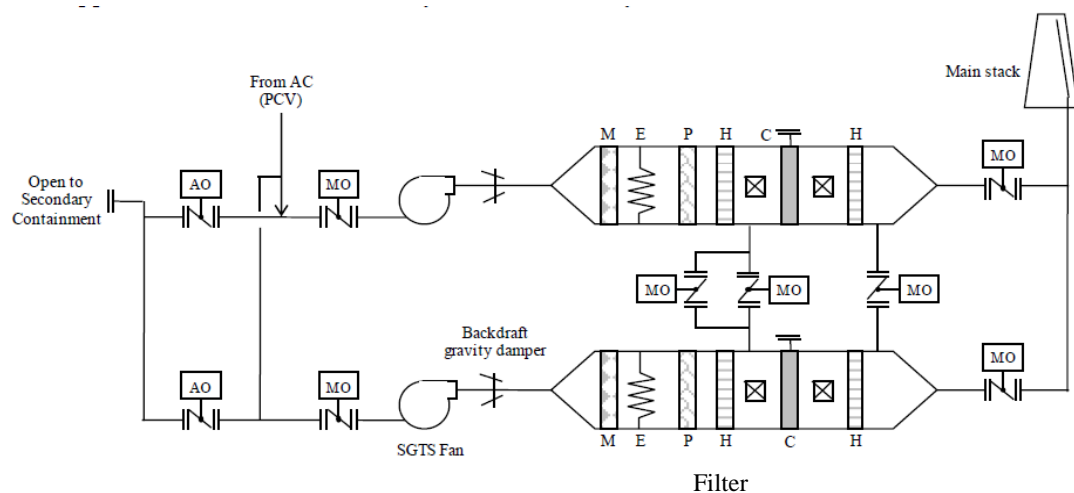
1020. There are many more aspects that will need to be considered by the future licensee to implement such a system within the UK ABWR design, including:

- Considering the consequences of inadvertent operation of the pH control system.
- Demonstrating the pH control additive can be effectively injected and distributed within the S/P.
- Examination, Maintenance, Inspection and Testing (EMIT) requirements for the system.
- ORs, including those related to chemistry requirements; and
- Measures and controls necessary to ensure that the chemistry requirements can be met, such as sampling arrangements.

1021. I am content that these are best undertaken by the future licensee, once the means of addition are finally decided. **This has been captured as part of Assessment Finding, AF-ABWR-RC-26, above.**

4.7.5.4 Standby Gas Treatment System

1022. The SGTS is designed to minimise radioactivity releases from the secondary containment in the event of DB faults. The SGTS is described in detail in the *Basis of Safety Cases on Standby Gas Treatment System* (Ref. 177). The filtration related safety function is assigned as Category B, Class 2. I consider this is reasonable, given the safety case claims placed on this function. The SGTS consists of two 100% capacity divisions including a fan and a filter train each as well as the necessary piping, valves, instruments and controllers. The system is designed to process a total of 50% per day of the free space volume in the secondary containment and the leakage from the PCV. The arrangement is shown in Figure 42.



KEY: MO – Motor Operated; M – Moisture separator; E – Process filter; P – Pre-filter; H – HEPA filter; C – Charcoal filter.

Figure 42: SGTS Outline

1023. The PCSR (Ref. 98) and Ref. 177 both state that the SGTS is designed to remove [REDACTED] of iodine, while particulate removal is [REDACTED]. The particulate retention factor is the same value assumed in the DB analysis, but the value for iodine is 10 times higher. It is not clear why this difference exists, particularly why conservatism is applied to particulate removal, but not iodine. I would expect it to be difficult to substantiate this performance during accident conditions, as demonstrated by the evidence given in Ref. 157, and discussed earlier in my assessment. However, the absolute difference these values make to the consequences of any accident would be negligible. As concluded in Section 4.7.4, I am content that the SGTS can meet the chemistry related performance requirements specified by the DB analysis, provided the temperature and RH are maintained within the specified limits.
1024. The SGTS is also one means claimed within the safety case for flammable gas control inside secondary containment. There are no chemistry claims related to this function, which is purely based on removal of gas to the plant stack. These wider aspects of the SGTS are assessed in Refs. 121 and 138.

4.7.5.5 Standby Liquid Control System

1025. The SLC system is the secondary means to provide reactor shutdown and maintain sub-criticality if the reactor cannot be shut down by the CRD or in an ATWS fault. PCSR Chapter 12 (Ref. 178) provides further details on the SLC, and is further supported by the *Basis of Safety Cases on Standby Liquid Control System* (Ref. 179). This system is identified as Category A, Class 2. From a chemistry perspective, I am satisfied with Hitachi-GE's categorisation and classification of the SLC System. The SLC consists of a storage tank containing a solution of sodium pentaborate. In the event of actuation this neutron absorbing solution is injected into the RPV through the High Pressure Core Flooder system sparger, thus providing boron directly into the reactor core. The solution is driven by two 100% capacity pumps. Also included within the SLC are the associated test vessel, piping and valves. An outline of the SLC system is given in Figure 43.

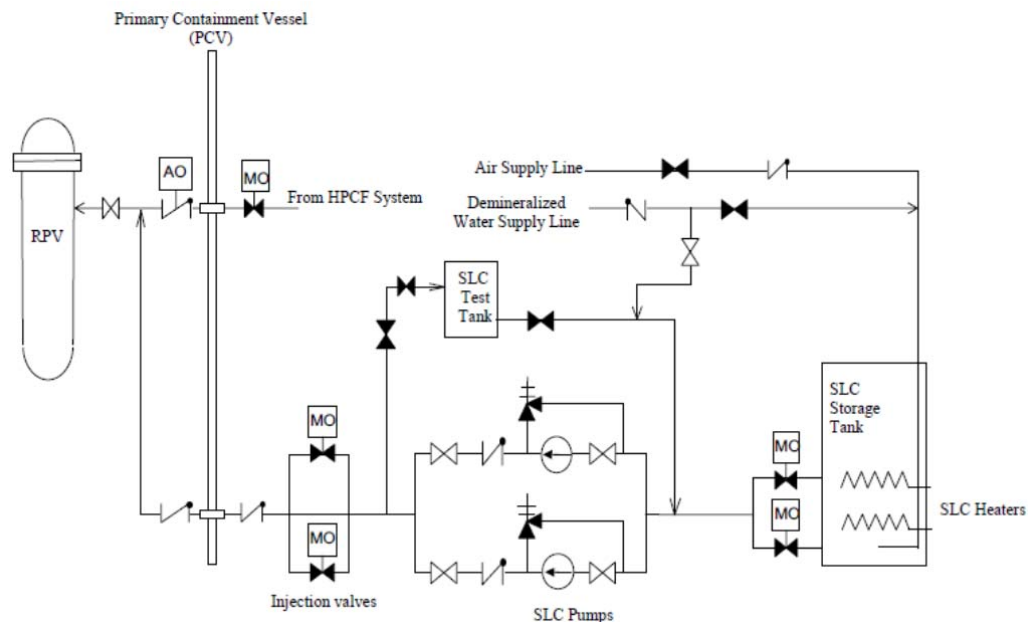


Figure 43: SLC System schematic

1026. The chemistry-related functions provided by the SLC are also considered within Chapter 23 of the PCSR (Ref. 11) and *Topic Report on Design Justification in*

Chemistry Aspect for Ancillary Systems (Ref. 19). The “top claim” and claim made on the SLC in Ref. 11 are:

- “[RC SC14] *The UK ABWR Standby Liquid Control System chemistry will ensure safe shutdown of the reactor in the event of Anticipated Transient Without Scram (ATWS) by injecting neutron absorber solution into the reactor water*”.
- “[RC SC14.1] *Sufficient and available sodium pentaborate solution is stored and maintained in an available state in the SLC*”.

1027. The scope of my chemistry assessment does not include examining Hitachi-GE’s detailed computer modelling of boron mixing in the core, or the reactor physics considerations of how much boron is required to achieve and maintain a safe sub-critical reactor shutdown. This is assessed in detail in Ref. 120. I have focussed on assessing:

- The measures in place to control the concentration of soluble boron and other relevant chemistry parameters.
- Design provisions for the SLC system which prevent the precipitation and/or stratification of boron.
- The adequacy of the materials selection for SLC SSCs.

1028. The size of the SLC tank and the concentration of soluble boron stored within it are determined by reactor physics and fault studies considerations. Hitachi-GE’s safety case documentation gives a number of boron concentrations that have been calculated to achieve a safe shutdown, which have each been derived to account for a number of conservatisms. Chapter 23 of the PCSR (Ref. 11) states, “*the capacity (of the SLC tank) is set at 28.7m³ and thus satisfies the minimum concentration of boron of 600 ppm claimed to be stored (in the SLC tank)*”. In this case the concentration of 600 ppm refers to the minimum concentration needed within the RPV, not in the SLC tank. The volume of the SLC tank is orders of magnitude smaller than the volume of water held in the RPV (~450 m³); therefore the concentration of boron in the SLC will be far higher (at least 24,500 ppm). I am not content with the ambiguities around this important parameter present in the generic safety case. **I consider this to be a Minor Shortfall:**

MS-ABWR-RC-12: The UK ABWR generic safety case contains ambiguities and inconsistencies in its presentation of the boron requirements for the Stand-by Liquid Control system. The licensee should review and update the UK ABWR safety case to provide a consistent and coherent set of nuclear safety requirements associated with this parameter.

1029. Considering its safety significance, the boron concentration within the SLC is included within the Generic Technical Specifications (Ref. 90), specifically within LCO 3.1.7. No other chemistry related ORs are defined for the SLC within the *Water Quality Specification* (Ref. 20). LCO 3.1.7 presents a simplified graph for the operator to use to correlate temperature with tank concentration (measured as weight % sodium pentaborate), with the boundaries between the “acceptable” and “unacceptable” regions shown. This effectively determines the point (temperature and concentration) at which boron starts to precipitate. While this does appear to define a safe operating envelope, I judge that the presentation could be improved to minimise potential error traps (for example, there is no mention of what form of boron the concentration refers to).

1030. Ref. 19 explains the requirement for the operator to sample the concentration of boron held within the SLC tank on a monthly basis and indicates the location of a sampling point on the tank. This is one of the requirements reflected in LCO 3.1.7. It is not clear why this frequency is adequate, although I note that this is down to the future licensee

to decide. The arrangements to sample the SLC are very straightforward, and given the inactive nature of the fluid, fairly benign.

1031. The US NRC prepared a report on ageing assessments of BWR SLC Systems (Ref. 180) in the early 1990s. Despite the report being dated, it is still relevant because BWR SLC designs have changed little and given their function, they are very rarely called upon. Ref. 180 identified that the chemical precursors to sodium pentaborate, boric acid and borax, can be one of the main sources of deleterious impurities in the SLC system. The report identifies the specific requirement to periodically sample the SLC system for the presence of impurities as a good practice to help prevent corrosion. If the future licensee were to choose to prepare sodium pentaborate in this way, the current set of ORs defined in Chapter 23 of the PCSR (Ref. 11) do not reflect this good practice.
1032. For the reasons above, I am therefore not content with the ORs currently defined for the SLC system in the generic safety case. **This has been captured as part of Assessment Finding, AF-ABWR-RC-02.**
1033. As noted above, one of the main risks for the SLC system is boron precipitation. The design of the SLC system recognises this risk. Ref. 19 indicates a number of features specifically address this risk, including:
- The temperature in the tank is controlled to 27 °C (± 3 °C) which provides sufficient margin to the 15 °C “limit”, which is the point at which precipitation begins to occur.
 - Two 100% redundant SLC tank heaters with independent normal and emergency power supplies are installed and the piping is trace heated.
 - If heater failure occurs, there is a 60 hour time window before the temperature drops to 15 °C, to be able to install a replacement.
 - SLC tank water level and temperature are measured continuously and the water level displayed in the control room with high and low temperature and level alarms.
 - Piping connected to the SLC tank is designed to prevent clogging if precipitation were to occur.
1034. The specific boron isotope which acts as neutron absorber is ^{10}B . Natural boron contains ~20% of ^{10}B . Some BWRs therefore use boron which is enriched in the ^{10}B isotope which means a lower concentration of total boron can be used, providing further margin to precipitation at a given temperature. If sufficiently enriched boron is used this may remove the need to actively heat the SLC tank and therefore provide a more passive design, consistent with the philosophy of SAP EKP.5. In Ref. 19 Hitachi-GE does acknowledge some of the benefits using enriched boron in the SLC System, but argues it is not necessary for a number of reasons (mainly commercial in nature, due to cost and availability). Most pertinent to chemistry would be that using enriched boron will introduce the requirement to implement further controls, including more advanced isotopic measurements, to confirm the levels of ^{10}B enrichment. It is also noted that natural boron is still widely used for Japanese BWRs and a number of US BWRs. I also note that there is nothing in the design which would foreclose a future licensee operating with enriched boron, if necessary. On the basis of the balance of benefits and detriments presented by Hitachi-GE and the measures included in the design to prevent boron precipitation, from a chemistry perspective, I accept that the use of natural boron is adequate for GDA. I would expect a future licensee to consider this choice as part of their normal business.
1035. There are some other detailed, yet important points, concerned with the design and operation of the SLC that are not considered. I would expect a future licensee to address these, plus related points, as a matter of normal business when developing the relevant operating processes and procedures. These are:

- The importance of knowing the chemical form of sodium pentaborate and applying this correctly in any calculations performed to determine the concentration of boron. For example, using the anhydrous form of sodium pentaborate instead of the decahydrate would result in a higher boron concentration.
 - Sodium pentaborate can stratify, which results in concentration gradients within the SLC tank if it is not agitated on a regular basis. The safety case is not clear on how the design has addressed this requirement, nor the operational importance of regularly agitating the SLC tank especially before sampling.
 - Means to add sodium pentaborate are not defined at present.
1036. Ref. 55 contains Hitachi-GE's justification for the materials selected for the SLC system. SS is selected for all main piping, but the more corrosion resistant Type 316L (NG) for Class 1 piping (inside PCV, contains reactor water at 287 °C) and Type 304L (NG) for Class 3 piping (outside PCV, contain SLC fluid at ambient temperature). In RQ-ABWR-1328 (Ref. 49) I asked Hitachi-GE why it was not reasonably practicable to use the more corrosion resistant Type 316 for all the piping. Hitachi-GE argues that the operating environment of the Class 3 piping is relatively benign (neutral pH and low temperatures), and purposefully controlled to remain that way, and Type 304 gives an acceptable performance. I am satisfied with this response, however I note, and as supported by the guidance contained in Ref. 180, that this relies on having adequate control of chemical impurities in the system. This materials selection also appears consistent with the information contained in the US NRC's report on ageing assessments for BWR SLC systems (Ref. 180), which explains that as expected, this has led to no serious corrosion issues. On this basis I am content with the material selection for the SLC main piping.
1037. On the basis of my assessment, I am satisfied that the SLC system should be capable of meeting the chemistry related claims placed upon it, although I note a number of matters where the overall chemistry-related aspects of the generic safety case requirement further improvement from a licensee. These are captured as a separate Minor Shortfall, **MS-ABWR-12**, and as part of Assessment Finding, **AF-ABWR-RC-02**.

4.7.5.6 Flooder System of Specific Safety Facility

1038. The FLSS consists of two trains of two pumps each of which can perform water injection into the RPV, when the reactor is in low pressure state, as well as flooding of various locations. The FLSS can be used to provide cooling water to the reactor core, the PCV spray header, the lower drywell, the reactor well or to the SFP spray depending upon the accident. A dedicated water source (FLSS Water Storage Tanks, ten units with capacity to provide a seven day supply) and individual piping (suction lines from the tanks and injection lines to the various destinations) are part of the FLSS, as well as the necessary valves, instrumentation and controls. The FLSS is described in PCSR Chapter 16 (Ref. 98) and *Basis of Safety Cases on Severe Accident Mechanical Systems* (Ref. 175). The FLSS functions are identified as either Category A, Class 2 or Category B, Class 2, dependent upon the safety function (as alternative means for DB faults it is Category A, as principal means for BDB or SA it is category B).
1039. From a chemistry perspective, the main interest in the FLSS is with regards to any water quality requirements for the water in the storage tanks. This is described in PCSR Chapter 23 (Ref. 11) and the *Topic Report on Design Justification in Chemistry Aspect for Ancillary Systems* (Ref. 19). The claim made on chemistry control is:
- “[RC SC17.1] *The water quality will be maintained and will not degrade the integrity of the structural material of the FLSS during the standby mode*”.

1040. The FLSS tanks are constructed from SS. This choice was justified in the material selection report for this system, Ref. 55, which I consider reasonable. The tanks are filled using demineralised water and the water is not chemically treated. However it is recognised by Hitachi-GE that there is a low probability that they may become contaminated by aggressive ions or microbial growth during storage, potentially leading to corrosion (most likely crevice corrosion). The tanks are also outside and trace heated and lagged to ensure they do not freeze, hence are only expected to see low temperatures. Corrosion rates are therefore very low, at least on internal surfaces, where water quality has an influence. External corrosion of such tanks, particularly under lagging, has proved to be an on-going concern on existing UK facilities. This is part of detailed site specific design, but should be considered by the future licensee as part of normal business.
1041. The *Water Quality Specification*, Ref. 20, defines the following ORs for the water quality within the FLSS storage tanks;
- Direct conductivity limited to $\leq 10,000 \mu\text{S m}^{-1}$ at 25 °C.
1042. This is not specified as a control or diagnostic parameter, but I assume it is the former. It is argued that this conductivity corresponds to less than 30 ppm of sodium chloride. It seems unlikely that the tank contents would approach anything like this level of contamination in normal use; however, even if it did the corrosion implications are limited in the short term. Periodic sampling of the FLSS tanks during routine pump testing would provide a means to monitor and take corrective action if necessary. The future licensee may also choose to implement additional lower level controls as forewarning. I am therefore satisfied with the ORs defined for the FLSS water quality for GDA.

4.7.5.7 Flammability Control System

1043. The FCS is part of the PCV gas control systems in UK ABWR, the other being the Atmospheric Control system. The latter does not provide any chemistry related functions and therefore is not considered further in my assessment. The FCS is provided to minimise the build-up of hydrogen from fuel cladding metal-water reaction, which might occur after a LOCA, and from the radiolysis of water. Further information on the FCS is provided in PCSR Chapter 13 (Ref. 173) and the associated *Basis of Safety Cases on PCV Gas Control Systems* (Ref. 181). The FCS consists of five PARs with associated instrumentation. Four units are located in the drywell, one unit is located in the wetwell. The safety function of hydrogen mitigation during accidents is identified as Category B, Class 2.
1044. It is also notable that the J-ABWR design for the FCS included thermal recombiners, not PARs within the PCV. This design change for UK ABWR makes the FCS a passive safety system. There are several chemistry-related claims made on the FCS:
- The FCS limits the hydrogen and oxygen concentrations in the gas inside the PCV to not exceed 4 and 5 percent by volume (vol. %) respectively after a LOCA, assuming the atmosphere within the containment is inert prior to the LOCA.
 - The hydrogen depletion rate is greater than 0.5 kg hr^{-1} , when the hydrogen and oxygen concentrations are $> 4.0 \text{ vol. \%}$ of each.
1045. These are reasonable claims, consistent with the overall requirements of the generic safety case. Both of the claims relate to the performance of the PARs. This will be impacted by numerous factors, including details of the PAR system design (for example number of units, type, size and location) and the accident scenario(s) used as the basis for the design, especially the rates of hydrogen production.

1046. *Basic Requirement Specification of Severe Accident Management Measures* (Ref. 155) states that, “Detailed design [of the FCS] will be completed in the specific plant design, with the selection of specific design of the PARs. The function and capacity of the system is to be confirmed by analysis. The method and assumption of the analysis and the result for a typical PAR design and number is shown in the related documents [Ref. 150]”. My assessment of the analysis presented in Ref. 150, in Section 4.7.3, concluded that, in principle, the use of PARs in the PCV may be a reasonable choice. However, I also noted a number of aspects with the current analysis which mean that I am not yet able to judge whether the current FCS design is able to meet the specific claims given above. Based on what is presented in Ref. 150, and my own knowledge of this topic, I would expect this to be possible with further analysis at the site specific phase and I do not expect major changes to the design to be necessary as a result. I am therefore content that sufficient evidence has been presented for the FCS during GDA.

4.7.5.8 PARs in the Secondary Containment

1047. Chapter 26 of the PCSR (Ref. 17) notes that a number of PARs will be installed on the operating deck inside the secondary containment. These are an additional measure proposed for UK ABWR that are not featured on existing J-ABWRs (but may be retrofitted). Further details are provided within Ref. 155. The PARs are installed to remove flammable gases that leak from the PCV, or are generated during accidents in the SFP, with the objective of maintaining them below flammable limits (i.e. 4 vol. % hydrogen in air). For some SA scenarios when the reactor is in a shutdown state (with the RPV open) or for the SFP, this objective cannot be met using PARs alone; they are therefore only one part of the overall measures proposed for UK ABWR to deal with these scenarios. The categorisation and classification of these PARs is not explained in the PCSR, or elsewhere. The basis for this system is consistent with the basis of the UK ABWR generic safety case, as per Ref. 169.

1048. As with the FCS, the detailed design for this system is not yet decided. The supporting analysis presented in Ref. 150, assumes 29 PARs will be installed. The locations are not yet fixed. My conclusions on the FCS are therefore equally valid for these PARs; namely, I am content that a suitable case could be made for this system, but as this depends on the detailed system design, I cannot as yet form a judgement on the adequacy of this system. The supporting analysis presented in Ref. 150 represents a reasonable starting point for this. I am therefore satisfied with what has been provided for GDA.

4.7.6 Accident Sampling

1049. As well as sampling and monitoring during normal operations, a proper response to faults and/or accidents may require such capabilities to be available to the operator. They may help during and following an accident. Sampling and analysis can provide an indication of plant condition, in particular the nature and extent of any fuel damage. Such information is important for accident management measures, as well as longer-term recovery. The scope of any sampling is likely to depend on both the nature and type of accident. The generic safety case for UK ABWR was initially limited in the information it provided in this regard, hence this was included as part of RO-ABWR-0081 (Ref. 25), as described earlier in Section 4.3.5.3 of my report.

1050. My assessment presented here is focussed on the adequacy of the SSC's which provide chemistry-related sampling capabilities during fault and/or accident conditions. There is no single system within UK ABWR that is responsible for accident sampling. Instead, many systems or part systems are involved at different times and locations. “Sampling” in this context means any chemistry-related parameter, whether determined via on-line or grab sampling means. Importantly, my assessment considers “accident sampling” as being any time after the initiating event, as opposed to the time

after the defined “safe state” is achieved. Based on this scope, a number of generic safety case documents are pertinent to my assessment here:

- *Monitoring Parameter in DBF* [Design Basis Faults] (Ref. 99);
- *Monitoring Parameters for Severe Accident Management in UK ABWR* (Ref. 100);
- *Topic Report on Sampling Capabilities of Chemistry-related Parameters in Design Basis Faults* (Ref. 105); and
- *Topic Report on Design Justification in Chemistry Aspect for Primary Water Systems*” (Ref. 18).

1051. My assessment covers DBAs (including BDB events in this context), through to SA. I have considered each of these separately, below.

4.7.6.1 Sampling for Design Basis Accidents

1052. The original report on this topic, Ref. 99, did not identify all chemistry parameters which may be required to be measured in a design basis accident. It did identify that the SAM system has a role, but did not provide an appropriate justification. In response to RO-ABWR-0081 (Ref. 25) and RQ-ABWR-1512 (Ref. 49), Hitachi-GE submitted Ref. 105. This report identifies and justifies the chemistry related parameters, along with the associated timing of any analysis. The following systems are identified to provide a function to confirm abnormal conditions at varying stages of a DBA:

- SAM system;
- Containment Radiation Monitor (CRM) system;
- Containment Atmospheric Monitoring System (CAMS);
- Off-gas System Post-recombination Hydrogen Monitor (OG PO HM); and
- the CIPS.

1053. To identify the parameters which may be required, Ref. 105 examines a representative 14 DBA faults groups to identify any chemistry related parameters and the required timing for measurements. This approach considers both during the fault (initiating event to safe shutdown state) and in the longer term (management and recovery). These faults groups are consistent with other parts of the generic safety case, and I consider this to be an adequate means of identifying any such requirements, with the response being logical and comprehensive. The result of this analysis is a tabulated list of parameters by accident. This mainly consists of measures for parameters such as radiation level and radioactivity concentration. Other parameters, such as flammable gas concentrations, are only analysed in specific accident sequences. Importantly this does not consider all normal sampling and monitoring parameters shown in the *Water Quality Specification* (Ref. 20) after achieving the safe state (via the SAM system), which are assumed to be undertaken.

1054. Justification of the identified chemistry-related parameters is also presented based on a comparison with good practice for European and US BWRs, as well as to guidelines from WENRA, the IAEA and US NRC. In general, Hitachi-GE concludes that they are aligned with the good practice and guidelines reviewed, and therefore justified in the selection of chemistry-parameters and timing of measurements for UK ABWR.

1055. Importantly in their review of international guidelines, Hitachi-GE references NEDO-32991 (Ref. 182), produced by the BWR Owners Group in 2001. This was prepared in response to OPEX that there was an unnecessary burden in maintaining post-accident sampling systems (PASS). Many BWRs include a specific PASS to enable chemistry-related sampling during post-accident situations, based on a requirement from the US NRC following the TMI-2 accident in the US. As noted above, UK ABWR does not include such a system. Ref. 182 concludes that many chemistry parameters are not

necessary to be monitored by a PASS in the short-term following an accident. As a result, Hitachi-GE conclude that:

- Chemistry-related parameters can instead be sampled and monitored using the SAM system, or other temporary equipment in line with the US NRC identified requirements of NUREG-1434 (Ref. 183).
- While European BWRs typically include a PASS, it is advised that this is not utilised to assess chemistry parameters and the dose rate monitors are more reliable. There remains a burdensome requirement to test the PASS system. As such the normal operational sampling system is proposed for use in DBAs.
- For US BWR operators, Hitachi-GE highlights that US NRC requirements have changed, with a number of utilities removing PASS requirements from their Technical Specifications, using calculations, as opposed to plant measurements. At least one US operator still maintains a requirement to demonstrate PASS operability every 18 months; however the general opinion in the US industry is that a specific PASS system is not required.

1056. Considering the arguments above, as well as the consistent and independent view presented by my TSC (Ref. 106), I am satisfied the exclusion of a dedicated PASS is a reasonable position for UK ABWR to adopt, provided the adequacy of the other identified systems for use in such conditions can be substantiated. Most of these (CRM, CAMS and OG PO HM) are C&I systems, assessed by ONR in Ref. 184. CIPS was assessed in the previous section of my assessment. I therefore concentrate on the SAM system below.

1057. In summary, it is Hitachi-GE's view is that chemical parameters obtained using the SAM system are not required for the detection or management of DBAs. However, they will provide supportive information relating to material integrity, fuel integrity and reduction of dose to workers, while also allowing fault conditions to be investigated once a safe state has been reached to support accident recovery in the longer-term. This is reflected in the claim made by Hitachi-GE in Ref. 124:

- “[SAM SFC 5-19.1] SAM provides supplemental information to check the plant condition if available during/after recovery from DBF condition”.

1058. As such, the SAM system sampling functions under these conditions is Category C, Class 3. I judged this is reasonable, given there is no specific requirement placed on the SAM system for accident mitigation. In principle, the SAM system is performing the same functions in normal and post-accident situations, and therefore my assessment of the SAM system presented in Section 4.3.5.3 of my report, is also relevant here. The main difference would be the potential concentrations of radioactivity within the samples. I queried the operability of the SAM system during periods of elevated dose including a DBA in RQ-ABWR-1512 (Ref. 49). In response, Hitachi-GE reiterated the position from Ref. 105 that, *“No chemistry-related parameter is required for the correct operation of the claimed safety systems”*, and that *“once the accident recovery systems have returned the plant to a relatively safe steady state condition, sampling will be re-established. From this time onwards, activity levels in all accident affected systems/component will be greatly reduced from the design basis maximum values and lie close to expected best estimate values”*. I judge these to be reasonable arguments. However, as explained in Section 4.3.5.3, there remains uncertainty over the capability of the SAM system to operate safely in situations where the level of radioactivity is elevated. **This has been captured as part of Assessment Finding, AF-ABWR-RC-16.**

1059. Overall, I am broadly satisfied sufficient evidence has been provided for sampling and monitoring chemistry-based parameters during DBAs in UK ABWR. The case made by Hitachi-GE includes a thorough consideration of the potential requirements, comparison to good practice, and OPEX. It provides clarity for the future licensee as

part of their accident management arrangements. However, for the reasons explained, I am not yet satisfied that the adequacy of the UK ABWR SAM system has been fully substantiated, but importantly for GDA, I am satisfied that a dedicated PASS (or similar) is not required in the design.

4.7.6.2 Sampling for Severe Accidents

1060. The approach to sampling and monitoring for SA events is detailed in, *Monitoring Parameters for Severe Accident Management in UK ABWR* (Ref. 100). This report uses experience gained from the management of the accident at Fukushima alongside relevant IAEA and WENRA guidance to select monitoring parameters for accident management and to identify accident stages. The approach is broadly similar to that adopted for DBA faults, described above. The first step is to categorise the parameters as either “primary”, “backup” or “specific”. In summary:

- “Primary” parameters are required in order to maintain the key safety functions and relate to preventing core damage, PCV damage, mitigating FP release, and SFP cooling.
- “Backup” parameters are noted as being “*preferable*” but not “*essential*”. They are either alternates to the “primary” parameters, or improve the reliability of the measurement, in combination with other measures.
- “Specific” parameters are a subset of both “primary” and “backup” parameters and are identified for very severe conditions, as a result of lessons learned from the accident at Fukushima.

1061. Ref. 100 identifies chemistry parameters related to reactor water (boron, conductivity and FP nuclides), S/P (conductivity and FP nuclides) and SFP water (FP nuclides, suspended solids, total organic carbon, conductivity and chloride) as “backup” parameters for both the medium and long-term. Drywell and wetwell gas concentrations (hydrogen and oxygen) are identified as “primary” parameters. While chemistry of the S/P and FCVS are described as “*items where monitoring might be required to control the fission product release from the containment*”, these are not specifically identified as falling within any of the assigned categories. Although I consider these to be gaps in the generic safety case, I judge this to be a result of the current status of the design for these systems. Hitachi-GE do give some consideration to these matters, noting that:

- While S/P pH is recognised as requiring control, Hitachi-GE argue that monitoring is not required prior or post containment venting as the amount of alkaline injected has been conservatively calculated. For longer-term control, Hitachi-GE suggests that “*if the S/P water quality condition will be needed to reconfirm additionally, grab sampling from RHR piping will be available*”.
- For the FCVS, Hitachi-GE claim that while it is important to maintain the scrubber tank solution at the required level and chemical condition, the design is still under consideration and will be determined as it progresses.

1062. Based on the above, I am satisfied Hitachi-GE has identified appropriate chemistry-related parameters, categorised them, and provided general time frames for their measurement to support severe accident management.

1063. Considering the categorisation of the identified chemistry-related parameters as “backup”, dedicated equipment to measure them is not provided in the UK ABWR generic design. Instead, the approach is that normal operational equipment (such as the SAM system) should be used, if it is available, or can be recovered. The generic design does not include any permanent specific design features solely to provide these functions for severe accidents. However, Hitachi-GE content there are other design features, originally provided for other purposes, which can be useful to connect

additional equipment (e.g. blank ports, piping connections, isolation valves, power connections etc.).

1064. In this regard, Chapter 16 of the PCSR (Ref. 98) describes the Outer Secondary Containment Grab Sampling (OSCGS) system. This highlights that prior to severe accident recovery or decommissioning activities, the OSCGS system may provide supporting information. Ref. 98 highlights that the UK ABWR is designed such that an accident can be managed purely through reliance upon process instrumentation monitoring, while post-severe accident grab sampling is subject to large uncertainties related to plant availability and accessibility. Hitachi-GE asserts that mobile sampling equipment is reasonable (alongside any remaining available normal SAM systems), given its flexibility and ability to connect to available sampling points and provide safe access for operators. Hitachi-GE note that the final grab sampling programme and system design will be determined at the site specific design stage, together with the decommissioning procedure, with sampling from the liquid in the RPV and S/P and gas in the wetwell and drywell, assumed for GDA. This is clearly consistent with the intent of Ref. 100, but there is no reference to the OSCGS system in this report and the generic safety case contains very little information about it. This aspect of the generic design and safety case is clearly still being developed.
1065. I am satisfied with the overall philosophy of using mobile or alternative sampling equipment in such SA situations, provided the UK ABWR design can be shown to support them. In response to RQ-ABWR-1352 (Ref. 49), Hitachi-GE provided the report, *Examples of Grab Sampling Provisions for Nuclide Analysis of Water Sample in Medium and Long Terms after Severe Accident* (Ref. 185). During my assessment, I considered the identified provisions for sampling the reactor water, S/P and SFP during SA conditions. In summary, arguments are made that the sample locations are available and suitable for SA conditions. However, I noted several aspects are not considered by Hitachi-GE in their response, including:
- potential dose to operators or any additional shielding requirements;
 - the representivity of any sample;
 - the availability of HVAC after a SA;
 - whether any alternative sample lines could be used; and
 - any links to the OSCGS system.
1066. Specifically, for the SFP, I consider that the capability to sample the SFP is important in potentially preventing a SA from deteriorating further. I judge this as being an important gap in the generic safety case presented for UK ABWR.
1067. My TSC specifically considered the identified reactor water sample point in a SA, noting that this was in good agreement with their view of good practice (Ref. 25). However, given the gaps identified above, I cannot yet agree this is a satisfactory approach.
1068. Overall, while Hitachi-GE have considered the provisions for sampling and monitoring in SA conditions, and strengthened this topic significantly throughout GDA, it is clear that this is still an on-going aspect of the development of the UK ABWR design and safety case. The sampling requirements have been identified, but there is more work to do to justify the adequacy of the proposed arrangements, and the potential consequences, such as ORE. I expect a future licensee to resolve these matters. **This has been captured as part of Assessment Finding, AF-ABWR-RC-16.**

4.7.7 Accident Chemistry Summary

1069. Hitachi-GE started GDA with a well-developed understanding of the fault analysis required to demonstrate the safety of the ABWR design, albeit not within a UK context. What was missing was a clear, consistent and transparent justification for the

adequacy of the chemistry assumptions used within the supporting analysis. These are important factors that impact on the transient behaviour and consequences, and the provision and design of, appropriate safety measures.

1070. In response to my assessment during GDA, Hitachi-GE worked hard and provided a range of supporting arguments and evidence which greatly improved these aspects of UK ABWR generic safety case. While I have identified five related Assessment Findings, **AF-ABWR-RC-022 to -026** and five Minor Shortfalls, **MS-ABWR-RC-08 to -12**, I am satisfied with the information provided during GDA, and judge that it represents a sound basis for the future licensee to further develop the UK ABWR design and safety case.
1071. Overall, in the context of a generic safety case, I am therefore satisfied the intent of SAP ECH.1 has been met. I am also satisfied the chemistry-related aspects of FA.1, AV.2 and AV.6 has been met. While it is beyond the scope of my assessment to assess the relevant numerical targets (including NT.1 to 3), I am satisfied with what I have seen from my own much narrower scoped assessment. Collectively, this means I am satisfied the overall objective of this part of my assessment has been met. This was to confirm whether adequate consideration is given to relevant chemistry effects within the UK ABWR fault analysis. I have also assessed the adequacy of the engineered safety systems which deliver chemistry-related claims and functions. For the most part I am satisfied that they are adequate, although some require further development. Where this is the case and where it is proportionate to do so, these have been captured in the Assessment Findings and Minor Shortfalls I have raised in this topic.
1072. It is also notable that Hitachi-GE has identified a number of important design changes for UK ABWR, related to accident chemistry. They include adding a FCVS, S/P pH control system, and minimising the quantity of halogenated cables inside the reactor containment. Proposing these changes for the UK ABWR generic design therefore demonstrates Hitachi-GE has given an appropriate consideration to the very important requirement in the UK health and safety regulatory context, to reduce relevant risks SFAIRP.

4.8 Overall Safety Case

1073. The following section briefly outlines my overall conclusions on the generic safety case presented by Hitachi-GE for the chemistry-related aspects of the UK ABWR design. It does not repeat any of the detailed commentary which is provided earlier, but attempts to summarise a number of common themes when considered against the relevant standards and guidance.

4.8.1 PCSR

1074. The UK ABWR generic PCSR (Ref. 11) is a very important document in the overall safety case hierarchy. It is the top-level document which provides the overall summary of the relevant information in the documentation which sits beneath it. My assessment in the preceding sections has presented and considered the chemistry claims made by Hitachi-GE in the PCSR (Ref. 11). This section presents my assessment of the PCSR against the expectations of SAP ECH.4 and NS-TAST-GD-051, *The Purpose, Scope and Content of Nuclear Safety Cases* (Refs. 2 and 4). In this regard, my assessment of the PCSR here is focussed on the presentational aspects of the generic safety case. The objective here is to judge whether, from a chemistry perspective, as the head document in the UK ABWR generic safety case, it provides an appropriate summary of the key aspects of the detailed case which sits beneath, and adequately justifies why the UK ABWR design is safe, and relevant risks reduced SFAIRP.

1075. Hitachi-GE worked very hard on developing their PCSR throughout GDA. For Chapter 23 (Ref. 11), this has resulted in what I consider to be a number of strengths:
- The scope covers all of the main topics I expect to see.
 - The link to the categorisation of safety functions and classification of SSCs is very clear for the SSCs of interest to chemistry.
 - The PCSR uses a sophisticated system to provide unique identification numbers for each of the claims made. This means the traceability of the claims through the different PCSR chapters and the lower-level generic safety case documentation is relatively straightforward. The other benefit this brings is that it makes the chemistry chapter itself more self-consistent, providing clear links to the relevant sub-sections of the document where specific claims are made.
 - Links to other PCSR chapters which contain relevant information are very good. This is particularly important for a topic like chemistry, which by its nature, is very cross-cutting.
1076. Hitachi-GE's approach to producing Chapter 23 of the PCSR (Ref. 11) is based on a mixture of a free-text narrative style, coupled with a more formal application of a claims, arguments and evidence approach. This leads to formal links between high-level safety functions, "top claims" and claims. One of the consequences of this approach is that the claims presented are relatively rigid and high-level. Nevertheless, my overall view is that for a generic safety case, an appropriate balance between these two styles has more or less been struck.
1077. There are other aspects though, mainly associated with the overall usability of Chapter 23 of the PCSR (Ref. 11), which I consider will require further improvement by a future licensee, for them to be able to achieve a document which is pitched at the right level and which is suitably operations focussed. The areas I have identified during GDA which will require improvements to be made include:
- There is a general lack of description of how the plant will be operated and what the consequences are for chemistry.
 - The document is very repetitive and verbose in places and reads more like a technical literature review of BWR chemistry matters, rather than an objective safety case document which is specific to the safety requirements of the design.
 - The document is particularly high-level, and therefore several aspects which I would expect a PCSR document covering the operational chemistry of a reactor to provide, are not transparent. In effect, the balance between what detailed information from the supporting documents the UK ABWR generic PCSR (Ref. 11) is summarised from, still needs further work.
 - The "*ALARP summary*" section of Chapter 23 of the PCSR (Ref. 11) could be improved in terms of describing the considerable efforts that have been made to demonstrate that relevant risks for UK ABWR are reduced SFAIRP.
1078. One other important matter is also worth raising. The early drafts of Chapter 23 of the PCSR (Ref. 11) originally contained information on source term and accident chemistry matters. Hitachi-GE decided, quite sensibly, that this was better placed elsewhere in the UK ABWR generic PCSR (Ref. 16 and 17). However, having reviewed these aspects of the wider PCSR, I am not content that this is achieved adequately. The information on these topics is very limited, for example, all that is provided on accident chemistry is a reference to Ref. 157. I do not consider this aspect of the UK ABWR generic PCSR (Ref. 11, 16 and 17) is adequate.
1079. Considering the overall balance of the strengths and areas for improvement I have identified, when judged against the expectations of SAP SC. 4 and relevant guidance contained in NS-TAST-GD-051 (Ref. 4), from a chemistry perspective, I am satisfied that the PCSR is adequate for GDA. In reaching this conclusion it is important to stress

that given the areas for improvement I have identified above. There is therefore a significant amount of work to do for a future licensee to turn the UK ABWR generic PCSR (Ref. 11) into a fit-for-purpose, operationally focussed safety case document for UK ABWR. The extent of the work required should not be underestimated.

4.8.2 Operating Rules

1080. ORs are an important output from the safety case. They represent a summary of the limits and conditions the safety case has defined and justified, which the plant is safe to operate within. They are particularly relevant to the operating chemistry requirements, many of which can be considered as ORs. Hitachi-GE placed considerable effort into the development of these throughout GDA and they are now an important part of the UK ABWR generic safety case, which the future licensee can develop further.

1081. Based on the preceding sections, many aspects of my assessment considered specific aspects of the chemistry-based ORs defined for UK ABWR in the generic safety case, where improvements will be required moving forward. These mainly relate to detailed technical aspects, their completeness, or applicability to the design. This means there is a considerable amount of work to do for this topic moving forward, as captured by Assessment Finding, **AF-ABWR-RC-02**. There are some more generic aspects of the approach to ORs that are worthy of mention. Many of these are for a future licensee to decide, hence were not within the scope of my assessment for GDA. They include:

- The use of control and diagnostic parameters is aligned with international good practice and guidance.
- Setting “*Action Levels*”, with defined actions and completion times, as well as “*expected values*” is a challenging, but commendable target.
- The approach suggested by Hitachi-GE in the generic safety case for where chemistry-based ORs should feature in the hierarchy of ORs does not appear to meet UK regulatory expectations. At present insufficient chemistry parameters are included within the highest level of ORs. Most are identified as “*low hazard operating rules*”, which at this stage, I do not agree with.
- The setting of values associated with the various limits is often somewhat arbitrary, and are less related to safety or even operational matters. All limits should be founded on a firm basis.
- At present the radiochemistry-related parameters are separated from chemistry. This distinction is somewhat artificial, and creates potential error traps, as in reality these measurements would be undertaken by similar personnel. This approach should be considered.
- There are a number of chemistry related “conditions” that are not represented within the water quality controls, but could directly impact them. The approach to capturing these needs careful consideration.

4.8.3 Supporting Documentation

1082. The first and most important thing to note is that I am generally satisfied that the information contained in Hitachi-GE’s lower level generic safety case documentation (mainly Topic Reports) contains an adequate generic safety case, from a chemistry perspective. The structure and hierarchy of the safety case documents is clear. Importantly, the level of detail and information does increase further down the hierarchy, as is expected. The supporting documentation approach adopted by Hitachi-GE works well. However, there are some aspects which could be further strengthened as the UK ABWR design and safety case progresses:

- The links between the operating chemistry and engineering documents is less clear. This was resolved for GDA by providing specific “*design justification*” TRs

(Ref. 18 and 19). Whether this separation should be retained moving forward should be considered.

- Some parts of the generic safety case contain large numbers of documents in their own right, for example for materials selection and source terms. In GDA, the use of summary type documents has helped provide the necessary clarity. This approach would also appear to be beneficial to other aspects of the UK ABWR safety case, for example, accident chemistry.
- A significant proportion of the OPEX supporting the UK ABWR design is for Japanese plants, including ABWRs. This information is of particular importance to chemistry, which is an operationally focussed topic. This is a valuable resource for decision making. However, on occasions during GDA, this restricted Hitachi-GE. The future licensee would benefit from expanding this approach, such that the most relevant and up-to-date OPEX can be drawn upon, irrespective of its source.

4.8.4 ALARP Considerations

1083. Demonstrating that risks are reduced SFAIRP is a key aspect in any nuclear safety case. From the preceding technical sections of my assessment it is clear that Hitachi-GE has applied this thinking to the UK ABWR design and safety case, and has incorporated a number of changes and modifications as a result. The approach adopted is consistent; applying the approach is where the greatest variation lies. However, as the outcome is often reasonable, the intent of NS-TAST-GD-005, *Guidance on the Demonstration of ALARP* (Ref. 4), to provide fit-for-purpose ALARP demonstrations, is met.

4.8.5 Use of Claims, Arguments and Evidence Approach

1084. One of the most important decisions made by Hitachi-GE in developing the UK ABWR generic safety case was the application of a formalised Claims, Arguments and Evidence structure. As described above, there are benefits with this approach, but also some detriments. The most obvious is the complexity of maintaining such a large number of links, but also the tendency for increasing the repetition and size of the documents as a result. Overall through, when considering the generic safety case as a whole, I am content that the net result is positive. This is because a suitable compromise has been reached between the formality and rigour of the application, and the use of a more narrative style.

4.9 Comparison with standards, guidance and relevant good practice

1085. The standards and guidance considered as part of my assessment are defined in Section 2, and included in Annexes 1 to 3. I have considered these throughout my assessment. The foremost standards considered for this assessment were the relevant SAPs (Ref. 2). A summary of my judgement against the most relevant of these is provided below:

- SC.2 to 4 and SC.5 relate to the production of an adequate safety case. I am content that Hitachi-GE has largely met the intent of these as part of the submissions provided for GDA. Developing an adequate generic safety case is one of the major achievements made by Hitachi-GE throughout the GDA process. There are a number of areas where further work will be required by the future licensee, but the generic case presented represents a good starting point and broadly meets my expectations for GDA.
- EKP.1 to 5 are key principles that apply to engineering systems. On the basis of my sample related to chemistry, I am satisfied that the generic safety case considers these matters to a degree which is commensurate with a generic design. These will need to be considered further as the detailed design develops, but the structured approach developed by Hitachi-GE will aid the

- future licensee in this regard, including for ECS.1 and 2, regarding categorisation and classification.
- EAD.1 to 4 and the EMC principles relate to ageing and degradation and metal components and structures. Owing to the inherent hazards for BWR technology, the interrelation between materials and the operating chemistry is of great importance. This is reflected in the submissions made by Hitachi-GE, which provide a structured consideration of these matters. While there are several areas where additional developments are required, these can be satisfactorily resolved by a future licensee, without undermining the fundamental design proposed for GDA. A number of positive changes have been made in this area, which reflects the constructive consideration given to these requirements by Hitachi-GE.
 - ECH.1 to 4 relates specifically to the chemistry aspects of safety cases. I am satisfied that Hitachi-GE has given due consideration to these expectations during GDA, in particular the expectations of ECH.1, on systematically examining the chemistry effects important to safety. This is best exemplified by the decision to change the RCS operating chemistry for UK ABWR, which also demonstrates the consideration given to SAP ECH.2. I judge that further justification is required for some of the chemistry control and monitoring aspects (ECH.3 and 4) defined by the generic safety case, but I am happy the intent for these SAPs has been considered sufficiently for GDA.
 - The fault analysis related SAPs (FA.2, 7, 15 and 16; AV.1 to 3 and 6) exemplify the expectations that chemistry-related impacts on faults are adequately considered. Hitachi-GE put a significant amount of effort into this topic throughout GDA, and the generic safety case improved greatly as a result. While I have identified a number of gaps against these SAPs, these are dealt with by either Assessment Findings or minor shortfalls. I am content that sufficient evidence has been provided for a generic safety case.

4.10 GDA Issues

1086. During my assessment I have not identified any matters which I consider to be significant enough to be GDA Issues, and prevent the issue of a DAC.

4.11 Assessment Findings

1087. In line with the ONR guidance (Ref. 5), during my assessment 26 items were identified for a future licensee to take forward in its site-specific safety submissions. Details of these are contained in Annex 8.
1088. 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.

4.12 Minor Shortfalls

1089. In line with ONR guidance (Ref. 5), during my assessment, 12 items were identified as being Minor Shortfalls with the generic safety case. They are not considered significant enough to require specific action to be taken by the future licensee. All of the Minor Shortfalls are listed in Annex 9.

5 CONCLUSIONS

1090. This report presents the findings of my Step 4 chemistry assessment of the Hitachi-GE UK ABWR generic design and safety case.
1091. For the purpose of this assessment chemistry is taken to be; “*the chemistry of the design including the effects of coolant chemistry on reactivity, pressure boundary integrity, fuel and core component integrity, fuel storage in cooling pools, radioactive waste generation and radiological doses to workers*”. My assessment therefore considered both the technical basis for the relevant chemistry effects, but also the engineering of the systems which deliver those safely, as part of the design. This therefore covers the underlying basis for:
- any requirement or constraint placed on the operating chemistry of the plant which must be met in order to allow the plant to be operated safely;
 - any chemistry related functional requirement which must be met to ensure that the plant is operated within its design basis; and
 - any effect or consequence of chemistry during operations, during faults or during severe accidents, which must be understood and controlled in order to ensure the safety of workers and the public.
1092. During GDA, Hitachi-GE provided a large suite of safety case documentation to justify the UK ABWR generic design is capable of meeting UK regulatory expectations and relevant standards and guidance. A large part of this undertaking was as a direct consequence of responding to the one RI and multiple ROs I raised during GDA. Hitachi-GE’s responses to all of these met their intent, and the RI and all ROs raised in the chemistry topic during GDA, were satisfactorily resolved. The scale of Hitachi-GE’s undertaking was significant, and has resulted in a generic safety case which contains a much improved scope, breadth and depth of supporting evidence. Major additions to the safety justifications relate to the operating chemistry, source terms, material choices, radiolysis gases and accident chemistry assumptions, amongst others.
1093. The most significant impact on the safety case for chemistry is the decision to change the at-power operating chemistry to HWC, OLNLC and DZO injection. This is in recognition of the extended 60 year lifetime proposed for UK ABWR and the overriding legal requirement in the UK of reducing risks SFAIRP. This is the most obvious, and important example of Hitachi-GE considering this fundamental requirement. As well as the direct impact on chemistry, this choice has implications for many other aspects of the generic design and safety case from materials of construction, to systems engineering and Operating Rules.
1094. In addition to the change in operating chemistry, throughout the production of the generic safety case, a number of enhancements have been identified to the reference plant design. From a chemistry perspective, these are important changes and further reinforce the demonstration that all reasonably practicable steps have been taken to reduce risks. Some of the most notable examples include:
- Bulk material changes in the CUW, CFDW and HD systems which have benefits for materials degradation threats and plant dose rates.
 - Further reductions in the use of high-cobalt alloys, minimisation of trace cobalt levels, adoption of surface treatments or operational practices to reduce the generation or deposition of radioactivity.
 - Plant lay-out and design changes to further reduce the possibility of radiolysis gas accumulation or deflagrations (during normal operations), and;
 - Including additional engineered safety features for certain fault and/or accident conditions. For example, a FCVS, S/P pH control system and CIPS.

1095. I have assessed this extended generic safety case against the applicable expectations of the SAPs and relevant international guidance. I am satisfied that, overall:

- Hitachi-GE has identified a suitable set of claims on the operating chemistry, for all modes of operation including commissioning, and has provided sufficient supporting evidence to demonstrate that the claims can be achieved by the generic design.
- Hitachi-GE has justified that the engineered design of those systems which provide chemistry-related functions are adequate to enable the functions to be delivered in a reliable and safe manner, consistent with the requirements of the generic safety case.
- Hitachi-GE has integrated chemistry within the wider aspects of the generic safety case, such as categorisation and classification and the fault schedule, to a degree that is acceptable for this stage in the design process.
- The definition and justification of an appropriate source term for the plant proved to be a particular challenge for Hitachi-GE, but they responded with a comprehensive suite of documentation which presents a fit-for-purpose definition of the plant radioactivity sources to use in the generic safety case.
- The major chemistry parameters which would be expected to form part of the plant Operating Rules have been identified, and while significant further work will be required by a future licensee to fully define and substantiate the associated limits for many of them, the basic approach is sound.
- Where chemistry-related assumptions are used in the supporting analysis, they have been identified and the adequacy or sensitivity to these has been demonstrated to a proportionate degree, and related uncertainty has been treated in a reasonable manner.
- An appropriate balance has been achieved in demonstrating that overall plant risks have been reduced SFAIRP for those aspects where chemistry can have an influence.
- The scope, structure and content of the generic safety case broadly meets my expectations for this stage of the project, and an adequate PCSR has been produced which provides a summary and links to the underlying evidence. I should note here though that the PCSR is fit-for-purpose for a generic safety case, but will require a substantial amount of work from a future licensee to develop it further into a document which is suitable to support eventual plant operations.
- Collectively, the documentation contained within the MDSL provides an appropriate starting point for a future license to develop the UK ABWR safety case and design to allow the transition to operations.

1096. However, some aspects of the UK ABWR generic design and safety case will merit particular enhanced attention moving forward, once GDA concludes. They include:

- Developing the full suite of chemistry-related Operating Rules, including the definition of appropriate limits and how these are linked to the safety case.
- Providing additional evidence and justifications for aspects of the UK ABWR design or safety case which are not sufficiently detailed at this stage, or where my assessment has identified further work is required to substantiate the decisions taken at this time.
- Justification for when important decisions are required by the future licensee for how they wish to operate the plant and the impacts this has on safety.

1097. To conclude, I am satisfied with the claims, arguments and evidence laid down within the generic PCSR and supporting documentation, at this stage of the project. I consider that from a chemistry view point, the Hitachi-GE UK ABWR design is suitable for construction in the UK, subject to future permissions and permits being secured.

1098. As a consequence of my assessment 26 Assessment Findings (Annex 8) were identified; these are for the future licensee to consider and take forward in their site-specific safety submissions. These matters do not undermine the UK ABWR generic safety submissions or design, and require future licensee input or decisions to resolve them.
1099. Similarly, I also raised 12 Minor Shortfalls (Annex 9). These are less significant matters than Assessment Findings, and represent aspects I have noted during my assessment of the UK ABWR generic safety case and design, which merit some further consideration by a future licensee.

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Annex 1

Safety Assessment Principles

SAP No	SAP Title	Description
The Regulatory Assessment of Safety Cases		
SC.2	Safety case process outputs	The safety case process should produce safety cases that facilitate safe operation.
SC.3	Lifecycle aspects	For each lifecycle stage, control of the hazard should be demonstrated by a valid safety case that takes into account the implications from previous stages and for future stages.
SC.4	Safety case characteristics	A safety case should be accurate, objective and demonstrably complete for its intended purpose.
SC.5	Optimism, uncertainty and conservatism	Safety cases should identify areas of optimism and uncertainty, together with their significance, in addition to strengths and any claimed conservatism.
Engineering principles: Key principles		
EKP.1	Inherent safety	The underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility.
EKP.2	Fault tolerance	The sensitivity of the facility to potential faults should be minimised.
EKP.3	Defence in depth	Nuclear facilities should be designed and operated so that defence in depth against potentially significant faults or failures is achieved by the provision of multiple independent barriers to fault progression.
EKP.4	Safety function	The safety function(s) to be delivered within the facility should be identified by a structured analysis.
EKP.5	Safety measures	Safety measures should be identified to deliver the required safety function(s).
Engineering principles: safety classification and standards		
ECS.1	Safety categorisation	The safety functions to be delivered within the facility, both during normal operation and in the event of a fault or accident, should be identified and then categorised based on their significance with regard to safety.
ECS.2	Safety classification of structures, systems and components	Structures, systems and components that have to deliver safety functions should be identified and classified on the basis of those functions and their significance to safety.
Engineering principles: design for reliability		
EDR.2	Redundancy, diversity and segregation	Redundancy, diversity and segregation should be incorporated as appropriate within the designs of structures, systems and components.

Annex 1

Safety Assessment Principles

SAP No	SAP Title	Description
Engineering principles: reliability claims		
EDR.2	Engineered safety measures	Where reliable and rapid protective action is required, automatically initiated, engineered safety measures should be provided.
Engineering principles: Commissioning		
ECM. 1	Commission testing	Before operating any facility or process that may affect safety it should be subject to commissioning tests defined in the safety case.
Engineering principles: maintenance, inspection and testing		
EMT.1	Identification of requirements	Safety requirements for in-service testing, inspection and other maintenance procedures and frequencies should be identified in the safety case.
Engineering principles: Ageing and degradation		
EAD. 1	Safe working life	The safe working life of structures, systems and components that are important to safety should be evaluated and defined at the design stage.
EAD. 2	Lifetime margins	Adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on structures, systems and components.
EAD. 3	Periodic measurement of material properties	Where material properties could change with time and affect safety, provision should be made for periodic measurement of the properties.
EAD. 4	Periodic measurement of parameters	Where parameters relevant to the design of plant could change with time and affect safety, provision should be made for their periodic measurement.
Engineering principles: Integrity of metal components and structures		
EMC. 2	Use of scientific and technical issues	The safety case and its assessment should include a comprehensive examination of relevant scientific and technical issues, taking account of precedent when available.
EMC. 3	Evidence	Evidence should be provided to demonstrate that the necessary level of integrity has been achieved for the most demanding situations identified in the safety case.
EMC. 13	Materials	Materials employed in manufacture and installation should be shown to be suitable for the purpose of enabling an adequate design to be manufactured, operated, examined and maintained throughout the life of the facility.

Annex 1

Safety Assessment Principles

SAP No	SAP Title	Description
EMC. 21	Safe operating envelope	Throughout their operating life, components and structures should be operated and controlled within defined limits and conditions (operating rules) derived from the safety case.
EMC. 22	Material compatibility	Materials compatibility for components should be considered for any operational or maintenance activities.
EMC. 24	Operation	Facility operations should be monitored and recorded to demonstrate compliance with, and to allow review against, the safe operating envelope defined in the safety case (operating rules).
Engineering principles: Control of nuclear matter		
ENM. 2	Provisions for nuclear matter brought onto, or generated on, the site	Nuclear matter should not be generated on the site, or brought onto the site, unless sufficient and suitable arrangements are available for its safe management on the site.
ENM. 3	Transfers and accumulation of nuclear matter	Unnecessary or unintended generation, transfer or accumulation of nuclear matter should be avoided.
Engineering principles: Chemical Engineering		
EPE.1	Design and operation	The design and operation of nuclear chemical processes and facilities should be fault tolerant and ensure safety functions are delivered with suitable capability and sufficient reliability and robustness.
Engineering principles: Chemistry		
ECH. 1	Safety cases	Safety cases should, by applying a systematic process, address all chemistry effects important to safety.
ECH. 2	Resolution of conflicting chemical effects	Where the effects of different chemistry parameters conflict with one another, the safety case should demonstrate that an appropriate balance for safety has been achieved.
ECH. 3	Control of chemistry	Suitable and sufficient systems, processes and procedures should be provided to maintain chemistry parameters within the limits and conditions identified in the safety case.
ECH. 4	Monitoring, sampling and analysis	Suitable and sufficient systems, processes and procedures should be provided for monitoring, sampling and analysis so that all chemistry parameters important to safety are properly controlled.
Engineering principles: Safety systems		
ESS. 1	Provision of safety systems	All nuclear facilities should be provided with safety systems that reduce the frequency or limit the consequences of fault sequences, and that achieve and maintain a defined stable, safe state.

Annex 1

Safety Assessment Principles

SAP No	SAP Title	Description
ESS. 2	Safety system specification	The extent of safety system provisions, their functions, levels of protection necessary to achieve defence in depth and reliability requirements should be specified.
ESS. 3	Monitoring of plant safety	Adequate provisions should be made to enable the monitoring of the facility state in relation to safety and to enable the taking of any necessary safety actions during normal operational, fault, accident and severe accident conditions.
Engineering principles: Reactor core		
ERC. 1	Design and operation of reactors	The design and operation of the reactor should ensure the fundamental safety functions are delivered with an appropriate degree of confidence for permitted operating modes of the reactor.
ERC. 3	Stability in normal operation	The core should be stable in normal operation and should not undergo sudden changes of condition when operating parameters go outside their permitted range.
ERC. 4	Monitoring of parameters important to safety	The core should be designed so that parameters and conditions important to safety can be monitored in all operational and design basis fault conditions and appropriate recovery actions taken in the event of adverse conditions being detected.
Engineering principles: Heat transport systems		
EHT. 4	Failure of heat transport system	Provisions should be made in the design to prevent failures of the heat transport system that could adversely affect the heat transfer process, and to maintain the facility in a safe condition following such failures.
EHT. 5	Minimisation of radiological doses	The heat transport system should be designed to minimise radiological doses.
Fault analysis		
FA. 2	Identification of initiation faults	Fault analysis should identify all initiating faults having the potential to lead to any person receiving a significant dose of radiation, or to a significant quantity of radioactive material escaping from its designated place of residence or confinement.
FA.7	Consequences	Analysis of design basis fault sequences should use appropriate tools and techniques and be performed on a conservative basis to demonstrate that consequences are ALARP.
FA.15	Scope of severe accident analysis	Fault states, scenarios and sequences beyond the design basis that have the potential to lead to a severe accident should be analysed.
FA.16	Use of severe accident analysis	Severe accident analysis should be used in the consideration of further risk-reducing measures.
AV.1	Theoretical models	Theoretical models should adequately represent the facility and site.

Annex 1

Safety Assessment Principles

SAP No	SAP Title	Description
AV.2	Calculation methods	Calculation methods used for the analyses should adequately represent the physical and chemical processes taking place.
AV.3	Use of data	The data used in the analysis of aspects of plant performance with safety significance should be shown to be valid for the circumstances by reference to established physical data, experiment or other appropriate means.
AV.6	Sensitivity studies	Studies should be carried out to determine the sensitivity of the analysis (and the conclusions drawn from it) to the assumptions made, the data used and the methods of calculation.
Radioactive waste management		
RW.2	Generation of radioactive waste	The generation of radioactive waste should be prevented or, where this is not reasonably practicable, minimised in terms of quantity and activity.

Annex 2

Technical Assessment Guides

TAG Ref	TAG Title
NS-TAST-GD-004	Fundamental principles.
NS-TAST-GD-005	ONR guidance on the demonstration of ALARP (as low as reasonably practicable).
NS-TAST-GD-016	Integrity of metal components and structures.
NS-TAST-GD-020	Containment for reactor plants.
NS-TAST-GD-021	Containment for chemical plants.
NS-TAST-GD-022	Ventilation.
NS-TAST-GD-023	Control of processes involving nuclear matter.
NS-TAST-GD-034	Transient analysis for DBAs in nuclear reactors.
NS-TAST-GD-035	The limits and conditions for nuclear plant safety.
NS-TAST-GD-037	Heat transport systems.
NS-TAST-GD-038	Radiological protection.
NS-TAST-GD-042	Validation of computer codes and calculation methods.
NS-TAST-GD-051	The purpose, scope and content of nuclear safety cases.
NS-TAST-GD-081	Safety aspects specific to storage of spent nuclear fuel.
NS-TAST-GD-088	Chemistry of Operating Civil Nuclear Reactors.

Annex 3

National and International Standards and Guidance

Reference	Title
IAEA	
Specific Safety Requirements No. SSR-2/1. IAEA. 2012.	Safety of Nuclear Power Plants: Design Safety Standard – Specific Safety Requirements.
Specific Safety Requirements No. SSR-2/2. IAEA. July 2011.	IAEA Safety Standards, Safety of Nuclear Power Plants: Commissioning and Operation.
Specific Safety Guide No. SSG-13. IAEA. July 2011.	IAEA Safety Standards, Chemistry Programme for Water Cooled Nuclear Power Plants.
Safety Guide No. NS-G-1.12. IAEA. April 2005.	IAEA Safety Standards, Design of the Reactor Core for Nuclear Power Plants.
Specific Safety Guide No. NS-G-1.4. IAEA. 2003.	IAEA Safety Standards, Design of Fuel Handling and Storage Systems for Nuclear Power Plants.
WENRA	
Western European Nuclear Regulators' Association. Reactor Safety Reference Levels, September 2014	Issue E: Design Basis Envelope of Existing Reactors, in particular; "2.2 The design shall prevent as far as practicable: - challenges to the integrity of the barriers; - failure of a barrier when challenged; - failure of a barrier as consequence of failure of another barrier."
	Issue H: Operational limits and conditions, in particular; "1.1 OLCs shall be developed to ensure that plants are operated in accordance with design assumptions and intentions as documented in the SAR. 1.2 The OLCs shall define the conditions that must be met to prevent situations that might lead to accidents or to mitigate the consequences of accidents should they occur."
	Issue I: Ageing Management, in particular; "1.1. The operating organisation shall have an Ageing Management Programme ⁴¹ to identify all ageing mechanisms relevant to structures, systems and components (SSCs) important to safety, determine their possible consequences, and determine necessary activities in order to maintain the operability and reliability of these SSCs."

Annex 3

National and International Standards and Guidance

Reference	Title
	Issue K: Maintenance, in-service inspection and functional testing, in particular: <i>"1.1 The licensee shall prepare and implement documented programmes of maintenance, testing, surveillance, and inspection of SSCs important to safety to ensure that their availability, reliability, and functionality remain in accordance with the design over the lifetime of the plant. They shall take into account operational limits and conditions and be re-evaluated in the light of experience."</i>
WENRA Statement on Safety objectives for new nuclear power plants, November 2010	O2: Accidents without core melt, in particular; <i>"reducing, so far as reasonably achievable, the release of radioactive material from all sources"</i>
	O3: Accidents with core melt, in particular; <i>"reducing potential releases to the environment from accidents with core melt"</i>
	O6: Radiation protection and waste management, in particular; <i>"reducing as far as reasonably achievable by design provisions, for all operating states, decommissioning and dismantling activities :</i> - <i>individual and collective doses for workers;</i> - <i>radioactive and non-radioactive discharges to the environment;</i> - <i>quantity and activity of radioactive waste."</i>
Other	
BWRVIP-190	BWR Vessel and Internals Project, BWR Water Chemistry Guidelines – 2008 Revision. www.epri.com
VGB-R 401 J	VGB Guideline for the Water in Nuclear Power Stations with Light Water Reactors (BWR). www.vgb.org

Annex 4

Chemistry Claims in Chapter 23 of the UK ABWR Generic PCSR

Top Claim		Claim	
Reactor Coolant System			
RC SC1	The UK ABWR reactor chemistry regime will contribute to the maintenance of the integrity of the Reactor Pressure Vessel (RPV) and SSCs in contact with the reactor coolant by controlling within the limits and conditions.	RC SC1.1	Stress Corrosion Cracking (SCC) propagation will be mitigated by appropriate Electrochemical Corrosion Potential Control by HWC and OLNC.
		RC SC1.2	Corrosion (such as SCC, FAC) and corrosion rates will be mitigated by control of impurities by operation of the clean-up systems.
		RC SC 1.3	Stress Corrosion Cracking (SCC) propagation will be mitigated during start-up by minimising the reactor water oxygen.
RC SC2	The UK ABWR reactor chemistry regime will contribute to the maintenance of the integrity of the associated SSCs which contact with reactor water outside of the RPV by controlling within the limits and conditions.	RC SC1.1	Stress Corrosion Cracking (SCC) propagation will be mitigated by appropriate Electrochemical Corrosion Potential Control by HWC and OLNC.
		RC SC1.2	Corrosion (such as SCC, FAC) and corrosion rates will be mitigated by control of impurities by operation of the clean-up systems.
		RC SC2.1	Corrosion such as Flow Accelerated Corrosion (FAC) and pitting corrosion will be reduced by O ₂ control.
		RC SC2.2	Impurities such as silica will be minimised to ALARP to reduce the scale adhesion in turbine system by operation of the CUW.
RC SC3	The UK ABWR reactor chemistry regime will contribute to the maintenance of the fuel integrity by control within limits and conditions.	RC SC3.1	The specified levels and timing of noble metal injection will have no detrimental effect on fuel integrity.
		RC SC3.2	Zinc injection within the upper LCO will have no adverse effect on fuel integrity.
		RC SC3.3	Iron (Fe) concentration in feedwater within the LCO will have no detrimental effect on the fuel.
		RC SC3.4	The application of HWC and management of the hydrogen concentration in the feedwater to LCO will ensure there is no detrimental effect on the fuel as a result of hydriding.
		RC SC3.5	Adequate control of metal impurities especially Copper in feedwater will have no detrimental effect on the fuel.
		RC SC3.6	Foreign materials which might cause fretting will be minimised to ALARP before 1st fuel loading.
RC SC4	The UK ABWR pipework and system design will mitigate build-up of hydrogen concentration and mitigate flammability risk and radioactive release from reactor coolant system.	RC SC4.1	Pipework and system design will minimise the build-up of hazardous concentrations of hydrogen.

Annex 4

Chemistry Claims in Chapter 23 of the UK ABWR Generic PCSR

Top Claim		Claim	
RC SC5	The UK ABWR reactor chemistry regime will ensure that the source term radiological dose to the worker is ALARP by optimising materials selection, operating chemistry and operating practices.	RC SC5.1	The source Term will be minimised to ALARP in association with material selection.
		RC SC5.2	The source term will be minimised to ALARP in association with operating practices.
		RC SC5.3	The source term will be minimised to ALARP in association with operating chemistry.
		RC SC5.4	Commissioning activities will be optimised to ensure that the longer term radiation exposure will be minimised to ALARP.
		RC SC5.5	Source of activation products loaded from condensate and feedwater system to the reactor will be minimised to ALARP by the operation procedure of the condensate purification system.
		RC SC6.1	The amount of the activation product (^{16}N) transferred to the steam will be maintained low by the control of HWC+OLNC.
RC SC6	The UK ABWR reactor chemistry regime will ensure that the radionuclide releases and exposure to public is ALARP.	RC SC6.1	The amount of the activation product (^{16}N) transferred to the steam will be maintained low by the control of HWC+OLNC.
RC SC7	The UK ABWR reactor chemistry regime will ensure that the radionuclide releases and exposure to worker is ALARP.	RC SC6.1	The amount of the activation product (^{16}N) transferred to the steam will be maintained low by the control of HWC+OLNC.
		RC SC7.1	Radionuclides in the reactor water will be kept below levels that results in increased radionuclide release and exposure, and minimised to ALARP in normal operations by CUW and OG system operations.
Spent Fuel Pool			
RC SC8	The UK ABWR Spent Fuel Storage Pool chemistry regime will contribute to the maintenance of the integrity of the structure and liner in the pool by operating within the limits and conditions.	RC SC8.1	Impurities will be kept below levels that may result in increased corrosion rates and minimised to ALARP in normal operations by the FPC.
RC SC9	The UK ABWR Spent Fuel Storage Pool chemistry regime will contribute to the maintenance of the spent fuel integrity by operating within the limits and conditions.		

Annex 4

Chemistry Claims in Chapter 23 of the UK ABWR Generic PCSR

Top Claim		Claim	
RC SC10	The UK ABWR Spent Fuel Storage Pool chemistry regime will ensure that the radionuclide releases and exposure to worker is ALARP.	RC SC10.1	Radionuclides in the Spent Fuel Storage Pool will be minimised to ALARP by the FPC and CUW.
RC SC11	The UK ABWR Spent Fuel Storage Pool chemistry regime will ensure that the radionuclide releases and exposure to public is ALARP.		
Suppression Pool			
RC SC12	The UK ABWR Suppression Pool chemistry regime will contribute to the maintenance of the integrity of structure and liner in the pool by operating within the limits and conditions.	RC SC12.1	Impurities will be kept below levels that may result in increased corrosion rates and minimised to ALARP in normal operations by the SPCU.
Sampling and Monitoring			
RC SC13	The UK ABWR sampling and monitoring systems will ensure that the plant is operated within the limits and conditions.	RC SC13.1	Chemistry and radiochemistry control parameters will be sampled and monitored in accordance with the sampling philosophy.
Standby Liquid Control system			
RC SC14	The UK ABWR Standby Liquid Control System chemistry will ensure safe shutdown of the reactor in the event of Anticipated Transient Without Scram (ATWS) by injecting neutron absorber solution into the reactor water.	RC SC14.1	Sufficient and available sodium pentaborate solution is stored and maintained in an available state in the SLC.
Component Cooling Water systems			
RC SC15	The chemistry of the component cooling water contributes to minimise the corrosion of its system materials to maintain their integrity and heat transfer function by ensuring a corrosion controlled environment when operated within the limits and conditions.	RC SC15.1	Corrosion of SSCs of carbon steel, stainless steel and copper will be minimised by the addition of nitrite based corrosion inhibitors.

Annex 4

Chemistry Claims in Chapter 23 of the UK ABWR Generic PCSR

Top Claim		Claim	
Component Cooling Water systems			
RC SC16	The water quality produced and stored by the MUWC and CST is very pure and therefore it will not contribute to the equipment material degradation that it comes into contact with.	RC SC16.1	Water quality control within specified limits in the CST will ensure the water quality of the RCS and associated systems.
		RC SC16.2	Water quality control within specified limits in the MUWP (Purified Water Storage Tank water) will ensure the water quality of the RCS and associated systems.
Component Cooling Water systems			
RC SC17	The UK ABWR FLSS water quality will ensure the system will provide cooling water to prevent core damage when required.	RC SC17.1	The water quality will be maintained and will not degrade the integrity of the structural material of the FLSS during the standby mode.

Annex 5

UK ABWR Water Quality Specifications

Notes:

Values in **Yellow** represent “control” parameters, remainder are “diagnostic” parameters

“-“ denotes no value is given

“N/A” denotes not applicable to this mode of operation

Parameter	Units	Limit (Expected) Value			
		Start-up	At-power	Shutdown	Outage
Reactor Water					
Direct Conductivity	$\mu\text{S m}^{-1}$ at 25°C	≤ 600 [redacted]	≤ 600 [redacted]	≤ 600 [redacted]	≤ 600 (-)
Chloride	ppb	≤ 600 [redacted]	≤ 600 [redacted]	≤ 600 [redacted]	≤ 600 (-)
Sulphate	ppb	≤ 600 [redacted]	≤ 600 [redacted]	≤ 600 [redacted]	≤ 600 (-)
Silica	ppb	≤ 9000 [redacted]	≤ 9000 [redacted]	≤ 9000 [redacted]	N/A
Zinc	ppb	≥ 2 [redacted]	≥ 2 [redacted]	≥ 2 [redacted]	N/A
Platinum	g yr^{-1}	N/A	$\geq 20, \leq 920$ [redacted]	N/A	N/A
pH	at 25°C	-	-	-	-
Dissolved Oxygen	ppb	-	-	-	-
Metal Impurities (Fe, Cu, Ni, Cr, Co)	ppb	-	-	-	-
Dissolved Hydrogen	ppb	-	-	-	-
ECP	mV vs. SHE	N/A	-	N/A	N/A
^{131}I (prior to head lift)	Bq cm^{-3}	N/A	N/A	N/A	$\leq 1 \text{ E-}03$ [redacted]
^{133}Xe (prior to head lift)	Bq cm^{-3}	N/A	N/A	- (-)	N/A
^{60}Co	Bq cm^{-3}	N/A	≤ 43.8 [redacted]	N/A	N/A
^{60}Co (prior to flood up)	Bq cm^{-3}	N/A	N/A	N/A	≤ 43.8 [redacted]

Annex 5 Water Quality Specifications

Parameter	Units	Start-up	At-power	Shutdown
Feedwater				
Direct Conductivity	$\mu\text{S m}^{-1}$ at 25°C	- (-)	- [REDACTED]	- (-)
Copper	ppb	≤ 1 [REDACTED]	≤ 1 [REDACTED]	≤ 1 [REDACTED]
Zinc	ppb	≤ 3 [REDACTED]	≤ 3 [REDACTED]	≤ 3 [REDACTED]
Iron	ppb	≤ 50 (≤ 5)	≤ 50 (≤ 1)	≤ 50 (≤ 5)
Dissolved Oxygen	ppb	$\geq 15, \leq 500$ [REDACTED]	$\geq 15, \leq 500$ [REDACTED]	$\geq 15, \leq 500$ [REDACTED]
Dissolved Hydrogen	ppm	$\geq 0.15, \leq 0.5$ [REDACTED]	$\geq 0.15, \leq 0.5$ [REDACTED]	$\geq 0.15, \leq 0.5$ [REDACTED]
Ni, Cr, Co	ppb	-	-	-
Condensate Water				
Direct Conductivity (CF inlet)	$\mu\text{S m}^{-1}$ at 25°C	- (-)	- (-)	- (-)
Direct Conductivity (CD outlet)	$\mu\text{S m}^{-1}$ at 25°C	- (-)	- (-)	- (-)
Dissolved Oxygen (CD outlet)	ppb	N/A	- [REDACTED]	N/A
Copper (CD outlet)	ppb	- (-)	- [REDACTED]	- (-)
Iron (CD outlet)	ppb	- (-)	- [REDACTED]	- (-)
Chloride (CD outlet)	ppb	N/A	- (-)	N/A
Sulphate (CD outlet)	ppb	N/A	- (-)	N/A
Ni, Cr, Co (CD outlet)	ppb	- (-)	- (-)	- (-)
Silica (CD outlet)	ppb	N/A	- (-)	N/A

Annex 5

Water Quality Specifications

Parameter	Units	All modes
Condensate Storage Tank		
Conductivity	$\mu\text{S m}^{-1}$ at 25°C	≤ 600 [REDACTED]
Chloride	ppb	≤ 600 [REDACTED]
Sulphate	ppb	≤ 600 [REDACTED]
pH	at 25°C	-
Total Organic Carbon	ppb	≤ 5000 [REDACTED]
Silica	ppb	-
Metal Impurities (Fe, Cu, Ni, Cr, Co)	ppb	-

Annex 5

Water Quality Specifications

Parameter	Units	All modes
Spent Fuel Pool, Dryer-Separator Pit and Reactor Well		
Direct Conductivity	$\mu\text{S m}^{-1}$ at 25°C	≤ 600 [REDACTED]
Chloride	ppb	≤ 600 [REDACTED]
Sulphate	ppb	≤ 600 [REDACTED]
pH	at 25°C	-
Total Organic Carbon	ppb	-
Metal Impurities (Fe, Cu, Ni, Cr, Co)	ppb	-
⁶⁰ Co (spent fuel pool only)	Bq cm ⁻³	≤ 2.86 [REDACTED]
Suppression Pool		
Direct Conductivity	$\mu\text{S m}^{-1}$ at 25°C	≤ 600 [REDACTED]
Chloride	ppb	≤ 600 [REDACTED]
Sulphate	ppb	≤ 600 [REDACTED]
pH	at 25°C	-
Total Organic Carbon	ppb	-
Metal Impurities (Fe, Cu, Ni, Cr, Co)	ppb	-

Annex 5

Water Quality Specifications

Parameter	Units	All modes
Component Cooling Water		
Direct Conductivity	$\mu\text{S m}^{-1}$ at 25°C	-
pH	at 25°C	$\geq 5, \leq 11$ [REDACTED]
Chloride	ppm	≤ 10 [REDACTED]
Insoluble Material	ppm	≤ 2 [REDACTED]
Nitrite	ppm	$\geq 100, \leq 4000$ [REDACTED]
Nitrate	ppm	≤ 200 [REDACTED]
Ammonium	ppm	≤ 10 [REDACTED]

Annex 6

Regulatory Issues / Observations

Reference	Title	Description	Date Closed	Report Section Reference
Regulatory Issue				
RI-ABWR-0001	Definition and Justification for the Radioactive Source Terms in UK ABWR during Normal Operations	The objective of this RI is to state the regulators (ONR and the Environment Agency) expectations with respect to Hitachi-GE providing a suitable and sufficient definition and justification for the radioactive source terms in UK ABWR during normal operations.	October 2016	4.3.3.1
Regulatory Observations				
RO-ABWR-0006	Source Terms	This RO and associated Regulatory Observation Actions (ROA) sets out the regulators' (ONR and Environment Agency) expectations regarding the use of source terms for operational states in UK ABWR during GDA.	April 2017	4.3.3.1
RO-ABWR-0019	UK ABWR Reactor Chemistry Safety Case: Strategy, Plan and Delivery	The objective of this RO is to state ONR's expectations related to the development and delivery of the reactor chemistry safety case for the UK ABWR as part of the GDA submission.	February 2016	4
RO-ABWR-0022	Demonstration that the Primary Cooling System Operating Chemistry Reduces Risks SFAIRP	The choice of operating chemistry for a given system can be a complex decision and there is a need to show that the chosen regime adequately balances the different benefits and detriments that this choice infers and therefore reduces risks SFAIRP.	October 2015	4.3.1.1
RO-ABWR-0034	Demonstrating the Inclusion of a 'Bottom Drain Line' in the UK ABWR Design Achieves Inherent Safety and Reduces Risks SFAIRP	This RO was raised to make clear ONR's expectations regarding Hitachi-GE's justification for the inclusion of a 'bottom drain line' in the UK ABWR design, specifically whether this achieves an inherently safe design which avoids radiological hazards rather than controlling them, and whether its presence reduces risks SFAIRP.	March 2017	4.3.2.2
RO-ABWR-0035	Robust Justification for the Materials Selected for UK ABWR	The choice of materials for a SSC of a nuclear reactor is influenced by many competing factors. Therefore, the justification of the most appropriate material selected for a particular SSC requires a balance to be struck. In reaching that balance, there should be a robust demonstration that all of the relevant risks have been considered and reduced SFAIRP.	September 2017	4.3.2.2

Annex 6

Regulatory Issues / Observations

Reference	Title	Description	Date Closed	Report Section Reference
RO-ABWR-0043	Demonstration of the Adequacy of pH Control in the Suppression Pool During Accident Conditions	This RO was raised to clarify ONR's regulatory expectations regarding what further evidence is required to understand the safety significance, likelihood and potential consequences for pH changes in the suppression pool. The RO also explains how ONR expect this evidence to be used by Hitachi-GE in their demonstration that the UK ABWR design does indeed reduce the risks associated with this phenomenon SFAIRP.	September 2017	4.7.4
RO-ABWR-0044	Demonstration UK ABWR has been Designed to Safely Manage Radiolysis Gases Generated under Normal Operations	The objective of this RO was to make clear ONR's expectations for the approach taken by Hitachi-GE to address the generation, accumulation, management and mitigation of radiolysis gases, and the development of the associated safety justification for the UK ABWR.	April 2017	4.3.6
RO-ABWR-0066	Demonstration of Suitable and Sufficient Consideration of Chemistry Effects in Fault Analysis	The objective of this RO was to state ONR's expectations with respect to Hitachi-GE demonstrating that they have given suitable and sufficient consideration to the chemistry effects important within the fault analysis. This should include proportionate evidence and a clear evaluation of their significance, sensitivity and uncertainty and should reconcile this with the overall UK ABWR safety case.	September 2017	4.7.4
RO-ABWR-0072	Suitable and Sufficient Consideration of Chemistry Control during UK ABWR Commissioning	The objective of this Regulatory Observation (RO) was to state ONR's expectations with respect to Hitachi-GE giving suitable and sufficient consideration of chemistry control during UK ABWR commissioning.	July 2017	4.3.1.3 4.3.2 4.3.3 4.3.4 4.3.5 4.5
RO-ABWR-0073	Robust Demonstration that the Design of the UK ABWR Off-gas System Reduces Risks SFAIRP	The objective of this RO was to state ONR's expectations with respect to Hitachi-GE producing a robust demonstration that the design of the UK ABWR off-gas system reduces risks SFAIRP.	September 2017	4.6.2

Annex 6

Regulatory Issues / Observations

Reference	Title	Description	Date Closed	Report Section Reference
RO-ABWR-0081	UK ABWR Sampling Capabilities	The objective of this Regulatory Observation (RO) was to state ONR's expectations with respect to Hitachi-GE producing a robust justification that the UK ABWR sampling systems are adequate to determine all chemistry and radiochemistry parameters necessary in the interests of safety, during normal operations and faults, in a manner that reduces risks SFAIRP.	September 2017	4.3.5.3 4.7.6

Annex 7

Residual Matters identified during Assessment of RI-ABWR-0001

Number	Description	Response
1	Assessment is required on how the derived source terms are used or applied within particular technical aspects of the safety case	These have been considered by ONR in preparing this assessment report, in addition as part of Refs 120, 138 and 185.
2	Assessment is required of updates to the PCSR to reflect responses to RI-ABWR-0001	
3	Assessment is required of updates to the Master Document Submission List to reflect responses to RI-ABWR-0001	
4	Each "end user" is required to assess the adequacy of the approach to down selection of the consolidated list of nuclides defined in the RI-ABWR-0001 response to the relevant EUST subset	
5	Further evidence is required on the impact on NMCA and Zn on radioactivity, specifically in terms of loss of dosing, timings or transients	This was added to the latest revision of Ref. 18 and is discussed in Ref. 80 from a radioactivity perspective
6	Assessment is required to ensure that the impact of feedwater Iron concentration control is appropriately reflected in the safety case for UK ABWR	See Paragraph 438 and Assessment Finding AF-ABWR-RC-11
7	Evidence is required on to what extent moisture carry over at the end of cycle is expected for UK ABWR, and the impact of this on any activity transport	See Paragraph 492
8	Further evidence is required on the actual performance expected in the UK ABWR clean-up systems (CUW, SFP, CD)	See Section 4.3.5.2. and Refs18 and 19
9	Evidence is required to show that Power Suppression Testing stops fuel failures from escalating, that suitable methods are available to detect further degradation before it become significant and that such operations reduce risks SFAIRP	See Section 4.3.3.5 and Section 4.3.4.3.
10	Further evidence is required to demonstration that the ¹³¹ I shutdown relationship is valid for plants outside Japan and for the UK ABWR operating chemistry regime	This was added to the latest revision of Ref. 73
11	Further evidence is required the characterise the release of fission products during a shutdown transient	This was added to the latest revision of Ref. 73
12	Further evidence is required to justify the defined ¹⁶ N concentrations in both reactor water and steam	Further information is available to Hitachi-GE, and has been shared with ONR, but currently cannot be added to Ref. 73 due to confidentiality issues. This data supports the existing data in Ref. 73.

Annex 7

Residual Matters identified during Assessment of RI-ABWR-0001

Number	Description	Response
13	Assessment is required for where the UK ABWR defined source terms reflect into the plant design, particularly in areas where the UK ABWR values are higher	This is considered by other ONR assessment reports; Refs 119 and 185
14	Further clarification is required within the source term documentation regarding the uncertainties and reliabilities of the various derived values, particularly for the PST.	This was added to the latest revision of Ref. 73
15	Further evidence is required on iodine behaviour in the off-gas system and effects as part of RO-ABWR-0066	See Section 4.7
16	Further evidence is required for the growth of DST in UK ABWR given the intention to operate with HWC, OLNLC and DZO from first operations	This was added to the latest revision of Ref. 75
17	Further evidence is required for BRAC [BWR Radiation Assessment and Control] dose rate data for plants screened out of the UK ABWR analysis	This was added to the latest revision of Ref. 75
18	Assessment is required on the impact of RI-ABWR-0001 methodologies and approach on ORs necessary for UK ABWR (both radioactive species or precursors)	This is considered throughout this report, in particular in Section 4.3.3 and Assessment Finding AF-ABWR-RC-02

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Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-ABWR-RC-01	<p>No internally pumped Boiling Water Reactor (like UK ABWR) has ever undertaken long-term continued operations using Hydrogen Water Chemistry. To demonstrate the responsiveness of UK ABWR to Hydrogen Water Chemistry under the operating conditions chosen by a future licensee, and to address limitations in the evidence provided during GDA on this topic, the licensee shall:</p> <ul style="list-style-type: none"> • Develop and implement an adequate radiolysis and electrochemical corrosion potential model to demonstrate UK ABWR can be controlled within the limits and conditions defined in the safety case. • Produce evidence to demonstrate adequate consideration has been given to the impact of coolant flow conditions on hydrogen mixing, platinum deposition, and the resultant electrochemical corrosion potential. 	<p>Section 4.3.2.1, paras: 138, 141, 143, 147, 150, 151, 152, 155 and 204</p> <p>Section 4.3.2.4, para 3232</p>

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Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-ABWR-RC-02	Defining and justifying adequate quantitative limits and conditions (Operating Rules) to execute the necessary control over a range of chemistry parameters important to safety is fundamental. During GDA, ONR's chemistry assessment established there are numerous gaps in the chemistry-based Operating Rules identified as the outputs from the UK ABWR generic safety case; including missing parameters and inappropriate values. To address these gaps, the licensee shall review and update the UK ABWR safety case to ensure an adequate set of chemistry-based Operating Rules are defined and justified.	<p>Section 4.3.2.1, paras: 157, 166, 167, 174, 192, 197, 201, 204</p> <p>Section 4.3.2.3, paras: 264, 279</p> <p>Section 4.3.3.4, paras: 428, 440, 447, 472, 474</p> <p>Section 4.3.4.1, paras: , 526, 534, 539, 543, 549</p> <p>Section 4.3.4.2, paras: 560, 561</p> <p>Section 4.3.4.4, para: 574</p> <p>Section 4.4.2, para: 751</p> <p>Section 4.4.3, paras:., 764, 765, 768</p> <p>Section 4.4.4, para: 780</p> <p>Section4.4.5, para: 786</p> <p>Section 4.5.3, para: 805</p> <p>Section 4.5.5, para: 809</p> <p>Section 4.7.1.5, paras: 1016, 1032, 1037</p> <p>Section 4.8.2, para: 1081.</p>

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Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-ABWR-RC-03	Chemistry control options implemented during UK ABWR commissioning may have a significant impact on the longer-term safety performance of the plant, including the generation and transport of radioactivity, amongst other factors. No Boiling Water Reactor has ever commissioned for the operating chemistry proposed for UK ABWR, nor operated using this chemistry from the very first fuel cycle. In this regard, the UK ABWR generic safety case therefore presents several options which require further work, to explore their relative benefits and detriments, before a final decision can be made by the future licensee. The licensee shall review and update the relevant conclusions in the UK ABWR safety case, and supplement them using their own further work on this topic, to demonstrate chemistry control implemented during commissioning reduces relevant risks SFAIRP. In particular, this shall include a much more comprehensive justification for when to implement: hydrogen, noble metals and zinc injection.	Section 4.3.2.1, para: 187 Section 4.3.3, paras: 456, 475 Section 4.3.4.1, paras: 552, 555 Section 4.3.5.1, para: 599
AF-ABWR-RC-04	The UK ABWR operating chemistry proposed for reactor start-up operations involves injecting hydrogen at 5-10% reactor power to mitigate the risk of stress corrosion cracking. During GDA, Hitachi-GE asserted injecting hydrogen at reactor powers lower than 5-10%, so called Early Hydrogen Water Chemistry (a further risk reduction measure), is a decision for a future licensee to make. The licensee shall review relevant plant operating experience and decide whether implementing Early Hydrogen Chemistry is necessary for UK ABWR, to ensure relevant risks are reduced SFAIRP.	Section 4.3.2.1, para: 199 Section 4.3.5.1, para: 598
AF-ABWR-RC-05	During GDA, ONR's assessment of the UK ABWR generic safety case across several topics, established there is an inconsistent approach to defining reactor operating modes. The chemistry-based operating modes defined are also inconsistent with equivalent international guidelines. This could lead to inadequate chemistry control being implemented by the future plant operator. The licensee shall review and update the definitions provided for chemistry-based operating modes in the UK ABWR safety case, and apply a coherent and consistent set of terminology, to ensure future plant operations will reduce relevant risks SFAIRP.	Section 4.3.2.1, para: 202 Section 4.3.3, para: 469
AF-ABWR-RC-06	To mitigate the risk of flow accelerated corrosion, the material of construction proposed for UK ABWR Class 3 Condensate and Feedwater system main piping is carbon steel, specified with a minimum level of residual chromium. The final decision regarding the appropriate minimum level of chromium to specify will depend on licensee design choices, and operator specific choices related to materials product form, and the selection of materials suppliers. Based on all of these considerations, the licensee shall justify the minimum level of residual chromium specified, reduces relevant risks SFAIRP.	Section 4.3.2.3, paras: 281, 282, 320
AF-ABWR-RC-07	During GDA, ONR's chemistry assessment established that some of Hitachi-GE's materials selection decisions for the Heater Drain system are sensitive to one of the important assumptions underpinning their materials selection criteria. The assumption is about the minimum acceptable levels of dissolved oxygen necessary to mitigate the risk of flow accelerated corrosion. ONR concluded this assumption is not robust. Based on this conclusion, the licensee shall review, revise and update their materials selection criteria for the Heater Drain system. To demonstrate relevant risks are reduced SFAIRP, the revised criteria should be applied to justify whether more corrosion resistant materials should be selected for some parts of the Heater Drain system, where carbon steel is currently proposed.	Section 4.3.2.3, para: 295 Section 4.3.3.3.3, para: 378

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Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-ABWR-RC-08	During GDA, as a result of the proposed change to the operating chemistry for UK ABWR, changes were also made to the materials selected for the reactor water clean-up system. Detailed analyses of materials degradation threats were unavailable during GDA, meaning it was necessary for Hitachi-GE to make certain important assumptions to underpin the materials selected. Based on the more detailed materials degradation analysis that will need to be conducted by the licensee; the licensee shall re-consider the use of low alloy steel and carbon steel for reactor water clean-up system main piping, to determine whether selecting more corrosion resistant materials, reduces relevant risks SFAIRP.	Section 4.3.2.3, para: 311
AF-ABWR-RC-09	The UK ABWR generic design includes high-cobalt alloys within the Control Rod Drives and valves, where mechanical properties and reliability requirements are important to safety. However, Hitachi-GE recognise that high-cobalt alloys make a significant contribution to radioactivity within the plant, (cobalt radioisotopes dominate doses received by workers during BWR outage maintenance activities). Licensee choices and further work are needed to fully substantiate the extent of the use of high-cobalt alloys in the UK ABWR design. The licensee shall justify that retaining high-cobalt containing alloys within components that contribute to the generation of radioactivity, reduces relevant risks SFAIRP. This shall include considering relevant operating experience, testing and the mechanical properties required.	Section 4.3.3.3, para: 359, 365, 400 Section 4.3.3.7, para: 504
AF-ABWR-RC-10	During GDA, Hitachi-GE developed and partially implemented an approach to evaluate if using surface treatment techniques could reduce both the generation, and deposition of radioactivity within certain UK ABWR plant systems. These techniques can result in significant benefits. For GDA, the approach was applied to several example components. The approach was greatly simplified and did not consider a complete review of relevant operating experience (including over multiple operating cycles), nor did it consider the full extent of the associated benefits and detriments associated with each technique. The final selection of surface treatment techniques will also depend on licensee specific choices and decisions. The licensee shall justify the surface treatments to be applied to UK ABWR structures, systems or components, which make significant contributions to the release of metallic impurities, or doses to workers, reduces relevant risks SFAIRP.	Section 4.3.3.3, para: 398
AF-ABWR-RC-11	The UK ABWR generic safety case places key claims on the importance of adequate feedwater iron control for minimising radioactivity and risks to fuel integrity. During GDA, ONR's chemistry assessment established these aspects of the generic safety case, as presented, are complex, inconsistent, lack evidence, and are not objective. The licensee shall review, update and supplement the UK ABWR safety case and supporting evidence on this topic. This shall include specifically justifying the impact feedwater oxygen injection has on radioactivity, and provide a proportionate justification for the decision on which approach will be implemented during UK ABWR operations; to practically control the feedwater iron concentration and ensure relevant risks are reduced SFAIRP.	Section 4.3.3.4, paras: 438, 442, 443, 474 Section 4.3.4.2, para: 561

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Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-ABWR-RC-12	During GDA, ONR's chemistry assessment established there is a specific gap in the UK ABWR generic safety case. There is no consideration of how shutting the reactor down with elevated levels of radioactivity contained within the reactor coolant, may impact on maintaining adequate levels of safety. Chemistry controls during shutdown assume "normal" levels of radioactivity. Other plants have specific controls and measures in place when radioactivity is elevated, to minimise effects which include the potential transfer of radioactivity to the turbine, and therefore the impact on worker doses. The licensee shall review and update the UK ABWR safety case to justify that the chemistry controls in place during a shutdown are adequate to account for such circumstances, and whether additional measures need to be adopted, to ensure relevant risks are reduced SFAIRP.	Section 4.3.3, para: 463
AF-ABWR-RC-13	During GDA, Hitachi-GE changed the operating chemistry proposed for UK ABWR. Considering this change, it is important to develop adequate chemical injection systems to control "new" chemistry parameters important to safety, within the limits and conditions (Operating Rules) defined in the safety case. The licensee shall justify that the specific design of the UK ABWR chemical injection system (including the hydrogen oxygen injection system, noble metal injection system and zinc injection system), enables the UK ABWR operating chemistry to be controlled within the limits and conditions defined in the safety case.	Section 4.3.5.1, paras: 585, 586, 590, 593, 602 Section 4.3.5.4, para: 680
AF-ABWR-RC-14	The chemical clean-up performance delivered by the reactor water clean-up and fuel pool cooling clean-up systems, is vital for removing radioactivity from the plant during normal operations, to ensure worker doses are reduced SFAIRP. To adequately substantiate the clean-up performance of these systems and address the requirement to demonstrate worker doses are reduced SFAIRP, the licensee shall review and update the UK ABWR safety case to address limitations in the level of detail and justifications provided on these matters during GDA. This shall include all operating modes and cover the removal of soluble and insoluble forms of radioactivity and impurities.	Section 4.3.5.2, paras: 612, 613, 615, 616, 618 Section 4.3.5.2, paras: 635, 636 Section 4.4.4., para: 778
AF-ABWR-RC-15	Considering the UK ABWR generic safety case claims, and the importance of being able to determine the UK ABWR operating chemistry's effectiveness at mitigating materials degradation risks, the licensee shall justify that suitable and sufficient locations have been included in the UK ABWR design to adequately monitor for: <ul style="list-style-type: none"> • Electrochemical Corrosion Potential; and • The deposition and distribution of platinum. The monitoring locations selected shall consider the requirement to undertake feedwater hydrogen benchmarking tests throughout the operating life of the plant. This shall include the need to validate the radiolysis and electrochemical corrosion potential models used to demonstrate the plant is operating within the limits and conditions defined in the UK ABWR safety case, throughout its operating lifetime.	Section 4.3.5.3, paras: 669, 674

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Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-ABWR-RC-16	There were limitations in the information provided during GDA and gaps identified by ONR's chemistry assessment in both the design, and justification, of the UK ABWR Sampling and Monitoring system. To address these gaps, the licensee shall provide an adequate justification to demonstrate the UK ABWR Sampling and Monitoring system will enable the "new" proposed operating chemistry to be controlled within the limits and conditions defined in the safety case. This should include all modes of operation and severe accident conditions, and provide a robust demonstration that relevant risks are reduced SFAIRP.	Section 4.3.5.3, paras: 646, 652, 657, 660, 676, 678, Section 4.4.4, para: 779 Section 4.7.6.1, paras: 1058, 1068
AF-ABWR-RC-17	The UK ABWR generic design includes a reactor pressure vessel head spray line. The line has been modified to include a vent to prevent the accumulation and potential combustion of hydrogen and/or oxygen. The reactor pressure vessel head spray line performs no claimed safety functions and its use is not universal across other Boiling Water Reactor plants. The licensee shall review their detailed design, outage plans and procedures to decide whether retaining the reactor pressure vessel head spray line within the UK ABWR design reduces relevant risks SFAIRP.	Section 4.3.6.3, para: 713, Section 4.3.6.6, para: 730
AF-ABWR-RC-18	During GDA, ONR's chemistry assessment has established that some of Hitachi-GE's assumptions regarding the accumulation of radiolysis gases (hydrogen and oxygen), may not be suitably conservative for all operating modes, and also to account for the implementation of Hydrogen Water Chemistry. Based on these conclusions, and considering their detailed design and reactor operating strategy (including potential prolonged reduced power operations), the licensee shall review and update the UK ABWR safety case, to provide safety analysis for radiolysis gas accumulation which is performed on a demonstrably, suitably conservative basis, and show that relevant risks are reduced SFAIRP.	Section 4.3.6.4, paras: 724, 725
AF-ABWR-RC-19	The UK ABWR generic safety case for the spent fuel pool assumes pure water is used as the storage medium, under adequately controlled conditions. During GDA, ONR's chemistry assessment established there were some limitations in the associated evidence provided; including a demonstration of the impact abnormal chemistry operating conditions may have on the integrity of the spent fuel itself, and spent fuel pool structures. The licensee shall review and update the UK ABWR safety case to provide evidence to demonstrate the impact deteriorated spent fuel pool water quality has on the integrity of the stored fuel itself, and/or the spent fuel pool structures, system and components. This shall include a proportionate justification to show adequate controls are in place to reduce relevant risks SFAIRP.	Section 4.4.2, para: 748, 755, 758 Section 4.4.4, para: 770
AF-ABWR-RC-20	Normal practice for spent fuel pool clean-up systems is to select entirely corrosion resistant materials. Some large sections of the UK ABWR spent fuel pool clean-up system main piping uses carbon steel. The licensee shall review and update the UK ABWR safety case to demonstrate that using carbon steel for sections of spent fuel pool clean-up system main piping is not detrimental to water quality, doses to workers and the generation of radioactive waste, and show that relevant risks are reduced SFAIRP.	Section 4.4.2, para: 757

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Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-ABWR-RC-21	The UK ABWR generic safety case concludes including a “guard bed” in the off-gas system design reduces relevant risks SFAIRP (and by definition is therefore reasonably practicable to implement). Despite this conclusion, the decision to include a “guard bed” is left for a licensee to make, but evidence is provided to demonstrate the generic design does not foreclose this is an option. The licensee shall review and update the UK ABWR safety case to provide the necessary information to demonstrate a “guard bed”, or an equivalent measure which delivers the same function, has been implemented in the UK ABWR off-gas system design.	Section 4.6.2.3, paras: 856, 860, 862 Section 4.6.3, para: 863
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.	Section 4.7.3.1, paras: 898, 915
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.	Section 4.7.3.2, paras: 908, 909, 910, 912
AF-ABWR-RC-24	The UK ABWR generic safety case makes important assumptions about the claimed efficiency of the scrubbing effect which the suppression pool provides for the radioactivity released during a severe accident. The approach effectively assumes that radioactive iodine is all released in the physical form of an aerosol. This overestimates the removal of volatile forms of radioactive iodine, and therefore potentially underestimates the potential radiological consequences. The licensee shall review and update the UK ABWR safety case and provide a proportionate justification to show that these aspects of the analysis adequately represent the physical and chemical processes taking place. This shall include considering the sensitivity and uncertainty in such events, and an examination of the overall impact on the postulated radiological consequences, to demonstrate relevant risks are reduced SFAIRP.	Section 4.7.4.3, paras: 935, 936 Section 4.7.4.9, para 1005

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Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF-ABWR-RC-25	During GDA, ONR's chemistry assessment of the UK ABWR generic safety case established the postulated radiological consequences of accidents are sensitive to several assumptions. These include the inventory of halogenated cables used within the Primary Containment Vessel, and the type and specification of coatings applied to the Suppression Pool. Both of these can directly influence the inventory of radioactive iodine which may be released during postulated accident scenarios. The generic safety case recognises that detailed design information, licensee choices and other further work is needed before the overall importance of these assumptions is clear. The licensee shall review and update the UK ABWR safety case to provide a proportionate demonstration that all reasonably practicable measures to minimise the inventory of halogenated cabling, and specifying a suitable Suppression Pool coating, have been taken, to be able to show that relevant risks are reduced SFAIRP.	Section 4.7.4.3, paras: 949, 964, 974
AF-ABWR-RC-26	The UK ABWR generic safety case captures the requirement for the design to include a Suppression Pool pH control system, to be used during, or following, some postulated accident scenarios. The aim of the system is to retain radioactive iodine and potentially other forms of radioactivity, within the suppression pool, to ensure it is unavailable for potential release to the environment during or following an accident. The final design of this system will be significantly influenced by licensee decisions on how this technology is implemented in the design. The licensee shall review and update the UK ABWR safety case to provide a proportionate demonstration to justify the final design of the UK ABWR suppression pool pH control system reduces relevant risks SFAIRP. This shall include a suitable suite of supporting analyses which adequately considers all relevant physical and chemical processes taking place within the reactor containment.	Section 4.7.4.3, paras: 950, 951, 958, 959, 961, 964, 972, 973, 983 Section 4.7.5.3, paras: 1021 Section 4.7.4.9, para 1005

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Minor Shortfalls

Minor Shortfall Number	Minor Shortfall	Report Section Reference
MS-ABWR-RC-01	In the UK ABWR generic safety case, the Topic Report containing the necessary evidence to underpin the decision to select On-line NobleChem™ as part of the at-power operating chemistry, is titled <i>Noble Metal Chemical Addition</i> . There are subtle, yet very important differences between these two technologies. The licensee should review and update the UK ABWR safety case to ensure the correct terminology is adopted throughout, when referring to noble metal chemistry.	Section 4.3.1.1, para: 84 Section 4.3.1.4, para: 113
MS-ABWR-RC-02	The safety case claims made on commissioning, start-up and shutdown chemistry in Chapter 23 of the generic UK ABWR Pre-construction Safety Report are inconsistent with the claims presented in the related Topic Reports, in the lower tiers of the hierarchy of UK ABWR generic safety case documentation. The licensee should revise and update the UK ABWR safety case to ensure a consistent set of claims are presented.	Section 4.3.1.2, para: 94 Section 4.3.1.3, para: 105
MS-ABWR-RC-03	The UK ABWR generic safety case makes the assumption that materials degradation risks for flow accelerated corrosion and erosion-corrosion are equivalent. Based on this assumption, no separate risk ranking for erosion-corrosion is performed. These degradation mechanisms are similar, but not the same. The licensee should revise and update the UK ABWR safety case to include a separate risk ranking, or similar risk assessment, for erosion-corrosion.	Section 4.3.2.3, para: 253, 257,320
MS-ABWR-RC-04	The UK ABWR generic safety case does not utilise any operating experience from the J-ABWR reference plant, as evidence to support the key safety case claims made on the operating chemistry's ability to mitigate the risk of flow accelerated corrosion, nor in support of the corresponding risk assessment. The licensee should revise and update the UK ABWR safety case to draw upon the findings of any flow accelerated corrosion inspection and monitoring programmes performed for the J-ABWR reference plant.	Section 4.3.2.3, para: 254, 257, 320
MS-ABWR-RC-05	¹⁶ N is the main radiation source during at-power operating conditions for UK ABWR. During GDA, Hitachi-GE was unable to reference plant operating experience to justify the expected concentrations of this nuclide, due to confidentiality issues. The licensee should review and update the UK ABWR safety case to include additional evidence to justify the defined ¹⁶ N concentrations in both reactor water and steam.	Section4.3.3.1, para: 342 Section 4.3.3.7, para: 504
MS-ABWR-RC-06	The UK ABWR generic safety case contains several gaps in the information provided to justify the claimed chemical clean-up performance of the Condensate Purification System. The system provides the primary means for producing main feedwater at the required flow and water quality, to secure continued, normal, at-power reactor operations. The licensee should review and update the UK ABWR safety case to address the gaps identified during GDA.	Section 4.3.5.1, para: 595 Section4.3.5.2, paras: 627, 633, 635, 636 Section 4.3.5.4, para: 680

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Minor Shortfalls

Minor Shortfall Number	Minor Shortfall	Report Section Reference
MS-ABWR-RC-07	During GDA, the UK ABWR generic design of the spent fuel pool cooling and clean-up system was modified to add an extra train of cooling. As part of this modification extra clean-up capacity was not added to the system. Only one train of the system is therefore capable of cleaning-up the spent fuel pool water. The licensee should review and update the UK ABWR safety case to justify whether there are reasonably practicable measures that can be applied to allow operation of the existing clean-up provisions using both cooling trains, or using other similar means.	Section 4.4.4, para: 771 Section 4.4.5, para: 785
MS-ABWR-RC-08	The quantities and speciation of radioactivity postulated to be released from fuel during accident conditions, derived in the UK ABWR generic safety case for the beyond design basis analysis, assumes Best Estimate radionuclide inventory values. This means the contribution damaged fuel makes to the coolant radioactivity levels is not included. The licensee should review and update the UK ABWR safety case to provide evidence to demonstrate the sensitivity of the consequences of such faults, to the use of the more conservative, Design Basis radionuclide inventory values.	Section 4.7.4.2, para: 923 Section 4.7.7, para: 1070
MS-ABWR-RC-09	The licensee should review and update the relevant aspects of the UK ABWR safety case which deal with the claimed impact of suppression pool scrubbing on radioactivity. This update should provide a clear, consistent and transparent demonstration that the assumptions used within the safety analysis are adequate.	Section 4.7.4.3, para: 936 Section 4.7.7, para: 1070
MS-ABWR-RC-10	The UK ABWR generic design includes a Filtered Containment Venting System. The detailed design of this system was not fixed during GDA. This meant the information contained in the generic safety case regarding the system's efficiency and the chemical and physical radioactivity removal processes taking place, were not specific to the UK ABWR design and safety case. The information was largely based on vendor data and theoretical studies. The licensee should review and update these aspects of the UK ABWR safety case to ensure the supporting analysis adequately represents the physical and chemical processes taking place, and that the information is specific to UK ABWR postulated conditions.	Section 4.7.4.3, para: 990 Section 4.7.7, para: 1070
MS-ABWR-RC-11	In the analysis of spent fuel pool accidents, the UK ABWR generic safety case assumes, and takes credit for, the spent fuel having small amounts of fuel deposits. This approach may underestimate the impact spalled fuel deposits have on the radiological consequences following postulated spent fuel pool accidents. The licensee should review and update the UK ABWR safety case to justify that the treatment of spalled fuel deposits in a loss of spent fuel pool cooling fault has been considered in a suitably conservative manner.	Section 4.7.4.3, para: 1003 Section 4.7.7, para: 1070
MS-ABWR-RC-12	The UK ABWR generic safety case contains ambiguities and inconsistencies in its presentation of the boron requirements for the Stand-by Liquid Control system. The licensee should review and update the UK ABWR safety case to provide a consistent and coherent set of nuclear safety requirements associated with this parameter.	Section 4.7.5.5, para: 1028 Section 4.7.7, para: 1070